Safety Evaluation Report

NUREG-75/080

U. S. Nuclear Regulatory Commission

related to construction of

Wolf Creek Generating Station, Unit No. 1

Kansas Gas & Electric Company
Kansas City Power & Light Company

Office of Nuclear Reactor Regulation

Docket No. STN 50-482

September 1975

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SAFETY EVALUATION REPORT

BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

U.S. NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF

KANSAS GAS & ELECTRIC COMPANY

AND

KANSAS CITY POWER & LIGHT COMPANY

WOLF CREEK GENERATING STATION UNIT 1

DOCKET NO. STN 50-482

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1.0 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

The Kansas Gas & Electric Company and the Kansas Power & Light Company (applicants) filed with the Nuclear Regulatury Commission (Commission) an application, docketed on May 17, 1974 (Docket No. STN-50-482), for a license to construct and operate the proposed Wolf Creek Generating Station, Unit 1 (Wolf Creek plant or facility) which will be located in Coffey County, Kansas approximately 28 miles east-southeast of Emporia, Kansas. The application was submitted and accepted for review under the Commission's standardization policy statement of March 5, 1973.

The Kansas Gas & Electric Company and the Kansas City Power & Light Company are two of the five utilities who have joined together under the acronym SNUPPS (Standardized Nuclear Unit Power Plant System) to submit applications for a standard plant design for review under the Commission's standardization policy using the duplicate plant option, described in Appendix N to the Commission's regulations in Part 50 of Title 10 of the Code of Federal Regulations (10 CFR Part 50), "Licensing of Production and Utilization Facilities." This option allows for a simultaneous review of the safety-related parameters of a limited number of duplicate plants which are to be constructed within a limited time span at a multiplicity of sites.

The other SNUPPS applications submitted for review are the Callaway Plant, Units 1 and 2 (Docket Nos. STN-50-483 and STN-50-486), to be located in Callaway County, Missouri (submitted by the Union Electric Company), the Sterling Power Project (Docket No. STN-50-485), to be located in Cayuga County, New York (submitted by the Rochester Gas & Electric Corporation), and the Tyrone Energy Park (Docket No. STN-50-484), to be located in Dunn County, Wisconsin (submitted by the Northern States Power Company).

A Preliminary Safety Analysis Report (PSAR) was submitted with the Wolf Creek application consisting of the SNUPPS PSAR (describing those portions of the Wolf Creek plant which are identical to the other three SNUPPS plants and which is incorporated into the other three SNUPPS applications) and the Wolf Creek Site Addendum Report (describing the specific site-related and applicant-related portions for the Wolf Creek application). The SNUPPS PSAR incorporates by reference the Reference Safety Analyysis Report (RESAR-3, Consolidated Version plus appropriate parts of Amendment 6 to RESAR-3) which was prepared by the Westinghouse Electric Corporation and addresses those portions of the SNUPPS plants for which the Westinghouse Electric Corporation has design responsibility. The information in the SNUPPS PSAR was supplemented by Revisions 1 through 9. The information in the Wolf Creek Site Addendum Report was supplemented by Revisions 1 through 8. Copies of these reports and revisions are available for public inspection at the U. S. Nuclear Regulatory Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. 20555, and at the Office of the County Clerk, Coffey County Courthouse, Burlington, Kansas 66839.

This Safety Evaluation Report summarizes the results of the technical evaluation of the proposed Wolf Creek plant design performed by the Commission's staff and delineates the scope of the technical matters considered in evaluating the radiological safety aspects of the Wolf Creek facility. A separate Safety Evaluation Report will be prepared to summarize the results of the Commission's technical evaluation for each of the other three SNUPPS plants. However, the Commission's technical evaluation of the Wolf Creek plant, as presented in this Safety Evaluation Report, can be considered as generally representative for the other SNUPPS applications. Aspects of the environmental impact considered in the review of the Wolf Creek facility, in accordance with 10 CFR Part 51 of the Commission's regulations, "Licensing and Regulatory Policy and Procedures for Environmental Protection," were discussed in the Commission's Draft Environmental Statement issued in July 1975.

Subject to the favorable resolution of the outstanding issues discussed herein and summarized in Section 1.ꞏ of this report, we will be able to conclude that the Wolf Creek plant can be constructed and operated as proposed without endangering the health and safety of the public. Our detailed conclusions are presented in Section 21.0 of this report.

The review and evaluation presented in this report is only the first stage of a continuing review by the Commission's staff of the design, construction, and operating features of the Wolf Creek facility. Construction will be accomplished under the surveillance of the Commission's staff. Prior to the issuance of an operating license, we will review the final design of the facility to determine that all of the Commission's safety requirements have been met. The facility may then be operated only in accordance with the terms of the operating license and the Commission's regulations under the continued surveillance of the Commission's staff.

1.2 General Plant Description

1.2.1 Standard Plant Portion

The nuclear steam supply system for the Wolf Creek plant will consist of a pressurized water reactor and a four-loop reactor coolant system designed for a core power output of 3411 thermal megawatts. The reactor core will be composed of uranium dioxide pellets enclosed in Zircaloy tubes with welded end plugs. The fuel tubes will be grouped and supported in assemblies. The reactor core will initially consist of three regions each containing fuel of a different enrichment of uranium-235. Water will serve as both the moderator and the coolant and will be circulated through the reactor vessel and core by four coolant pumps. The water, heated by the reactor, will flow through four steam generators 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 will provide a surge chamber and a water reserve to accommodate reactor coolant volume changes during operation.

The nuclear steam supply system will be housed in a containment structure. The containment will consist of a steel-lined, prestressed, post-tensioned concrete structure. The

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containment structure, including its penetrations, will be designed to safely confine, within the leakage limit of the containment, the radioactive material that could be released in the event of an accident. An auxiliary building, to be located adjacent to the containment structure, will house the radioactive waste treatment systems, components of engineered safety features, and various related auxiliary systems. The fuel handling building, also to be located adjacent to the containment structure, will house a spent fuel pool and new fuel storage facility.

The steam and power conversion system will be designed to remove the heat energy from the reactor coolant in the four steam generators and convert it to electrical energy.

The reactor will be controlled by control rod movement and regulation of the boric acid concentration in the reactor coolant. The control elements, whose drive shafts will penetrate the top head of the reactor vessel, will be moved vertically within the core by individual control rod drives. A reactor protection system will be provided that automatically initiates appropriate action whenever a condition monitored by the system approaches pre-established limits. 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 emergency core cooling system will consist of accumulator tanks and high pressure injection and low pressure injection systems, with provisions for recirculation of the borated coolant after the end of the injection phase. Various combinations of these systems will assure core cooling for the complete range of postulated coolant pipe break sizes.

A containment spray system will provide borated water containing sodium hydroxide to remove heat and radioactive iodine in the event of an accidental coolant release. A containment ventilation system, which will include a containment fan cooling system consisting of four fan coolers located within the containment structure, will serve to maintain normal plant operation. During accident conditions, the containment fan coolers are capable of maintaining the containment pressure below the containment design pressure even in the event of a single active failure in either the spray system or the fan cooling system.

1.2.2 Portion of Plant Outside the Scope of the Standard Plant Design

The circulating water system will be used to discharge unusable heat from the steam and power conversion system to an onsite cooling lake. The safety-related portion of the ultimate heat sink will be the water impounded by a submerged seismic Category I dam within the cooling lake.

The facility will be provided with electrical power from two independent offsite power sources and independent and redundant onsite emergency power supplies capable of supplying power to the engineered safety features. Portions of the onsite power supplies will be included in the standard plant design scope.

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1.3 Comparison with Similar Facility Designs

The principal features of the design of the Wolf Creek plant are similar to those we have evaluated and approved previously for other nuclear power plants. The nuclear steam supply system is, for example, comparable to that of other plants which utilize the four-loop configuration described in RESAR-3, Consolidated Version, such as Millstone Point Nuclear Power Station Unit No. 3 (Docket No. 50-423) and Comanche Peak Steam Electric Station Units Nos. 1 and 2 (Docket Nos. 50-445 and 50-446). To the extent feasible and appropriate, we have made use of our previous evaluations of these plants in conducting our review of the Wolf Creek plant. 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 Nuclear Regulatory Commission's Public Document Room at 1717 H Street, N. W., Washington, D. C.

1.4 Identification of Agents and Contractors

The Kansas Gas & Electric Company will be responsible for the design, construction and operation of the Wolf Creek plant. The Kansas Gas & Electric Company and the Kansas City Power & Light Company have joined together with the other three SNUPPS utilities to form a SNUPPS Project Organization (with representation from each utility) to manage the design and procurement of the standard portions of the four SNUPPS plants.

The SNUPPS Project Organization, acting on behalf of the SNUPPS utilities, has retained the Bechtel Power Corporation to provide architect-engineer services, including procurement, for the standard portions of the SNUPPS plants. The Westinghouse Electric Corporation has been contracted to design, manufacture and deliver to the appropriate site the nuclear steam supply system and the initial core for each of the five SNUPPS units. The turbine generators will be purchased from the General Electric Company.

The applicants have retained Sargent & Lundy as an architect-engineer to provide engineering and technical services for those portions of the Wolf Creek facility which are not included in the standard portion of the Wolf Creek plant. The applicants will also use consultants as required in specialized areas, for example Dames and Moore in areas relating to meteorology, demography, hydrology, seismology and geology, and Industrial Bio-Test Laboratories, Inc. to perform environmental studies.

1.5 Summary of Principal Review Matters

Our technical review and evaluation of the information submitted by the applicants considered the principal matters summarized below.

We reviewed the population density and use characteristics of the environs of the site, and the physical characteristics of the site, including seismology, meteorology, geology, and hydrology, to determine that these characteristics had been determined adequately and had been and will be given appropriate consideration in the design of the Wolf Creek plant, and that the characteristics of the site were in accordance with the Commission's siting criteria in 10 CFR Part 100, "Reactor Site Criteria," taking into consideration the design of the facility including the proposed engineered safety features.

We reviewed the design, fabrication, construction, and testing criteria, and expected performance characteristics of the structures, systems, and components important to safety to determine that they are in accord with the Commission's General Design Criteria, Quality Assurance Criteria, Regulatory Guides, and other appropriate codes and standards, and that any departure from these criteria, codes, and standards has been identified and justified.

We considered the response of the facility to certain anticipated operating transients and postulated accidents. We considered the potential consequences of a few highly unlikely postulated accidents (design basis accidents), and performed conservative analyses of these accidents to determine that the calculated potential offsite doses that might result in the very unlikely event of their occurrence would not exceed the Commission's guidelines for site acceptability given in 10 CFR Part 100.

We evaluated the design of the systems provided for the control of the radioactive effluents from the facility to determine that these systems can control the release of radioactive wastes within the limits of the Commission's regulations, 10 CFR Part 20, "Standards for Protection Against Radiation," and that the equipment to be provided will be capable of being operated by the applicants in such a manner as to reduce radioactive releases to levels that are as low as practicable within the contemplation of the Commission's regulations, 10 CFR Part 50, "Licensing of Production and Utilization Facilities."

We are evaluating the financial data and information provided by the applicants as required by the Commission's regulations, Section 50.33(f) of 10 CFR Part 50 and Appendix C to 10 CFR Part 50, to determine the financial qualifications of the applicants to design and construct the proposed facility. The conclusions of our evaluation of this matter will be reported in a supplement to this report.

1.6 Modifications as a Result of Staff Review

During the review of the Wolf Creek application, numerous meetings were held with representatives of the applicants and their contractors and consultants to discuss the design of the facility and the technical material submitted in the application. A chronological listing of the meetings and other significant events in our review of the application is given in Appendix A to this report. During the course of the review the applicants proposed, or we requested, a number of technical and administrative changes. These changes are described in various amendments to the application. We have listed below the more significant modifications that have resulted from our review. Included are references to the sections of this report where each matter is discussed more fully.

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(1) Modification of the cooling lake main dam design to include an auxiliary spillway so as to provide adequate freeboard on the dam for a postulated probable maximum flood (Section 2.4.2).

(2) Modification of the safe shutdown earthquake intensity and resulting ground acceleration for the Wolf Creek site (Section 2.5.4).

(3) Modification of the design criteria for the cooling lake main dam and saddle dams to safety withstand a postulated operating basis earthquake (Section 2.6).

(4) Modification of the roof design of the seismic Category I structures to properly account for the vertical velocities of tornado generated missiles (Section 3.5).

(5) Modification of the design criteria for protection against postulated high energy line breaks outside containment (Section 3.6.2).

(6) Modification of the seismic design criteria to account for relative displacements between component support points or piping support points in an acceptable manner (Section 3.7.2).

(7) Installation of a loose parts monitoring system (Section 5.4).

(8) Installation of a gross failed-fuel monitoring system (Section 5.5).

(9) Modification of the filter system design for the auxiliary building to include safety grade filters which are effective in removing iodine (Section 6.2.3).

(10) Modification of the control room breathing apparatus capacity (Section 6.4).

(11) Modification of the anticipatory trips portion of the reactor trip system design criteria to meet the requirements of the Institute of Electrical and Electronic Engineers Standard 279-1971 (Section 7.2).

(12) Modification of the criteria for interrupting devices to meet the recommendations of Regulatory Guide 1.75 (Section 8.4).

(13) Modification of the spent fuel pool cooling system to meet seismic Category I requirements (Section 9.2.3).

(14) Modification of the fuel transfer tube access to reduce radiation exposures to personnel (Section 12.2).

(15) Modification of the Bechtel Power Corporation Quality Assurance Program to include review and approval of inspection procedures and instructions (Section 17.3).

1.7 Requirements for Future Technical Information

The applicants have incorporated by reference Section 1.5 of the RESAR-3, Consolidated Version, to identify certain development programs applicable to the Wolf Creek plant. These programs, that are aimed at verifying the nuclear steam supply system design and confirming the design margins, are all being conducted by the Westinghouse Electric Corporation. The objectives, schedules for completion, and current results are summarized in RESAR-3, Consolidated Version. In addition, the Westinghouse Electric Corporation is conducting an integrated test program to confirm the design margins associated with the 17 x 17 fuel assembly design, which is discussed further in Section 4.1 of this Safety Evaluation Report.

We have concluded that the applicants have identified and will perform development tests necessary for verification of the design and safe operation of the Wolf Creek plant on a timely schedule, and that if the results of any of this work are not successful, appropriate alternate actions, as discussed in Section 1.5 of RESAR-3, Consolidated Version, or restrictions in operation can be imposed to protect the health and safety of the public.

1.8 Outstanding Issues

We have identified certain outstanding issues in our review which require that the applicants provide additional information to confirm that the proposed design will meet our requirements. These items are summarized below and are discussed further in the indicated sections of this report.

(1) Additional analysis to confirm that the plant's emergency core cooling system design meets the requirements of the Final Acceptance Criteria (Section 6.3.3).

(2) Information on periodic testing of the engineered safety features actuation system (Section 7.3).

We have also identified certain issues where we are currently reviewing information provided by the applicants, and where our review is not yet complete. These items are summarized below and are discussed further in the indicated sections of this report.

(1) Evaluation of containment temperature and pressure response to a spectrum of main steam line breaks (Section 6.2.1).

(2) Evaluation of the plant design to withstand the effects of anticipated transients without scram (Section 7.2).

(3) Evaluation of the design for manually-controlled, electrically-operated valves to meet the single failure criterion (Section 7.3).

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(4) Information on design criteria and procedures for fire stops and seals to evaluate the fire detection and protection system for electrical cables (Section 8.5).

(5) Reevaluation of the plant design to demonstrate compliance with the new Appendix I to 10 CFR Part 50, which became effective June 4, 1975 (Sections 11.2 and 11.3).

(6) Evaluation of the financial qualifications of the applicants (Section 20.0).

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2.0 SITE CHARACTERISTICS

2.1 Geography and Demography

2.1.1 Site Location and Description

The proposed site of the Wolf Creek Generating Station consists of about 10,000 acres of land located in Hampden Township, Coffey County, Kansas, about 28 miles east-southeast of Emporia, Kansas, 75 miles southwest of Kansas City, Kansas, and 100 miles northeast of Wichita, Kansas. The geographic location of the Wolf Creek site and the other proposed sites for the SNUPPS plants are shown on Figure 2.1. Figure 2.2 shows the location of the Wolf Creek plant site within the property lines identified by the applicants, and with respect to the John Redmond Reservoir and the nearby communities of Burlington, Sharpe, and New Strawn, Kansas.

The applicants have defined an exclusion area which consists of the land surrounding the planned location of the plant structures out to a radious of 1200 meters (3936 feet) measured from the center of the reactor building. Except for several public roads which presently traverse the designated exclusion area, the applicants own or control all of the area within the designated exclusion area including the mineral rights. The applicants have provided reasonable assurance that these public roads which traverse the exclusion area can and will be abandoned prior to the start of construction. A low population zone with a radius of 4000 meters (2.5 miles) has been selected by the applicants. Figure 2.2 illustrates the exclusion area and low population zone with respect to the applicants' property lines and the site environs. The applicants state that the population center, as defined in 10 CFR Part 100, or city closest to the site with a population greater than 25,000 persons, is Topeka, Kansas, 53 miles north of the site. Topeka had a 1970 population of about 155,000. Emporia, Kansas, which is about 28 miles west-northwest of the site, had a 1970 population of about 23,000 and is expected to exceed 25,000 early in the plant's estimated 40 year life. Therefore, we consider Emporia to be the population center. The distance from the outer boundary of the 2.5-mile low population zone to the nearest boundary of Emporia is well in excess of the minimum population center distance of one and one-third times the low population zone distance, as required by 10 CFR Part 100.

2.1.2 Population and Population Distribution

The applicants state that the population within the designated low population zone is 101 persons and project that by 1980 the population will have decreased to about 40 persons due to construction of the plant cooling lake. Figure 2.3 shows the 1970 cumulative population surrounding the Wolf Creek site out to a distance of 50 miles. For comparison, the population in a moderately populated area with a density of 500 people per square mile is also shown. The comparison shows that the area of the Wolf Creek site is not heavily populated.

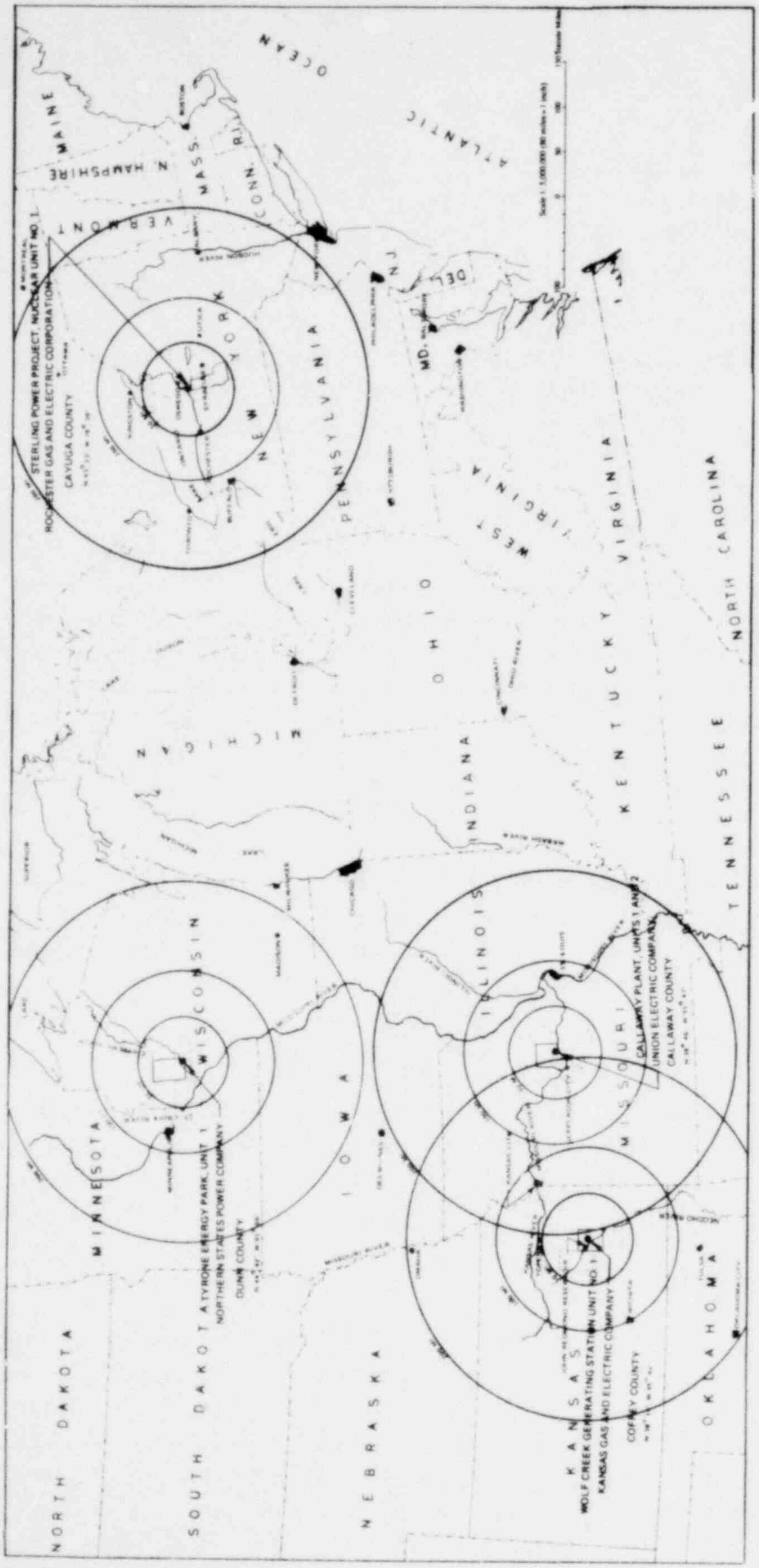

FIGURE 2.1 SNUPPS SITE LOCATIONS

POOR ORIGINAL

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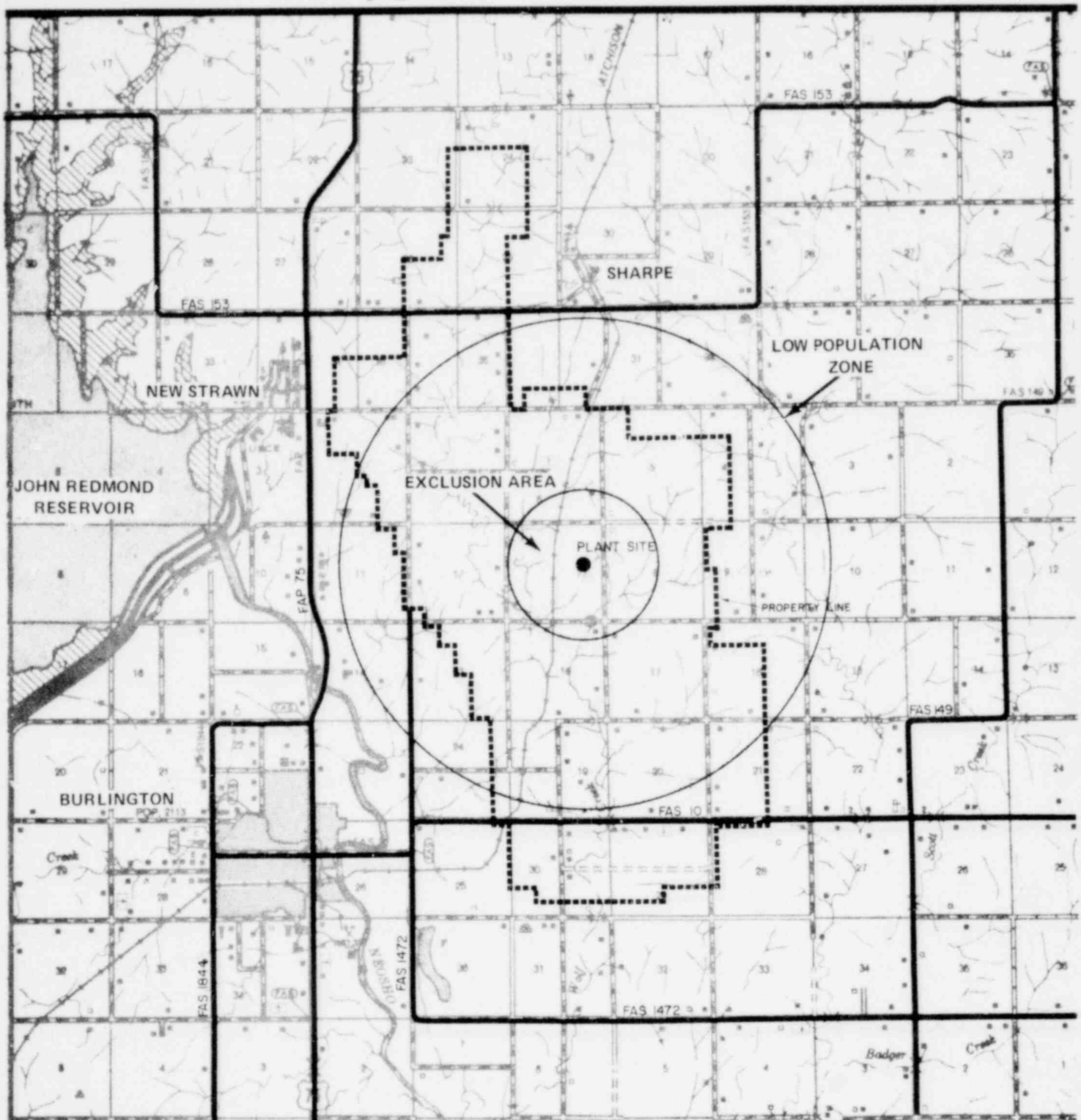

Figure 2.2 LOCAL SITE VICINITY

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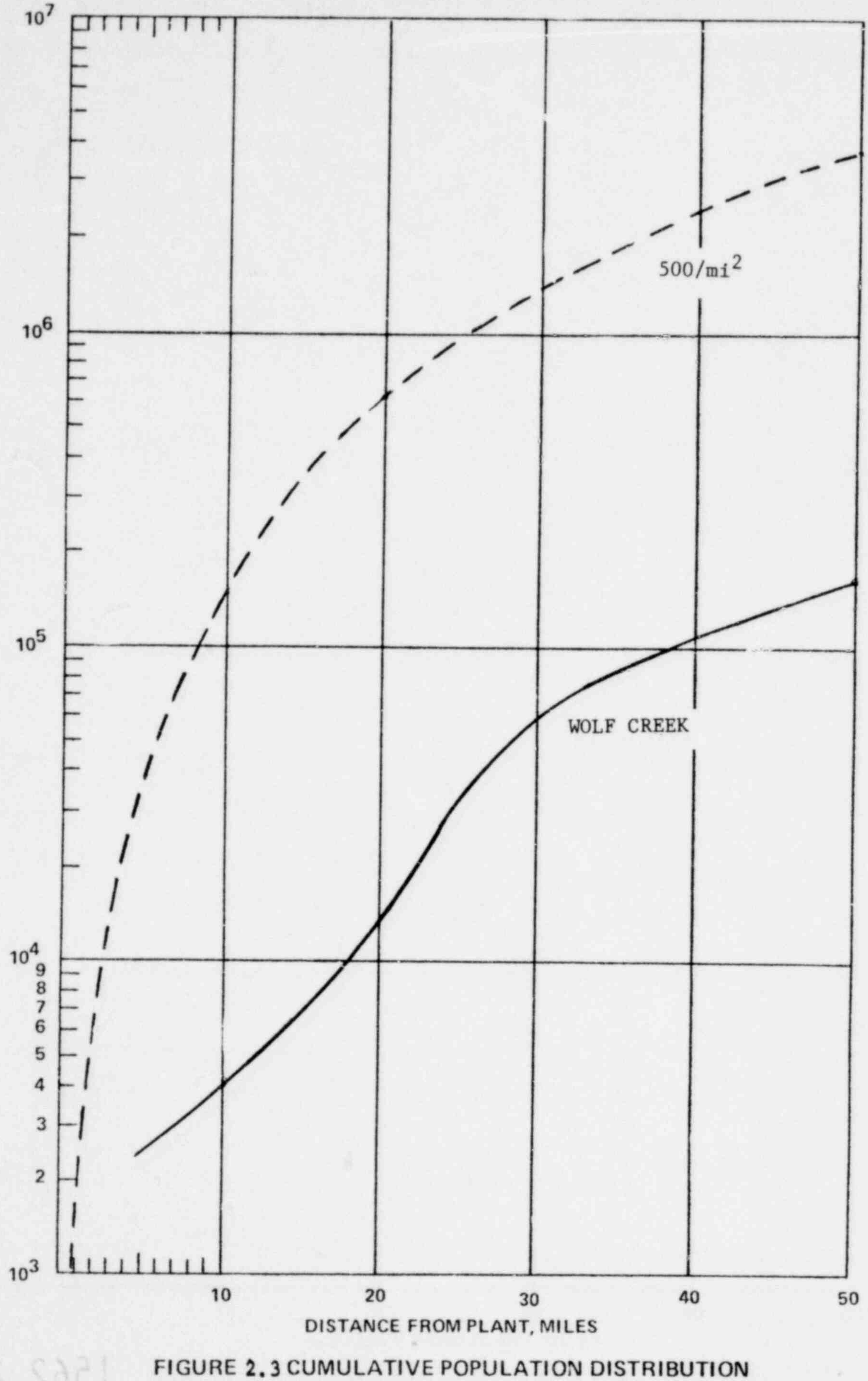

FIGURE 2.3 CUMULATIVE POPULATION DISTRIBUTION

We have compared the applicants' count of the total 1970 population within 50 miles of the site with an independent count based on the Bureau of the Census data for the 1970 census. The difference between the two counts is approximately two percent and, therefore, we conclude that the applicants' population count is adequate.

The applicants have projected a population growth of 19 percent within a 50 mile radius of the proposed plant during the period 1970 to 2020. We have compared this projected growth with population projections made by the Bureau of Economic Analysis, U.S. Department of Commerce, for Economic Areas 109, 110, 111, and 116 which surround Coffey County, Kansas, as shown on Figure 2.4. The Bureau projects a population growth of 23 percent for these combined Economic Areas during the period 1970 to 2020. We conclude, therefore, that the applicants' projected population growth compares reasonably well with independent projections of population growth made for that area of the United States by the Bureau of Economic Analysis, and is acceptable.

2.1.3 Conclusions

On the basis of the 10 CFR Part 100 definitions of the exclusion area, low population zone, and population center, and the calculated radiological consequences of postulated design basis accidents (presented in Section 15.0 of this report), we conclude that the exclusion area, low population zone, and population center distances for the proposed Wolf Creek plant meet the requirements of 10 CFR Part 100 and are acceptable.

2.2 Nearby Industrial, Transportation, and Military Facilities

There are no significant manufacturing facilities within five miles of the planned location of the plant. Two firms engaged in the manufacture of boats and fibre glass are located at New Strawn, Kansas about 3.3 miles northwest of the site. These firms employ about 40 persons. There are no chemical plants or significant chemical storage facilities within five miles of the site. There are facilities for the storage of petroleum, petroleum products, and fertilizer at Sharpe and Burlington, Kansas, located about 3.2 miles north of the site and 4.7 miles southwest of the site, respectively. The largest petroleum product storage facility is the 90,000- gallon propane storage tank owned by the Phillips Pipeline Company, located at Sharpe, 3.2 miles north of the proposed plant site. The applicants have analyzed the consequences of a postulated failure of the tank and determined that a flammable cloud of propane and air mixture would not occur beyond 1.6 miles downwind of the tank, or about 1.6 miles from the proposed plant site. We have conservatively assumed that the unconfined vapor cloud is detonated at that point, and calculate a peak reflected overpressure of about one pound per square inch at the proposed plant site, which is signficantly less than the external pressure resulting from the design basis tornado wind (about 2.3 pounds per square inch). Therefore, we conclude that the postulated detonation would have no adverse effects on the safety-related plant structures.

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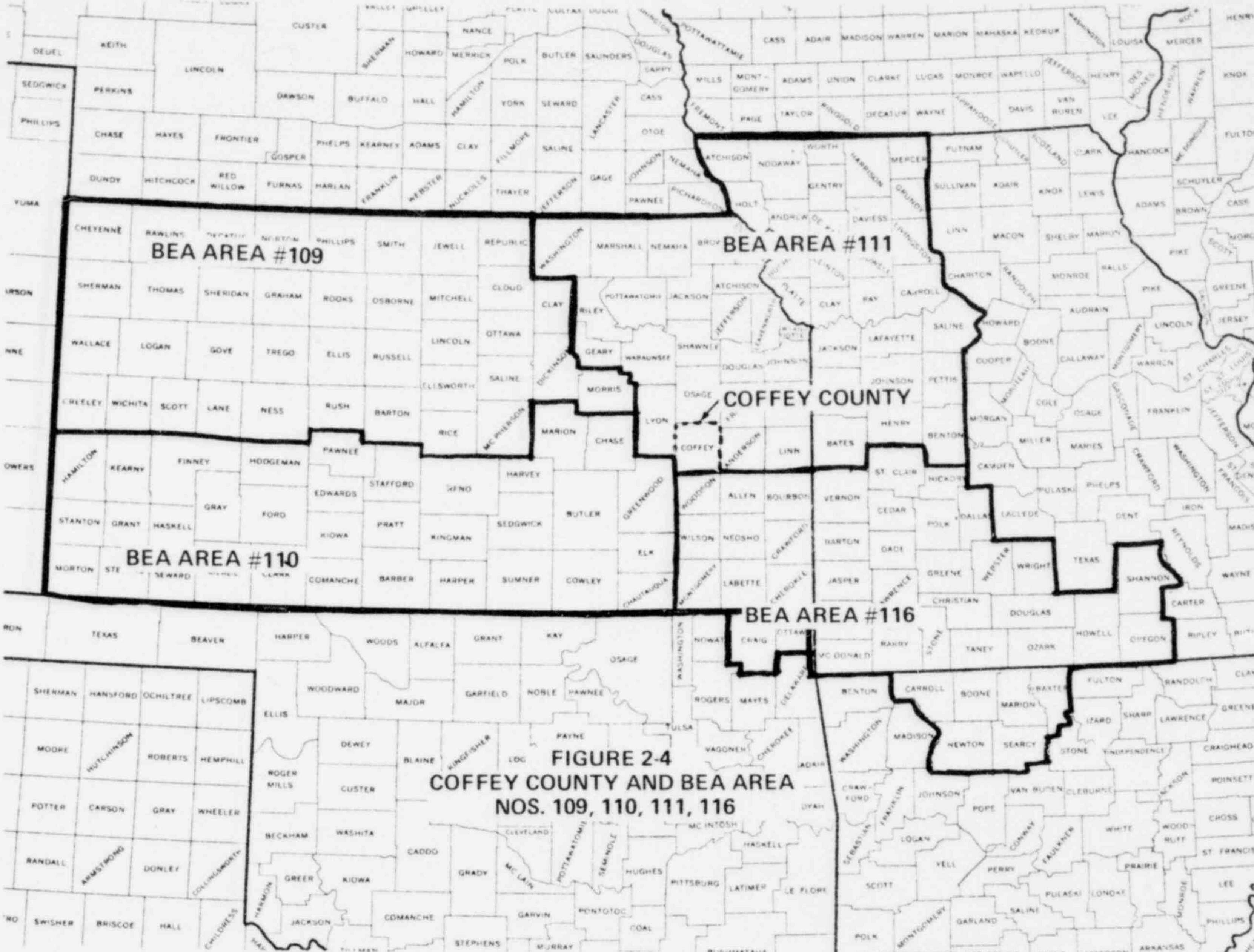

FIGURE 2-4
COFFEY COUNTY AND BEA AREA
NOS. 109, 110, 111, 116

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There are no significant military facilities near the proposed plant. The only military facility within five miles of the site is the Kansas Army National Guard Armory in Burlington, located about 3.9 miles southwest of the site.

There are two operating quarries within five miles; the Sharpe quarry located three miles northeast of the site, and the Peeca quarry located 2.2 miles west-southwest of the site. Blasting is conducted at both quarries using amounts up to 1,000 pounds of dynamite or 20 bags of ammonia nitrate. The resulting ground shock at the minimum distance of 2.2 miles is significantly below the ground shock resulting from a postulated safe shutdown earthquake (See Section 2.5.4).

There are no significant highways, railroads, water transportation routes, or airports near the site. The nearest major highway is U.S. Route 75 which passes 2.8 miles west of the site. The nearest railroad track is a line of the Missouri Pacific Railroad passing about nine miles southeast. The Santa Fe Railroad track, which passes through the site, has been abandoned and title to the right-of-way has reverted to the applicants.

The closest pipelines are 8-inch and 12-inch diameter lines which carry refined hydrocarbons (propane, butane, gasoline, etc.). These pipelines pass 2.6 miles northwest of the plant site. We have evaluated the consequences of a postulated failure of these pipes and have determined that this postulated failure results in effects at the plant site that are similar in magnitude to the postulated propane storage tank failure discussed above. Therefore, we conclude that postulated pipeline accidents would have no adverse effects on the safety-related plant structures.

We conclude that the identified activities at nearby industrial, military, and transportation facilities do not require special consideration in the design of the Wolf Creek plant, and, therefore, in this respect the proposed Wolf Creek site is acceptable.

2.3 Meteorology

Information concerning atmospheric diffusion characteristics of a proposed nuclear power plant site is required for a determination that radioactive effluents from postulated accidental releases, as well as routine operational releases, are within Commission guidelines. Evaluation of regional and local climatological information, including extremes of the climate and severe weather occurrences which may affect the design and siting of a nuclear plant, is required to assure that the plant can be designed and operated within the requirements of Commission regulations.

2.3.1 Regional Climatology

The climate of the region in which the proposed site is located can be described as continental, characterized by rapid changes in temperature and marked extremes, resulting in hot summers and cold winters. The proposed site lies near the principal track of winter and spring storms that move northeast and east through the region. Hence, severe weather is not uncommon.

Thunderstorms can be expected to occur on about 59 days per year, being most frequent in May, June, and July. The maximum observed hailstone in the United States, which weighed 1.67 pounds and measured about 5.5 inches in diameter, was reported at Coffeyville, Kansas (about 80 miles south of the site). The applicants have also examined data from "Storm Data," a monthly publication by the U.S. Department of Commerce, for the period 1959-1973, and stated that hailstones of three-inch diameter or greater are not uncommon.

During the period 1955-1967, 50 tornadoes were reported in the one-degree latitude-longitude square containing the proposed site, giving a mean annual tornado frequency of 3.8. The computed recurrence interval for a tornado at the plant site is 340 years. May is the month with the highest frequency of tornado occurrences. The design basis tornado characteristics selected by the applicants conform to the recommendations of Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants," for this region of the country and, therefore, are acceptable.

The "fastest mile" wind speed reported at Topeka, Kansas (about 50 miles north of the site) was at least 81 miles per hour (June 1958). The operating basis wind speed of 100 miles per hour (defined as the "fastest mile" wind speed at a height of 30 feet with a return period of 100 years) has been selected by the applicants for the design of their proposed plant.

In the period 1936-1970, there were only about two atmospheric stagnation cases reported in the vicinity of the proposed site. The total time of the stagnation conditions was about nine days.

2.3.2 Local Meteorology

Climatological data from Topeka and Chanute, Kansas, available data from Garnett, Ottawa, and Emporia Kansas and available onsite data have been used to assess local meteorological characteristics of the site.

Mean monthly temperatures at the site may be expected to range from about 29 degrees Fahrenheit in January to about 80 degrees Fahrenheit in July. Extreme temperatures of 118 degrees Fahrenheit and -28 degrees Fahrenheit have been reported in Ottawa.

Annual average precipitation in the site area is about 32 inches, with about 71 percent occurring in the period April through September. The maximum 24 hour rainfall reported at Topeka was about 8.1 inches. Annual average snowfall is about 20 inches, although 16 inches of snow has been reported in 24 hours at Topeka.

Wind data from the 10 meter level of the onsite meteorological tower for the period June 1, 1973 through May 31, 1974 indicate a prevailing wind direction from the south (20 percent of the time), with winds from the south-southeast, south, and south-southwest totaling about 42 percent of the time. Ten years (1955-1964) of wind data from Chanute also indicate prevailing winds from the south, occurring about 16.5 percent of the time. Mean wind speeds at the proposed site, at Topeka, and at Chanute are all about 11 miles per hour.

2.3.3 Onsite Meteorological Measurements Program

The onsite meteorological measurements program for the Wolf Creek site became operational in May 1973. Measurements are made from an instrumented 300-foot high tower located about 2600 feet north-northeast of the proposed location of the main reactor structures. Wind speed and direction are measured at the 10-meter, 35-meter and 60-meter levels on the tower. The vertical temperature gradients are determined by measurements between the 10- and 35-meter levels, between the 10- and 60-meter levels and between the 10- and 90-meter levels. The ambient dry bulb temperature and the dewpoint temperature are measured at the 10-meter level; and precipitation and solar radiation are measured at the two-meter level. The primary system for recording the data is digital while analog strip charts are used as the secondary system.

The meteorological measurements program conforms to the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs," and, therefore, is acceptable.

The applicants have provided two sets of joint frequency distributions of wind speed and direction by atmospheric stability (defined by vertical temperature gradient) from the onsite meteorological tower for the period June 1, 1973 through May 31, 1974. The first set was based on the wind speed and direction measured at the 10-meter level and the vertical temperature gradient measured between the 10-meter and 90-meter levels. At our request, the applicants also provided a second set of data based on the wind speed and direction at the 10-meter level and the vertical temperature gradient measured between the 10-meter and 60-meter levels. We have used the second set of joint frequency distributions for our evaluation of atmospheric dispersion characteristics at the proposed site because we find these meteorological data are more representative of conditions in the atmospheric layer into which effluents from facility buildings and vents would be released. Data recovery for each set of meteorological data was 95 percent.

We conclude that the one year of onsite meteorological data obtained from June 1, 1973 through May 31, 1974, provides a reasonably representative and conservative basis for estimating atmospheric dispersion conditions at the proposed site since the onsite measurements program conforms to the recommendations of Regulatory Guide 1.23 and the data compares favorably with existing long term data for the site area.

The applicants have collected a second year of onsite meteorological data. The applicants also state that a decision to continue the onsite meteorological measurements program during construction will be made after the second year of data have been analyzed and effects of the proposed Wolf Creek cooling lake on local meteorological conditions have been discussed with us.

2.3.4 Short-Term (Accident) and Long-Term (Routine) Diffusion Estimates

Utilizing standard staff practices discussed below, we have evaluated the meteorological diffusion characteristics of the site for both accident analysis and routine release analysis purposes. The evaluation of the calculated offsite doses resulting from radioactive releases due to postulated accidents required calculations of the relative concentration for the first 30 days following an assumed accident. The impact of routine radioactive releases required calculations of an annually averaged relative concentration. These relative concentrations were then incorporated into dose analyses. Accident dose analyses utilize calculated relative concentration values which vary with time and distance. The staff uses its most conservative assumptions when calculating the relative concentration values for the first eight hours following an assumed accident. Additional credit is given for diffusion and spread of the gaseous plume for time peiods beyond the first eight hours. The calculated dose at the minimum exclusion area boundary at the end of the first two hours and the 30-day dose at the low population zone boundary must be within 10 CFR Part 100 limits. In our evaluation of short-term doses (the first two hours after the release at the exclusion area boundary distance and the first eight hours after the release at the low population zone boundary distance) due to accidental releases from buildings and vents, a ground level release with a building wake factor of 1325 meters squared was assumed.

Using the diffusion model described in Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of A Loss-of-Coolant Accident for Pressurized Water Reactors," we calculated the relative concentration value for the two hour time period, which is exceeded no more than five percent of the time, to be 1.9×10^{-4} seconds per cubic meter at the exclusion area boundary distance of 1200 meters. This value of relative concentration is equivalent to dispersion conditions produced by Pasquill type F stability with a wind speed of 1.5 meters per second.

The relative concentration values we calculated at the outer boundary of the low population zone distance of 4032 meters for various time periods, are as follows:

Time Period	Relative Concentration (seconds per cubic meter)
0-8 hours	2.7×10^{-5}
8-24 hours	1.8×10^{-5}
1-4 days	7.4×10^{-6}
1-4 days	7.4×10^{-6}
4-30 days	2.0×10^{-6}

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The average annual relative concentration value was calculated using the model described in Regulatory Guide 1.42, "Interim Licensing Policy on As Low As Practicable for Gaseous Radioiodine Release from Light-Water-Cooled Nuclear Power Reactors," for vent releases, assuming a ground level source. The highest value of relative concentration occurred at the site boundary 1800 meters north of the proposed release point, and was 1.1×10^{-6} seconds per cubic meter.

2.4 Hydrology

2.4.1 Hydrologic Description

The proposed site for the Wolf Creek plant is located in the Neosho River Basin. Plant grade will be 1099.5 feet above mean sea level with entrance levels to the plant structures at elevation 1100.0 feet above mean sea level. Primary plant structures will be located on the east side of a proposed 5090-acre cooling lake. The lake, will be created by damming Wolf Creek a tributary of the Neosho River. It will have an estimated normal operating level of 1087 feet above mean sea level. The lake will be created by constructing an earth dam with a crest elevation of 1100 feet above mean sea level and a length of 13,000 feet. Five saddle dams will be located around the west side of the lake and one on the east side. Their crest elevation will also be 1100 feet above mean sea level. Three discharge pipes will be provided on the east abutment of the main dam for low level releases to primarily prevent the concentration buildup of total dissolved solids in the cooling lake. A service spillway and an auxiliary spillway will be constructed on the east abutment of the main dam. The service spillway will be an uncontrolled 100-foot long ogee with a crest elevation of 1088 feet above mean sea level. The auxiliary spillway will be an open cut type with a crest length of 500 feet and a crest elevation of 1090.5 feet above mean sea level.

The cooling lake will provide cooling water for normal operation and for shutdown requirements. It will impound runoff from a drainage area of 27.4 square miles. Additional make-up water (a minimum of 41 cubic feet per second and a maximum of 120 cubic feet per second) will be pumped from the John Redmond Reservoir, located four miles northwest of the plant on the Neosho River. Two baffle dikes and three channels will be included in the cooling lake to increase circulation and improve heat dissipation of circulating water discharges.

A 95-acre submerged pond will be formed by excavating a portion of a finger of the proposed cooling lake and constructing a submerged seismic Category I dam. This submerged pond (safety-related portion of the ultimate heat sink) will provide essential cooling water for shutdown in the event water is not available from the main cooling lake.

A survey of wells in the site area made by the applicants indicates as many as about 163 wells within five miles of the plant site. These wells are used for domestic and livestock purposes except for the municipal well at New Strawn, approximately three miles northwest of the plant site. The applicants have committed to sealing all wells in the cooling lake area with puddled clay.

2.4.2 Flood Potential

Several potential flood producing sources were investigated by the applicants. These potential sources included postulated dam failures on the Neosho and Cottonwood Rivers, inadequate drainage of runoff in the plant site area, and a probable maximum flood on Wolf Creek.

The Neosho River Basin in Kansas is an elongated area of about 5790 square miles. It originates in Morris County, Kansas and flows southeastward and south through the state. The major tributary to the Neosho River above Burlington, Kansas, is the Cottonwood River, which orginates in Marion County and joins the Neosho River about six miles east of Emporia, Kansas. There are three major reservoirs presently in operation and a fourth is authorized in the upper Neosho River Basin. These are the John Redmond, Council Grove, Marion and Cedar Point (not yet constructed) Reservoirs. The domino failure (failure of upstream dams causing successive failures of downstream dams) of the four dams coincident with a standard project flood (described in Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants") was postulated by the applicants in their analysis. We evaluated the potential effects of this postulated accident and we concur that the resultant flood would have no effect on safety-related buildings at the Wolf Creek site, since the resulting maximum flood stage would not spill into the Wolf Creek drainage basin.

The applicants proposed site drainage facilities will be designed such that a local probable maximum flood will not constitute a threat to safety-related buildings. The plant area will be graded to elevation 1099.5 feet above mean sea level and sloped away from the facility buildings toward peripheral roads. Safety-related structure entrances will be at elevation 1100.0 feet above mean sea level. The applicants also stated that the roofs of all safety-related structures will be designed to withstand loads in excess of those generated by rain accumulation during a local probable maximum precipitation storm. We conclude that the design bases for the site drainage facilities are acceptable.

The applicants estimated the water level in the cooling lake produced by a probable maximum flood on Wolf Creek. For this evaluation, the probable maximum precipitation was applied on an antecedent flow resulting from one-half the probable maximum precipitation over the total 27.4 square mile drainage area, of which 8.2 square miles is lake surface area. The results of our independent analysis agreed with the applicants' estimated peak flow of 82,089 cubic feet per second due to this probable maximum flood.

The resultant peak outflow from the probable maximum flood over the 100-foot long spillway, as originally proposed, was 4,069 cubic feet per second, corresponding to a maximum water surface elevation of 1096.7 feet above mean sea level for the cooling lake. This did not allow for adequate freeboard on the main dam. As a result of our

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concerns, the applicants included a 500-foot long auxiliary spillway with a crest elevation of 1090.5 feet above mean sea level in the design of the main dam. With this design change, the applicants' estimated peak outflow over the two spillways was predicted to be 20,076 cubic feet per second for a probable maximum flood on Wolf Creek, resulting in a freedboard of 4.8 feet.

We independently estimated the maximum water level in the cooling lake, due to a probable maximum flood, by routing the probable maximum flood through the lake and over the two spillway. We concluded that the applicants' predicted maximum water level in the lake is acceptable. However, since we used the spillway design values supplied by the applicants in estimating this water level, we require that a detailed description of the design of the spillways be submitted for our review and approval prior to the construction of the spillways. The applicants have committed to provide, for our review and approval, the detailed drawings for both the spillways and the water surface profiles downstream of the service spillway after the detailed designs are finalized. We find this commitment acceptable.

The amount of freeboard required on the main dam was determined by the applicants by superimposing the significant wave (which is higher than 67 percent of all the waves) effects, due to a 40-mile per hour overland wind, on the maximum lake level due to a probable maximum flood. The resultant freeboard required on the main dam by this determination was 3.98 feet as compared to the 4.8 feet available. We estimated the wind wave activity on the main by dam using the maximum wave (which is higher than 99 percent of all the waves) and determined that, in this case, 5.2 feet would be required to prevent overtopping. The applicants' justification for using the significant wave instead of the maximum wave to calculate runup was based on the dam's crest width of 20 feet and that a gravel service road will be provided on the crest. Therefore, the crest design will inhibit the erosive action of the water before it reaches the downstream slope of the dam. Based on our review of the applicants' analysis, we conclude that the amount of freedboard to be provided for the main dam is acceptable.

The applicants determined that the water elevation in the cooling lake due to a probable maximum flood plus wind wave activity will be below the plant grade level of 1099.5 feet above mean sea level. Our independent analysis of wave effects indicates that the applicants' analysis is acceptable and we conclude that the probable maximum flood on Wolf Creek will not adversely affect the plant site.

The applicants estimated the maximum lake level plus wind wave activity at the essential service water pumphouse to be 1100.4 feet above mean sea level. The structure will be designed to withstand this high water elevation. We independently evaluated the wind wave activity at this structure and conclude that the applicants' value is acceptable.

2.4.3 Water Supply

The cooling water necessary to remove waste heat from the plant during normal operation or postulated accident conditions, will be obtained from the proposed cooling

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lake via pumps to be located in the essential service water pumphouse. The intake structure at the pumphouse will be located on the edge of the cooling lake near the proposed plant site and will be connected to a 95-acre submerged pond. This submerged pond will be formed by excavating the floor of a finger of the cooling lake and by constructing a seismic Category I dam. The essential service water intake and discharge structures will both be seismic Category I and will be located on opposite ends of the V-shaped submerged pond to prevent short-circuiting of the flow between the intake and discharge structures. The submerged pond will contain 330 acre-feet of water at 1070 feet above mean sea level (the top of the submerged dam).

At our request, the applicants have committed to a monitoring program for the submerged pond to assure that the capacity of the pond remains above the level required to provide a cooling water heat sink for the safe shutdown of the reactor under postulated accident conditions. Prior to filling the cooling lake, the bottom of the excavated pond will be surveyed at 18 stations. Subsequently, soundings will be taken to verify the submerged pond capacity and to establish a sediment deposition pattern. We will review the details of the applicants' dredging procedures, that will be required to restore the capacity of the submerged pond if it is reduced below a certain minimum level, at the operating license stage of review.

2.4.4 Ground Water

Ground water in the Wolf Creek site vicinity occurs in three types of aquifers, i.e. alluvial, soil and weathered bedrock, and consolidated bedrock. The alluvial aquifers in the region are composed of silts, sands and gravels. The soil and weathered bedrock aquifers are composed of weathered shales, siltstones, sandstones, limestones and the soil derived from them. Where the alluvial and soil and weathered bedrock aquifers are contiguous, they are hydraulically connected and recharge to both occurs from local precipitation percolating through the soil. The bedrock aquifers are composed of sandstones and limestones. None of these bedrock units near the site are capable of yielding large quantities of water to wells.

The applicants do not plan to use ground water at or near the site during plant construction or operation. The design water level for ground water induced hydrostatic loading at the plant site is conservatively established at plant site grade elevation 1099.5 feet above mean sea level. The actual water table will be several feet below the surface.

Ground water movement is in a southwesterly direction from the plant site, across the proposed cooling lake, and toward the Neosho River. Within a five-mile radius of the plant site, the shallow ground water table closely parallels the topographic surface. The cooling lake is expected to alter ground water movement in the area, reducing gradients from upstream directions, increasing gradients in downstream directions, and generally increasing seepage and ground water flow.

We have analyzed the effects of a postulated radioactive liquid spill into the ground water from the proposed plant. A spill, postulated to occur in the radwaste building, will enter the ground water and, due to the southwesterly gradient for ground water movements, would travel to the cooling lake. The travel time to the lake was estimated to be 38 years. The dilution factor due to mixing in the cooling lake alone was estimated to be 2.4×10^7. An evaluation of this postulated accident, which is discussed in Section 11.2, indicates that the resultant concentrations of radioactive materials in the cooling lake would be a small fraction of the limits of Appendix B to 10 CFR Part 20. The resultant concentrations in ground water beyond the cooling lake area would be less than that in the cooling lake due to further dilution effects.

2.4.5 Conclusions

Based upon our independent review and analyses, we conclude that acceptable flood design bases have been provided, an acceptable water supply can be assured for safety-related purposes, and plant construction and operation would not adversely affect regional or local ground water users. We will review the detailed design of the spillways prior to their construction.

2.5 Geology and Seismology

2.5.1 Regional Geology

The proposed Wolf Creek site is located in the Central Stable Region tectonic province (Eardley, 1963). The site is also located within the Osage Plains Section of the Central Lowland physiographic province (Fenneman, 1969).

Stratigraphically, the site region is characterized by gently westward dipping well-consolidated, northeast-striking Paleozoic sediments of varied lithologies approximately 2,500 feet in thickness, overlying Precambrian sediments and metasediments. Erosional remnants of the Late Tertiary - Early Pleistocene elevated gravel terrace, which are largely chert, have been identified within an approximate 60-mile radius of the site. Quarternary deposits consist predominantly of stream alluvium. The plant site area is unglaciated with the southern terminus of glaciation about 50 miles north of the site. Very localized intrusives, peridotite and granite of undetermined age, have been mapped in Woodson County, Kansas some 30 miles south of the proposed site (Jewett, 1964) (King and Beikman, 1974).

Structurally, the proposed site region is characterized generally by numerous gentle broad arches and basins, accompanied locally by some faulting and tight folding. Large regional, relatively nearby structures of significance to the proposed facility include the faulted Nemaha anticline, the Bourbon arch and the Forest City and Cherokee basins. With the exception of the Forest City basin, the above structures, as well as the relatively nearby (20 miles to the north at its nearest approach) northwest-trending Chesapeake fault zone and an unnamed northwest striking fault 35 miles southwest of the site, are at least pre-middle Pennsylvanian in age (320 million years before the present).

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The age of the Forest City basin is considered to be pre-Pleistocene. The faulted Nemaha anticline of pre-Middle Pennsylvanian age, trends North 20 degrees East and extends through Kansas into adjacent states. This structure approaches within approximately 50 miles west of the proposed facility. According to the criteria in Appendix A to 10 CFR Part 100, the structures are considered incapable and thus constitute no hazard to the proposed nuclear power plant. Capable faults are unknown in the site region. Our independent review of the pertinent literature confirms the opinion, as expressed by the Kansas Geological Survey (Hambleton, 1973a), that there is apparently no evidence indicating either single or multiple movement of surface faults within the past 35,000 years.

Subsidence due to the extraction of fluid (oil and water), gas and the dissolution or dry mining of solids (in particular salt and some limestone) has been noted in many areas throughout the continental United States. Within the continental interior, including the proposed site area, subsidence due to either fluid or gas extraction is unknown and extremely unlikely because of the well-consolidated Paleozoic bedrock underlying the site and region within which the fluids are contained. The Kansas Geological Survey (Hambleton, 1973b) reports no known cases in Kansas of ground subsidence resulting from the removal of oil and gas.

Salt is common to both Silurian and Permian deposits. The Silurian deposit beneath the site area however is thin, less than ten feet thick, and consists solely of limestone at a depth of approximately 1,900 feet. Extensive Permian evaporitic deposits (including salt) are found throughout the western two-thirds of Kansas. No Permian strata are present in Coffey County, Kansas (Merriam, 1962).

Limestone solutioning in the form of sinkholes or other depressions is unknown in the site area (Merriam, 1962). In our meeting with the Kansas Geological Survey in September 1974, they stated they have seen no evidence of any limestone solutioning at the site. The applicants own or control all mineral rights in the site area, thus precluding any potential hazard associated with mineral exploitation beneath the site.

We have concluded that no known regional geotechnical phenomena, attributable to capable faults, subsidence or solutioning due to mineral or fluid extraction, or other geologic features, present a potential hazard to the proposed Wolf Creek facility.

2.5.2 Site Geology

Surficial deposits in the planned area for plant structures consist of a thin veneer of residual soils (silty sands to soft clays) six to eleven feet thick, and from zero to ten feet of alluvial material (clays, silty clays and clayey silts) in the vicinity of the proposed ultimate heat sink seismic Category I submerged dam and the essential service water structures. Overburden is underlain by approximately 2,500 feet of well-consolidated gently-dipping Paleozoic sediments. Tertiary chert gravels, with coarse to fine sand up to four feet in thickness, cap some of the higher hills within the site boundary. These sediments are mapped along a segment of the proposed main and saddle dam system, south and southwest of the areas for the proposed plant structures and the ultimate heat sink.

Well-consolidated Upper Pennsylvanian sediments, consisting of alternating layers of clay shales and fine-grained limestones with shale interbeds, were encountered directly beneath the thin soil overburden. Bedrock exposures are common throughout the site boundaries. In addition to the shales and limestones, sandstones and thin (less than one foot thick) coal beds are found below the principal foundation strata. The three foundation stratigraphic units consist, in increasing age, of the Jackson Park shale, the Heumader shale and the Plattsmouth limestone. The thickest of these units is the Heumader shale (34 feet maximum) while the Plattsmouth limestone attains a thickness range of 11 to 14 feet. The site borings terminate from 46 to 453 feet in depth in the competent Upper Pennsylvanian rocks. No anomalous or otherwise potentially hazardous conditions were noted or observed during the site investigations.

No safety-related problems, such as subsidence or collapse, are anticipated in the site area with regard to the extraction of coal, oil, gas, water, or salt. All limestone encountered in the site exploration program is relatively thin (less than 20 feet thick), competent and devoid of solution cavities. Past or future limestone solutioning, thus, does not present a hazard to the site.

No faults have been identified in the site area or its immediate vicinity. The applicants have agreed to geologically map all excavations for seismic Category I structures. If required to adequately interpret the site geology, the applicants have further agreed to geologically map non-Category I excavations. We find these commitments acceptable.

We conclude that there are no capable faults or other geologic features or activities attributable to man (mining or extraction of fluids or gases) in the site vicinity representing a hazard or potential hazard to the proposed facility. Therefore, with regard to geologic considerations, we conclude that the proposed site is acceptable.

2.5.3 Surface Faulting

Based on our review of the data, we conclude that there is no evidence to indicate surface faulting in the site area or any geologic structure in the vicinity of the site that could cause surface displacement. Therefore, we conclude that faulting is not considered a potential hazard at the proposed site of the Wolf Creek facility.

2.5.4 Vibratory Ground Motion

Our review of the vibratory ground motion potential for the proposed Wolf Creek site was conducted to:

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(1) Determine an acceptable value for the intensity of the safe shutdown earthquake for the Wolf Creek site and the resultant ground acceleration value to be applied to the design of those seismic Category I structures outside the scope of the standard (SNUPPS) portion of the Wolf Creek facility; and

(2) To assure that the resultant ground acceleration value for the Wolf Creek site safe shutdown earthquake does not exceed the design value of 0.2g at the foundation established for the standard (SNUPPS) portion of the Wolf Creek facility.

The proposed Wolf Creek site is located in the Central Stable Region tectonic province (Eardley, 1962). The historical record of earthquakes of this large province includes numerous events of low to moderate intensity. The spatial distribution of these shocks appears generally random, but shows some clustering and a fairly prominent north-south alignment of the larger earthquakes through the states of Oklahoma, Kansas and Nebraska. Docekal (1970) refers to the latter feature as the Midcontinent Seismic Trend.

Results of earthquakes obtained from instruments for the Central Stable Region are few in number and generally lacking in precision. Hypocentral depths in particular are poorly determined; however, the earthquakes are almost certainly of crustal origin. The only focal mechanism solutions which have been reported in the literature are those for earthquakes in southeastern Missouri and southern Illinois determined by Street, et al. (1974). These mechanisms, which are on the perimeter of the Mississippi Embayment, appear to have no direct bearing on an evaluation of the Wolf Creek site.

Given the lack of well determined earthquake parameters, it is in general not possible to establish reasonable relationships between earthquakes and tectonic structures in the Central Stable Region. The association of the Midcontinent Seismic Trend with the Nehema Uplift (also referred to as the Nemaha anticline) is a likely exception. The Wolf Creek site lies about fifty miles east of the Nemaha Uplift.

In the Wolf Creek Site Addendum Report, the applicants stated that the Wolf Creek site is located in the Central Stable Region, but did not acknowledge the region's status as a tectonic province. The applicants took the position that because it is composed of a series of large basins and uplifts, each having its own tectonic and seismological history, the Central Stable Region is too simplified a subdivision to be used for evaluating a maximum earthquake. The safe shutdown earthquake was then evaluated by considering these large structural features to be tectonically independent entities.

Regarding the Nemaha Uplift, the applicants took the position that at least four historical earthquakes of intensity Modified Mercalli (MM) VII were associated with this structure. In order to evaluate a maximum intensity at the site which could result from a Nemaha Uplift earthquake, two postulated events were considered by the applicants. One of these events was assumed to have an epicentral intensity MM VII

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and postulated to occur at the closest approach of the Nemaha Uplift to the site, a distance of about 50 miles. The resultant intensity at the site would be about an intensity MM V. The second event was assumed to have an epicentral intensity MM VIII and postulated to occur at the closest approach of the western flank of the Nemaha Uplift to the site, a distance of about 75 miles. In the second case the resultant intensity at the site would be about an intensity MM VI. In both cases the applicants determined attenuation by using curves developed from four historical Nemaha Uplift earthquakes.

The applicants stated that no intensity MM VII or greater earthquakes not related to structure have occurred within the boundaries of any large scale feature within 200 miles of the site. On this basis, a site intensity MM VI and a ground acceleration value of 0.10g were chosen by the applicants for the Wolf Creek site safe shutdown earthquake.

We agree with the applicants that there is historical earthquake activity associated with the Nemaha Uplift. Accordingly, we concluded that the Nemaha Uplift must be considered an active structure as was proposed by Merriam (1963). We differed with the applicants regarding the implication of this activity. It was our view that the applicants had not adequately supported the conservatism of the earthquakes postulated to occur on this structure.

The Nemaha Uplift has a total length of about 400 miles. Merriam (1963) cites reports of west-dipping reverse faulting in association with it and suggests that these faults of unknown extent could be the earthquake sources. Thus we saw the possibility that earthquakes of larger source dimensions, than those which have occurred during the relatively short historical record, could occur during the operating life of the Wolf Creek facility. Although the applicants had given some consideration to this possibility by postulating an intensity MM VIII earthquake on the western flank of the anticline where the historical activity has occurred, they did not place this earthquake at the closest approach of the structure to the site. Moreover, they had provided no basis for the assumption that an intensity MM VIII is an adequate upper bound. Consequently, we required the applicants to expand on their discussion of the Nemaha Uplift to justify the assumption that intensity MM VIII is a reasonable upper bound for earthquake activity associated with the Nemaha Uplift.

In response to our question regarding the Nemaha Uplift, the applicants cited Docekal's (1970) hypothesis that the activity related to the Nemaha Uplift occurs in Kansas and Nebraska along a zone of weakness at the contact between the western boundary of Nemaha Uplift at the Keweenawan volcanics and in Oklahoma where the Nemaha Uplift intersects the Ouachita Tectonic Trend. On this basis the applicants claimed that no strain accumulation would occur either along the strike of the Nemaha Uplift or towards its eastern flank beyond where earthquakes have occurred historically. The applicants also pointed out that an intensity MM VIII represents an energy release about ten times that of an intensity MM VII.

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We also viewed the applicants' treatment of the random earthquake to lack adequate justification in terms of the regional tectonics. Consequently, we required the applicants to compare the site area with other regions of the Central Stable Interior which have experienced earthquakes of intensity MM VII in order to support any differentiation which they might make.

In support of their treatment of the maximum random earthquake, the applicants asserted that differences in seismicity between one subregion of the Central Stable Region and another may be greater than between the Central Stable Region and other large tectonic provinces. The applicants also stated that the subregion in which the site is located differs from other subregions in that it lacks Precambrian instability which they claimed is responsible for the continuing seismic activity outside of the region in which the site is located.

When we completed our review of these responses, we concluded that Docekal's hypothesis does not rule out earthquakes of intensity greater than MM VIII for the Nemaha Uplift, because it would not imply sufficient truncation of the earthquake source area to make this a conservative upper bound. However, in our view, if earthquakes of intensity MM X or greater have been occurring on this structure, some geologic evidence of recent movement would be observed. No such evidence is known. Therefore, we consider that a conservative upper bound for earthquakes associated with the Nemaha Uplift would be less than intensity MM X. Due to the distance between the Nemaha Uplift and the proposed site, which is as close as 50 miles away, the attenuation of this upper bound earthquake would result in a site intensity no greater than intensity MM VII.

The applicants' response on the random earthquake cites no evidence in support of their statements concerning the treatment of the random earthquake. We therefore took the position that an intensity MM VII at the site, which is equivalent to the intensity of the earthquake that occurred in Catoosa, Oklahoma (near Tulsa) in 1956, is an appropriate value for the random earthquake that should be considered in the development of the safe shutdown earthquake for the Wolf Creek site.

We chose a horizontal acceleration of 0.12g for the Wolf Creek safe shutdown earthquake which represents the trend of the means of the observed acceleration data at intensity MM VII. We intend that this acceleration be used as the zero period limit of suitable response spectra applied at the foundations of seismic Category I structures. Although the applicants disagreed with the basis for our position, they have agreed to design those seismic CategoryI structures which are outside the scope of the standard plant portion of the facility to a safe shutdown earthquake acceleration of 0.12g applied at the foundation level. For the standard plant portion of the facility, the resultant safe shutdown earthquake acceleration at foundation level would be below the design value of 0.2g, and therefore is acceptable.

2.5.5 Conclusions

Based on our review of the geological and seismological data for the proposed Wolf Creek site, we conclude (1) there are no geological structures that would tend to localize earthquakes in the site vicinity or cause surface faulting at the site, (2) the seismic design bases as modified by our position on the ground acceleration value due to a postulated safe shutdown earthquake, are appropriately conservative for the earthquake potential at the site, and (3) there are no known geologic features at the site which could represent a potential hazard due to solution activity and/or subsidence. Therefore, we conclude that the proposed Wolf Creek site is acceptable with regard to geology and seismology considerations.

2.6 Foundation Engineering

The proposed plant site is covered by residual soils developed from weathering of the underlying rock strata. The soil deposits are relatively thin, generally four to eight feet thick. The soil profile shows wide variations in properties ranging from plastic clays to silty sand. The underlying bedrock consists of alternating strata of limestones, shales, and sandstones which dip gently to the south and southwest. The topography at the plant site is generally flat, with ground surface elevations ranging from 1,102 to 1,109 feet above mean sea level. The final grade in the plant area will be elevation 1,099.5 feet above mean sea level.

The main bearing stratum for seismic Category I structures will be the Heumader shale. The Heumader shale is a dark gray shale that yields an average core recovery of 99 percent and a rock quality designation average of 89 percent. The thickness of the Heumader formation varies from 24 to 31 feet. The upper surface of this formation in the planned plant area is at an elevation of approximately 1,092 feet above mean sea level.

Underlying the Heumader shale is the gray, fine-grained, Plattsmouth limestone. Core recovery in this limestone averaged 90 percent and the rock quality designation averaged 85 percent. The thickness ranges from 12 to 14 feet, with the Heumader contact at an elevation of approximately 1,065 feet above mean sea level. The deepest structural excavation will terminate in the upper portion of the Plattsmouth limestone.

Foundation excavations will be drained by a system of sumps and ditches. Only minor dewatering will be required because foundation materials have low permeabilities. The lower portion of the Heumader shale and the upper portion of the Plattsmouth limestone will be blasted and excavated using the presplitting method. Some areas will be overexcavated to provide a more uniform support for the foundations. The overexcavation zones will be backfilled with lean concrete.

Excavated soils will be used as backfill around plant structures. Backfill material will be compacted to a dry density of at least 90 percent of the maximum dry density as determined by American Society for Testing Materials Test Designation 1557-70. All safety-related structures will be founded on rock or concrete fill.

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The foundation design, with bearing pressures ranging from 4,000 to 8,000 pounds per square foot on rock formations, provides a large bearing capacity factor of safety. The maximum computed settlement in the planned plant area is one inch, with the maximum differential settlement computed to be one-half inch. The settlements will be elastic and will occur essentially upon application of the load. These results indicate that the bearing material will provide very good foundation support for the proposed structures.

The plant cooling water requirements will be supplied by the proposed main cooling lake and a seismic Category I pond adjacent to the planned plant area. The main cooling lake will be impounded by a main dam and saddle dams, all of which are earth fill dams. As a result of our concerns expressed during the review of the Wolf Creek ultimate heat sink design, the applicant modified the design of the main dam and saddle dams to safely withstand a postulated operating basis earthquake. Design, construction, and inspection of these dams and the cooling lake will be in accordance with the criteria and standards developed by the Corps of Engineers and the Bureau of Reclamation.

The seismic Category I pond, which serves as the safety-related portion of the ultimate heat sink, is adjacent to the plant area and will be impounded by a 20-foot high submerged earth dam designed to withstand a postulated safe shutdown earthquake. The submerged seismic Category I dam and pond will be located in a finger of the main cooling lake southeast of the plant site. The seismic Category I dam will be constructed on a rock foundation with four horizontal to one vertical side slopes. The earth embankment will be constructed with material excavated from the cooling lake. This material is described as a brown clay with silt and a trace of sand. Fill materials will be compacted to densities greater than 95 percent of the maximum dry density based on American Society for Testing Materials D698-70 specifications. The crest elevation of the seismic Category I dam will be 1,076 feet above mean sea level, 11 feet below the main cooling lake normal pool elevation. The combination of the foundation design for the main cooling lake dam and the seismic Category I submerged dam provides an acceptable foundation engineering design for the ultimate heat sink.

The only other seismic Category I slopes will be along the intake channel for the emergency service water pump house. These slopes will be excavated in rock with a one to one slope designed to withstand the safe shutdown earthquake.

The emergency service water pump house and discharge structure will be founded on competent rock foundations similar to the plant area structures. The emergency service water pipelines will be buried in excavations, into the residual soil and rock formations, at depths which will assure that the pipelines will be protected from damaging external loads. Compacted granular backfill will be used as bedding material below the pipe and compacted cohesive backfill will be placed over it.

Based on our review of the information presented in the Wolf Creek Site Addendum Report, we conclude that the foundation engineering design for the proposed power plant facility meets the requirements of Appendix A to 10 CFR Part 100 and, therefore, is acceptable.

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3.0 DESIGN CRITERIA FOR STRUCTURES, SYSTEMS AND COMPONENTS

3.1 Conformance with General Design Criteria

The applicants have stated that the Wolf Creek facility will be designed, constructed and operated in accordance with the Commission's General Design Criteria for Nuclear Power Plants (Appendix A to 10 CFR Part 50). On the basis of our review of the documentation supporting this commitment, we have concluded that the proposed facility described in the SNUPPS PSAR and the Wolf Creek Site Addendum Report can be designed, constructed and operated to meet the requirements of the General Design Criteria. Discussions regarding compliance with each criterion are presented in Section 3.1 of the SNUPPS PSAR.

3.2 Classification of Structures, Systems and Components

3.2.1 Seismic Classification

Structures, systems and components important to safety that are required to be designed to withstand the effects of a safe shutdown earthquake and remain functional have been properly classified as seismic Category I items. These plant features are those necessary to assure (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shut down the reactor and maintain it in a safe shutdown condition, or (3) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of 10 CFR Part 100.

All other structures, systems and components that may be required for operation of the facility are designed to other than seismic Category I requirements. Included in this classification are those portions of seismic Category I systems which are not required to perform a safety function. Structures, systems and components important to safety that are designed to withstand the effects of a safe shutdown earthquake and remain functional have been identified in an acceptable manner in Table 3.2-1 of the SNUPPS PSAR.

The basis for our acceptance has been conformance of the applicants' designs, design criteria, and design bases for structures, systems and components important to safety with the Commission's regulations as set forth in Criterion 2 of the General Design Criteria, and to Regulatory Guide 1.29, "seismic Design Classification," Commission staff positions, and industry standards.

We conclude that structures, systems and components important to safety that are designed to withstand the effects of a safe shutdown earthquake and remain functional have been properly classified as seismic Category I items in conformance with the Commission's regulations, the applicable regulatory guide, Commission staff positions, and industry standards and are considered acceptable.

3.2.2 System Quality Group Classification

Fluid system pressure-retaining components important to safety will be designed, fabricated, erected and tested to quality standards commensurate with the importance of the safety function to be performed. The applicants have applied the American Nuclear Soceity classification system (Safety Classes 1, 2, 3 and NNS) in accordance with the American National Standards Institute Standard N18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," which corresponds to the Commission's Quality Groups A, B, C and D in Regulatory Guide 1.26, "Quality Group Classifications and Standards," to those fluid containing components which are part of the reactor coolant pressure boundary and other fluid systems important to safety where reliance is placed on these systems to: (1) prevent or mitigate the consequences of accidents and malfunctions originating within the reactor coolant pressure boundary, (2) permit shutdown of the reactor and maintenance in the safe shutdown conditions, and (3) contain radioactive material. These fluid systems have been classified in an acceptable manner in Tables 3.2-1 and 3.2-2 of the SNUPPS PSAR and on system Piping and Instrumentation Diagrams in the SNUPPS PSAR.

The basis for our acceptance has been conformance of the applicants' designs, design criteria, and design bases for pressure-retaining components such as pressure vessels, heat exchangers, storage tanks, pumps, piping and valves in fluid system important to safety with the Commission's regulations as set forth in Criterion 1 of the General Design Criteria, the requirements of the codes specified in Section 50.55a of 10 CFR Part 50, and to Regulatory Guide 1.26, Commission staff positions, and industry standards.

We conclude that fluid system pressure-retaining components important to safety that are designed, fabricated, erected and tested to quality standards in conformance with the Commission's regulations, the applicable regulatory guide, Commission staff positions and industry standards are acceptable. Conformance with these requirements provides reasonable assurance that the plant will perform in a manner providing adequate safeguards to the health and safety of the public.

3.3 Wind and Tornado Design Criteria

All seismic Category I structures, which are exposed to wind forces, will be designed to withstand the effects of forces imposed by the design wind. The design wind specified for the site has a velocity of 100 miles per hour based on a recurrence interval of 100 years which we find acceptable.

The procedures that will be used to transform the wind velocity into pressure loadings on structures and the associated vertical distribution of wind pressures and gust factors are in accordance with Bechtel Power Corporation's Topical Report BC-TOP-3, "Tornado and Extreme Wind Design Criteria for Nuclear Power Plants," which has been

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reviewed and found acceptable by the staff as documented in the Commission's letter dated October 4, 1974. The design wind loads will be combined with other applicable loads as discussed in Section 3.8 of this report.

All seismic Category I structures, which are exposed to tornado forces, will be designed to withstand the effects of the design basis tornado. All seismic Category I systems and components located within these structures will, therefore, be protected from the effects of a postulated tornado. The design basis tornado specified for the site has a tangential wind velocity of 290 miles per hour and a translational velocity of 70 miles per hour. The simultaneous atmospheric pressure drop associated with the tornado is three pounds per square inch in one and one half seconds. These values for the design basis tornado are in compliance with Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants," and are, therefore, acceptable.

The procedures that will be used to transform the tornado wind velocity into pressure loadings are similar to those to be used for the design wind loadings, as discussed in Bechtel Topical Report BC-TOP-3, with the exception that no gust factors will be used and no change in velocity with height will be assumed. The pressure drop associated with the design tornado will be treated as a static load. The total effect of the design basis tornado on seismic Category I structures will be of the individual effects of the tornado wind pressure, pressure drop and tornado associated missiles. Tornado-generated loads will be combined with other applicable loads as discussed in Section 3.8 of this report. Structures will be arranged on the plant site and protected in such a manner that collapse of structures not designed for the tornado will not affect other safety-related structures.

We conclude that the procedures to be utilized to determine the loadings on seismic Category I structures induced by the design wind and the design basis tornado specified for the plant are acceptable since these procedures provide a conservative basis for engineering design to assure that the structures will withstand such environmental forces.

The use of these procedures provides reasonable assurance that in the event of the design wind or the design basis tornado, the structural integrity of seismic Category I structures will not be impaired and, in consequence, safety-related systems and components located within these structures will be adequately protected and will perform their intended safety functions if needed. Conformance with these procedures is an acceptable basis for satisfying the applicable requirements of Criterion 2 of the General Design Criteria.

3.4 Water Level (Flood) Design Criteria

The finished plant grade elevation of the Wolf Creek plant will be above the probable maximum flood elevation of the site. Therefore, no above grade flood protection measures will be required for the safety-related structures at the Wolf Creek plant site. The seismic Category I essential service water system's pump house will be designed for dynamic forces resulting from the probable maximum precipitation coincident with wave activity in the cooling lake. Below-grade penetrations will be provided with water proof seals to protect against postulated groundwater intrusion.

We have reviewed the applicants' procedures that will be used to determine the loadings on seismic Category I structures induced by the design flood or highest groundwater level specified for the plant. We conclude that these procedures are acceptable since, they provide a conservative basis for engineering design to assure that the structures will withstand such environmental forces.

The use of these procedures provides reasonable assurance that in the event of floods or high groundwater, the structural integrity of the plant seismic Category I structures will not be impaired and, consequently, seismic Category I systems and components located within these structures will be adequately protected and will perform their intended functions, if needed. Conformance with these design procedures is an acceptable basis for satisfying the applicable requirements of Criterion 2 of the General Design Criteria.

3.5 Missile Protection

The plant will be designed so that postulated missiles generated from internal sources and from outside of containment do not cause or increase the severity of an accident.

The applicants have considered missiles generated by pressurized components. To protect the essential systems and components from the damaging effects of these missiles, compartmentation, restraints, separation, orientation and/or missile barriers will be provided. The criteria to be used to design structures for missile impact are described in Section 3.5 of the SNUPPS PSAR. The applicants' criteria regarding the protection of essential structures and vital equipment from internally generated missiles are in accordance with Criterion 4 of the General Design Criteria. Based on the above, we conclude that the applicants can develop an acceptable facility design to prevent internal missiles from damaging structures and equipment required for the safe shutdown of the plant.

The originally proposed design of the safety-related structures for the SNUPPS plants included provisions for 24-inch walls and 18-inch roofs, using reinforced concrete with a comprehensive strength of 4000 pounds per square inch to withstand the impacts of tornado generated missiles. In order to be consistent with our requirement that the roofs of seismic Category I structures be designed to withstand the impact velocity of a tornado missile equivalent to 80 percent of the horizontal tornado missile velocity,

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the applicants have committed to revise the design by either increasing the roof thickness from 18 inches to 21 inches using concrete with a compressive strength of 4000 pounds per square inch, or by using concrete with a compressive strength of 5000 pounds per square inch for the 18-inch roof thickness. We find this commitment acceptable.

Based on this commitment and since the proposed wall and roof thicknesses for the seismic Category I structures are at least equivalent to those for plants which we have reviewed and approved, we conclude that the proposed facility is acceptable from the standpoint of tornado missile protection.

We are currently evaluating tornado missiles on a generic basis. Should the generic resolution of our evaluation demonstrate the adequacy of the original roof design for the seismic Category I structures, we will permit the applicants to revise the design in accordance with the new requirements.

The Wolf Creek plant will be arranged such that the turbine-generator is in a peninsular orientation with respect to the containment building. We have concluded that the turbine orientation is acceptable and that no additional provisions for protection against turbine missiles are necessary.

3.6 Criteria for Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping

For a discussion of the postulated ruptures of the reactor coolant system piping, the applicants incorporate by reference Section 3.6 of RESAR-3, Consolidated Version. Section 3.6 of the SNUPPS PSAR is concerned with the dynamic and environmental effects of postulated ruptures in other high energy piping systems outside the reactor coolant system boundary; i.e., those piping systems having an operating temperature higher than 200 degrees Fahrenheit or having an operating pressure higher than 275 pounds per square inch gauge.

3.6.1 Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping Inside Containment

The criteria to be used by the applicant for identifying high energy fluid piping and for postulating pipe break locations, break orientations and break flow areas inside containment are consistent with the criteria set forth in Regulatory Guide 1.46, "Protection Against Pipe Whip Inside Containment."

These provisions for protection against the dynamic effects associated with pipe ruptures and the resulting discharging coolant provide acceptable assurance that, in the event of the occurrence of the combined loadings imposed by an earthquake of the magnitude specified for the safe shutdown earthquake and a concurrent single pipe break of the largest pipe at one of the design basis break locations, the following conditions and safety functions will be accommodated and assured:

(1) The magnitude of the design basis loss-of-coolant accident cannot be aggravated by potential multiple failures of piping.

(2) The reactor emergency core cooling systems can be expected to perform their intended function.

(3) The containment structure's leak-tight integrity can be expected to be maintained in order to contain, within the leakage limits of the containment, any radioactive materials released from the discharging coolant into the containment atmosphere.

Section 3.6 of RESAR-3, Consolidated Version incorporates by reference Westinghouse Topical Report WCAP 8082, "Pipe Breaks for the LOCA Analysis of the Westinghouse Primary Coolant Loop." This topical report describes proposed quantitative protection criteria for the reactor coolant system piping to provide an equivalent level of protection to that recommended in Regulatory Guide 1.46. The Commission's staff has reviewed this report and found it acceptable as documented in the Commission's letter, dated May 22, 1974. We concur with the applicants that the design of the reactor coolant system piping is similar to the design used in Topical Report WCAP 8082 and that the stresses in the system will not exceed those of the system described in WCAP 8082.

For the design loading of piping and piping restraints following a rupture in high energy piping systems other than the reactor coolant system, the applicants incorporate by reference Bechtel Power Corporation Topical Report BN-TOP-2, Revision 2, "Design for Pipe Break Effects." The staff has also reviewed Topical Report BN-TOP-2 and found it acceptable as documented in the Commission's letter, dated June 17, 1974.

On the basis of our review, we conclude that the criteria that will be used for the identification, design and analysis of piping systems where postulated breaks may occur constitute an acceptable design basis in meeting the applicable requirements of Criteria 1, 2, 4, 14 and 15 of the General Design Criteria.

3.6.2 Protection Against Dynamic Effects Associated with the Postulated Rupture of High Energy Piping Outside Containment

Plant design criteria to be applied to the design of the facility will require the design to accommodate the effects of postulated pipe breaks and jet impingement from piping systems. Where possible or practical, the plant general arrangement and the layout designs of high energy systems will utilize physical separation. In instances where the separation or isolation criteria of the Commission's July 12, 1973 letter cannot be met, pipe whip restraints will be employed.

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We reviewed the applicants' proposed design criteria and bases for protection against postulated piping failures in fluid systems outside conta ent and found that they were not consistent with our positions and therefore, found them unacceptable. The basis for our conclusion is the applicants' exclusion of certain moderate energy systems from postulated leaks and lack of sufficient detail of the proposed criteria to permit proper evaluation. Therefore, we required that the applicants commit to design the plant in accordance with our staff Technical Position APCSB 3-1, titled "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," which is included with Standard Review Plan Section 3.6.1, "Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," or to the criteria described in the Commission's letter of July 12, 1973. The applicants have committed to design the plant in accordance with the staff technical Position APCSB 3-1.

On the basis of this commitment, we now conclude that the criteria that will be used for identification and protection against postulated piping failures in fluid systems outside containment are acceptable.

3.7 Seismic Design

3.7.1 Seismic Input

The input seismic design response spectra to be applied in the design of seismic Category I structures, systems, and components comply with the recommendations of Regulatory Guide 1.60 "Design Response Spectra for seismic Design of Nuclear Power Plants." The damping values to be used in the seismic system analysis of Category I structures, systems and components are in conformance with Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants."

The applicants have proposed higher damping values than those listed in Regulatory Guide 1.61 for component analysis of the fuel assemblies and control rod drive mechanisms. In support of these higher damping values, the applicants have provided test data and have committed to provide additional test data in the Final Safety Analysis Report. We find this commitment acceptable and we will complete our review of the component damping values at the operating license stage of our review.

The seismic time history to be used for seismic design of seismic Category I plant equipment will be adjusted in amplitude and frequency to obtain response spectra that envelop the response spectra defined in Regulatory Guide 1.60.

Conformance with Regulatory Guide 1.60 and 1.61 requirements provides reasonable assurance that, for an earthquake whose intensity results in a ground acceleration of 0.12g for the operating basis earthquake, and 0.20g for the safe shutdown earthquake, the system seismic inputs for seismic Category I structures, systems and components are adequately defined to assure a conservative basis for the design of such structures and systems to withstand the consequent seismic loadings. Compliance with the requirements of these guides constitutes an acceptable basis for satisfying the provisions of Criterion 2 of the General Design Criteria.

We conclude that the seismic input criteria proposed by the applicants are acceptable for system seismic design.

3.7.2 Seismic System and Subsystem Analyses

The scope of our review of the seismic system and subsystem analyses included the seismic analysis methods for all seismic Category I structures, systems and components. It included review of procedures for modeling, seismic soil-structure interaction, development of floor response spectra, inclusion of torsional effects, evaluation of seismic Category I structure overturning, and determination of composite damping. The review has included design criteria and procedures for evaluation of interaction of non-Category I structures and piping, with seismic Category I structures and piping, and the effects of parameter variations on floor response spectra. The review has also included criteria and seismic analysis procedures for reactor internals and seismic Category I buried piping outside the containment.

The system and subsystem analyses will be performed by the applicants on an elastic basis. Modal response spectrum multidegree of freedom and time history methods will form the bases for the analyses of all major seismic Category I structures, systems and components. When the modal response spectrum method is used, governing response parameters will be combined by the square root of the sum of the squares rule. However, the absolute sum of the modal responses will be used for modes with closely spaced frequencies. The square root of the sum of the squares of the maximum codirectional responses will be used in accounting for three components of the earthquake motion for both the time history and response spectrum methods. Floor spectra inputs to be used for design and test verifications of structures, systems and components will be generated from the time history method taking into account variation of parameters by peak widening. A vertical seismic system dynamic analysis will be employed for all structures, systems and components where analyses show significant structural amplification in the vertical direction. Torsional effects and stability against overturning will be considered.

The lumped soil spring method will be used to evaluate soil-structure interaction effects. The applicants have also committed to perform a finite element analysis for the essential service water system pump house. The applicants' proposed finite element analysis will consider appropriate nonlinear stress-strain and damping relationships for the soil. If the results of this analysis, when compared to the lumped soil spring method analysis, indicate greater structural response and/or significant frequency shift finite element analyses will be performed for all structures with embedment ratios (depth of embedment to the minimum base dimension) greater than 0.15. We find this commitment acceptable.

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The applicants' proposed criteria for seismic design to account for relative displacements between component support points or piping support points, as discussed in Sections 3.7.2.7 and 3.7.3.6 of the SNUPPS PSAR, are not acceptable. The staff position on acceptable criteria is as follows:

> Where the response spectrum method is used, the acceptable procedure involves two steps: First, using static analysis methods, the response quantity of interest, induced by differential seismic movement of supports, is computed on a modal basis. The total response is determined by combining modal responses according to the method recommended in Regulatory Guide 1.92, "Combination of Modes and Spatial Components in Seismic Response Analysis." Second, a dynamic analysis is made by assuming no relative displacement between support points, but using the worst floor response spectrum when the support points are in the same structure, or the enveloped floor response spectrum when the support points are in separate structures. Results from these two steps, static and dynamic, should be combined in an absolute manner. (For piping components, these results should be used in accordance with the American Society of Mechanical Engineers Boiler and Pressure Vessel Code Section III, Paragraphs NB-3652 and NB-3653.1.)

Therefore we required the applicants to revise their criteria to comply with the staff position. The applicants have submitted documentation providing the revised criteria which are in compliance with our position. Therefore, the revised criteria are acceptable.

We conclude that, the seismic system and subsystem analysis procedures and criteria to be used by the applicants, as modified by our position on accounting for relative displacements between support points, provide an acceptable basis for the system and subsystem seismic design.

3.7.3 Seismic Instrumentation Program

We have reviewed the applicants' proposed seismic instrumentation program. The proposed installation of the specified seismic instrumentation in the reactor containment structure and at other seismic Category I structures, systems, and components constitutes an acceptable program to record data on seismic ground motion as well as data on the frequency and amplitude relationship of the seismic response of major structures and systems. A prompt readout of pertinent data at the control room can be expected to yield sufficient information to guide the operator on a timely basis for the purpose of evaluating the seismic response in the event of an earthquake. Data to be obtained from such installed seismic instrumentation will be sufficient to determine that the seismic analysis assumptions and the analytical model used for the design of the plant are adequate and that allowable stresses are not exceeded under conditions where continuity of operation is intended. Based on our review, we conclude that the seismic instrumentation program complies with Regulatory Guide 1.12, "Instrumentation for Earthquake," and, therefore, is acceptable.

3.8 Design of Category I Structures

3.8.1 Concrete Containment

The reactor coolant system will be enclosed in a prestressed, post-tensioned concrete containment building as described in Section 3.8.1 of the SNUPPS PSAR. The containment will be designed in accordance with the methods described in the Bechtel Power Corporation Topical Report, BC-TOP-5, "Prestressed Concrete Nuclear Reactor Containment Structures" to resist various combinations of dead loads, live loads, and environmental loads including those due to wind, tornadoes, the operating basis and safe shutdown earthquakes, and loads generated by the design basis accident, including pressure, temperature and associated pipe rupture effects. We have reviewed Topical Report, BC-TOP-5 and found it acceptable as documented in the Commission's letter, dated March 28, 1975. Topical Report BC-TOP-5 will also be utilized for the detail design of prestressing tendons, reinforcing steel at penetrations, splices, anchorage zones, connections, joints and other discontinuities.

In response to our inquiry, the applicants have stated that prototype tests applicable to the SNUPPS reactor containment building, will be performed on the containment structure of the San Onofre Unit 2, Docket No. 50-361. The San Onofre Unit 2 containment structure is similar in design to the SNUPPS containment structure, and will be the first of its kind capable of being used as a prototype as defined in the Regulatory Guide 1.18, "Structural Acceptance Test for Concrete Primary Reactor Containment."

We conclude that the criteria to be used in the analysis, design, and construction of the concrete containment structure to account for anticipated loadings and postulated conditions that may be imposed upon the structure during its service lifetime are in conformance with established criteria, codes, standards, guides, and specifications acceptable to the staff.

The use of these criteria as defined by applicable codes, standards, guides and specifications; the loads and loading combinations; the design and analysis procedures; the sturctural acceptance criteria; the materials, quality control programs and special construction techniques and the testing and inservice surveillance requirements provide reasonable assurance that, in the event of winds, tornadoes, earthquakes and various postulated accidents occurring within and outside the containment, the structure will withstand the specified design conditions without impariment of structural integrity or safety function. Conformance with these criteria constitutes an acceptable basis for satisfying the applicable requirements of Criteria 2, 4, 16, and 50 of the General Design Criteria.

3.8.2 Other Category I Structures

The containment interior structures (shield walls, interior walls, compartments and floors), the auxiliary building, the fuel handling building, foundations and other seismic Category I structures will consist of slabs, walls, beams and columns and will be built from structural steel and concrete.

The principal code to be used in the design of concrete seismic Category I structures is the American Concrete Institute (ACI) 318-71 Code, "Building Code Requirements for Reinforced Concrete." For steel seismic Category I structures, the American Institute of Steel Construction (AISC), "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," will be used.

The concrete and steel seismic Category I structures will be designed to resist various combinations, as applicable, of dead loads, live loads, environmental loads including those associated with extreme winds, tornadoes, and earthquakes, and pipe rupture induced loads including loads associated with reaction and jet impingement forces, compartment pressures, and impact effects of shipping pipes.

The proposed design and analysis procedures that will be used for these seismic Category I structures are the same as those approved on previously licensed applications and are in accordance with the applicable procedures delineated in the ACI 318-71 Code for concrete and in the AISC Specification for steel structures. The materials of construction, their fabrication, construction and installation will be in accordance with the ACI 318-71 Code for concrete and with the AISC Specification for steel structures.

We conclude that the criteria used in the analysis, design and construction of all the plant seismic Category I structures to account for anticipated loadings and postulated conditions that may be imposed upon each structure during its service lifetime, are in conformance with established criteria, codes, standards and specifications acceptable to the staff.

The use of these criteria as defined by applicable codes, standards and specifications; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria, the materials, quality control programs and special construction techniques; and the testing and in-service surveillance requirements provide reasonable assurance that, in the event of winds, tornadoes, earthquakes and various postulated accidents occurring within the structure, the structures will withstand the specified design conditions without impariment of structural integrity or the performance of required safety functions. Conformance with these criteria, codes, specifications, and standards constitutes an acceptable basis for satsifying the applicable requirements of Criteria 2 and 4 of the General Design Criteria.

3.9 Mechanical Systems and Components

3.9.1 Dynamic System Analysis and Testing

The applicants have committed to perform a preoperational vibration dynamic effects test program, with procedures to be submitted at the operating license stage of review, to check the vibration performance of piping important to safety. The preoperational vibration dynamic effects test program that will be conducted on safety-related ASME Class 1, 2 and 3 piping systems and their restraints during plant startup and the initial operating conditions constitutes an acceptable program.

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This program will provide adequate assurance that the piping and piping restraints of the system have been designed to withstand vibrational dynamic effects due to valve closures, pump trips, and operating modes associated with the design operational transients. The tests, as planned, will develop loads similar to those experienced during reactor operation. A commitment to proceed with such a program constitutes an acceptable design basis at the construction permit stage of review in fulfillment of the applicable requirements of Criterion 15 of the General Design Criteria.

The applicants have submitted procedures which use acceptable dynamic testing and analysis techniques to confirm the adequacy of mechanical equipment, including their supports which are seismic Category I, to function during and after an earthquake of magnitude up to and including the safe shutdown earthquake. Subjecting the equipment and supports to these dynamic testing and analysis procedures provides reasonable assurance that in the event of an earthquake at the site, the seismic Category I mechanical equipment will continue to function during and after a seismic event. Implementation of these dynamic testing and analysis procedures constitutes an acceptable basis for satisfying the requirements of Criteria 2 and 14 of the General Design Criteria.

3.9.2 ASME Code Class 2 and 3 Components

All seismic Category I pressure retaining systems, components and equipment outside of the reactor coolant pressure boundary, including active pumps and valves, are designed to sustain normal loads, anticipated transients, the operating basis earthquake, and the safe shutdown earthquake within stress limits which are comparable to those outlined in Regulatory Guide 1.48, "Design Limits and Loading Combinations for Seismic Category I Fluid System Components."

The specified design basis combinations of loading, as applied to the design of the safety-related ASME Code Class 2 and 3 pressure-retaining components in systems classified as seismic Category I, provide reasonable assurance that in the event (1) an earthquake should occur at the site, or (2) an upset, emergency or faulted plant transient should occur during normal plant operation, the resulting combined stresses imposed on the system components will not exceed the allowable design stress and strain limits for the materials of construction. Limiting the stresses under such loading combinations provides a conservative basis for the design of the system components to withstand the most adverse combinations of loading events without gross loss of structural integrity. The design load combinations and associated stress and deformation limits specified for all ASME Code Class 2 and 3 components, including the active pumps and valves, constitute an acceptable basis for design in satisfying Criteria 1, 2 and 4 of the General Design Criteria and are consistent with recent staff positions.

In addition to the limits on stress and deformation, the applicants have agreed to utilize an operability assurance program to qualify active ASME Class 2 and 3 seismic Category I pumps and valves. Such a program will include component testing, or a combination of tests and predictive analysis supplemented by seismic qualification testing of motors, operators, and component appendages to provide assurance that such components (1) can withstand postulated seismic loads in combination with other significant loads without loss of structural integrity, and (2) can perform the active function (i.e., valve closure or opening or pump operation) when a safe plant shutdown is to be effected, or the consequences of an accident are to be mitigated. A commitment to develop and utilize a component operability assurance program satisfactory to the staff constitutes an acceptable basis for implementing the requirements of Criterion 1 of the General Design Criteria as they relate to operability of ASME Code Class 2 and 3 active pumps and valves.

The criteria to be used in developing the design and mounting of the safety and relief valves of ASME Code Class 2 systems will provide adequate assurance that, under discharging conditions, the resulting stresses are expected not to exceed the allowable design stress and strain limits for the materials of construction. Limiting the stresses under the loading combinations associated with the actuation of these pressure relief devices provides a conservative basis for the design of the system components to withstand these loads without loss of structural integrity and impairment of the overpressure protection function. The criteria to be used for the design and installation of overpressure protection devices in ASME Code Class 2 systems, are consistent with Regulatory Guide 1.67, "Installation of Overpressure Protection Devices," and constitute an acceptable design basis in meeting the applicable requirements of Criteria 1, 2 and 4 of the General Design Criteria.

3.10 Seismic Qualification of Category I Instrumentation and Electrical Equipment

The applicants have proposed a seismic qualification program that will be implemented for seismic Category I instrumentation and electrical equipment, and the associated supports for this equipment.

For equipment outside the scope of the nuclear steam supply system, the proposed seismic qualification program is consistent with the requirements of the Institute of Electrical and Electronic Engineers (IEEE) Standard 344-1971, "Trial-Use Guide for Seismic Qualification of Class I Electric Equipment for Nuclear Power Generating Stations," as supplemented by the procedures and requirements stated in the staff technical paper, "Electrical and Mechanical Equipment Seismic Qualification Program," included as Appendix B to this report. For this equipment, the general program, as specified in the PSAR, constitutes an acceptable basis for satisfying our requirements and the applicable requirements of Criterion 2 of the General Design Criteria, and is acceptable.

For the equipment which is to be supplied under the scope of the nuclear steam supply system, the applicants proposed a seismic qualification program as described in RESAR-3, Consolidated Version, and as outlined in Westinghouse Topical Reports WCAP-8373, "Qualification of Westinghouse Seismic Testing Procedure for Electrical Equipment Tested Prior to May 1974" and WCAP-7817, "Seismic Testing of Electrical and

Control Equipment." We have determined that the program as described is not acceptable. We have informed the Westinghouse Electric Corporation of the deficiencies in the program, and have stated that we are anxious to resolve these deficiencies. We and Westinghouse are currently discussing this issue to resolve it on a generic basis. The applicants have made a commitment to accept the generic resolution of this issue. We find this commitment to be acceptable.

3.11 Environmental Design of Electrical Equipment

The applicants have made a commitment in the PSAR that all instrumentation, control and electrical equipment outside the scope of the nuclear steam supply system, that is important to safety, will be environmentally qualified in accordance with the provisions of IEEE Standard 323-1974, "IEEE Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations." Therefore, we conclude that the proposed environmental testing program for this equipment is acceptable.

The applicants had stated that all Class IE equipment supplied by the Westinghouse Electric Corporation will be environmentally qualified according to the environmental qualification program described in Westinghouse Topical Report WCAP-7744, "Environmental Testing of Engineered Safety Features Related Equipment (NSSS-Standard Scope)" dated August 1971. The Westinghouse environmental qualification program was evaluated and determined to be unacceptable. We and Westinghouse are currently discussing this issue to resolve it on a generic basis. The applicants have made a commitment to accept the generic resolution of this issue for Class IE equipment supplied under the scope of the nuclear steam system supplier. We find this commitment to be acceptable.

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4.0 REACTOR

4.1 Summary Description

The reactor design for the Wolf Creek plant is identical to that of the other SNUPPS plants and will be essentially the same as the designs reviewed and approved for Comanche Peak, Units 1 and 2 and Millstone, Unit 3. Like Comanche Peak, Units 1 and 2, and Millstone, Unit 3, the SNUPPS plants will use fuel assemblies with a 17 x 17 fuel rod array and a proposed design thermal core power of 3411 megawatts.

To further support the proposed design, Westinghouse has identified an integrated test program to confirm the design margins associated with the 17 x 17 fuel assembly design. These test programs are outlined in Table 4.1. The tests are generally of a thermal, hydraulic and mechanical nature and will be used to verify present design values. These test programs are scheduled for completion during the latter part of 1975.

We are following these test program results for the 17 x 17 fuel assembly. Additional information on the design and nuclear characteristics of this fuel has been received from Westinghouse and is under review. This information is being reviewed for a number of facilities whose operating license applications are now under consideration. These reviews will be completed well before an application for an operating license for the Wolf Creek plant or for any SNUPPS plant is submitted. If the test results demonstrate the need for improvements or modifications of the proposed design, they can be taken into account in the development of the final fuel design of the SNUPPS plants.

4.2 Mechanical Design

4.2.1 Fuel

The SNUPPS fuel element mechanical design, as shown in Table 4.2, is identical to the design currently approved for use in Comanche Peak, Units 1 and 2 and Millstone, Unit 3. The fuel assemblies will consist of two hundred and sixty four fueled rods, twenty four guide thimbles and one instrumentation thimble arranged in a 17 x 17 array. The instrumentation thimble will be at the center of the assemblies, while the guide thimbles will provide channels for inserting various reactivity controls. The reactor fuel rods will consist of slightly enriched uranium dioxide ceramic cylindrical pellets contained in slightly cold worked zircaloy-4 cladding. The fuel rods will be internally pressurized during final welding with helium to minimize compressive stresses during service. The level of prepressurization is designed to both preclude cladding tensile stresses due to a net internal pressure even during anticipated transients and to preclude clad flattening.

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TABLE 4.1

TEST PROGRAMS FOR THE 17 x 17 FUEL ASSEMBLIES

TEST	PURPOSE OF TEST
Rod Cluster Control Spider Tests	Structural adequacy
Grid Tests	Load deflection Static and dynamic buckling Impact damping
Fuel Assembly Structural Tests	Mechanical strength Impact response for blowdown loads
Guide Tube Tests	Guide tube deflection under blowdown loads Fatigue strength Natural frequencies
Prototype Assembly Tests	Assembly pressure drop Hydraulic loads, rod drop time and stall velocity Fuel rod vibration Control rod, drive-line, guide tube and guide thimble wear
Departure from Nucleate Boiling (DNB)	Effect of 17 x 17 geometry on DNB Effect of cold-wall cells in the 17 x 17 geometry
Incore Flow Mixing	Determine the proper value of Thermal Diffusion Coefficient (TDC)
Impile Fuel Densification	Define material characteristics and manufacturing processes to minimize fuel densification (stable fuel)
Loss-of-Coolant Accident (LOCA) Heat Transfer Verification Tests	Determine blowdown, refill and reflood heat transfer coefficients Data for use in small break analysis (bubble rise model)
Delayed Departure from Nucleate Boiling (D^2NB) Test	Determine time that DNB occurs under LOCA conditions Verify Westinghouse transient DNB correlations
Single Rod Burst Test	Determine maximum average assembly flow blockage

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TABLE 4.2

FUEL ELEMENT MECHANICAL DESIGN OF COMANCHE PEAK, UNITS 1 AND 2

MILLSTONE UNIT 3, AND SNUPPS UNITS

DESIGN PARAMETER	VALUE
Rod Array Per Assembly	17 x 17
Rods per Assembly	264
Number of Assemblies	193
Guide Thimbles per Assembly	24
Spacer Grids per Assembly	8
Rod Outside Diameter (inches)	0.374
Clad Wall Thickness (inches)	0.0225
Ratio of Diameter-to-Wall Thickness	16.6
Pellet Theoretical Density (percent)	95
Pellet Length (inches)	0.530
Fuel Column Height (inches)	144
Fuel Weight (pounds)	222,739

In our evaluation of the fuel thermal performance, we assume that densification of uranium dioxide fuel pellets may occur during irradiation in power reactors. The initial density of the fuel pellets and the size, shape, and distribution of pores within the fuel pellet influence the densification phenomenon. The effects of densification on the fuel rod will increase the stored energy, increase the linear thermal output, increase the probability for local power spikes, and decrease the thermal conductance.

The primary effects of densification on the fuel rod mechanical design are manifested in calculations of time-to-collapse of the cladding and fuel-cladding gap conductance. Time-to-collapse calculations predict the time required for unsupported cladding to become dimensionally unstable and to flatten into an axial gap caused by fuel pellet densification. Gap conductance calculations predict the decrease in thermal conductance due to opening of the fuel-clad radial gap.

The analytical models employed by the applicants have been shown to be acceptable by comparison with measurements on fuel rods which have been subjected to reactor operating conditions. These models, described in Westinghouse Topical Reports WCAP-7982, "Fuel Densification Penalty Models," and WCAP-8218, "Fuel Densification-Experimental Results and Model for Reactor Application," are based on data for fuel similar to that proposed for use in the SNUPPS units. We have reviewed these analytical models and conclude that they provide acceptable assessments of the anticipated fuel rod behavior as documented in the Commission's letter dated June 25, 1974.

On the basis of our review of the current analytical models and the confirmatory results from tests on irradiated fuel rods, we have concluded that the fuel rod mechanical design will provide acceptable engineering safety margins for normal operation, and the effects of densification will be acceptably accounted for in the fuel design.

4.2.2 Reactor Vessel Internals

We have reviewed the selection of materials proposed for the reactor vessel internals and components relied upon for providing reactor shutdown and adequate core cooling. All materials are compatible with the reactor coolant, and have performed satisfactorily in similar applications. Undue susceptibility to intergranular stress corrosion cracking will be prevented by avoiding the use of sensitized stainless steel in accordance with the methods recommended in Regulatory Guide 1.44, "Control of the Use of Sensitized Steel." The use of materials proven to be satisfactory by actual service experience, and avoiding sensitization by the methods recommended in Regulatory Guide 1.44, will provide reasonable assurance that the reactor vessel internals will not be susceptible to failure by corrosion or stress corrosion cracking.

The applicants have described the measures that will be taken to assure that deleterious hot cracking of austenitic steel welds is prevented. All weld filler metal will be of selected composition and welding processes will be controlled to produce welds with at least five percent delta ferrite in conformance with the recommendation in Regulatory Guide 1.31, "Control of Stainless Steel Welding," and the requirements of the ASME Code. We conclude that following these recommendations and requirements will provide reasonable assurance that no deleterious hot cracking will be present that could contribute to loss of integrity or functional capability of the reactor internals.

The applicants have referenced the Indian Point Unit 2 reactor (Docket No. 50-247) as the prototype design for a four-loop plant on which vibrational testing has been performed. The SNUPPS internals design is very similar to the Indian Point Unit 2 internals design. Two differences in the SNUPPS design are (1) the use of the Westinghouse 17 x 17 fuel assembly instead of the Westinghouse 15 x 15 fuel assembly and (2) the replacement of the circular thermal shield with neutron shielding pads.

To evaluate these differences in design, Westinghouse is presently planning to instrument and test the internals of the four-loop Trojan Nuclear Plant (Docket No. 50-334). The Trojan plant internals design will include the Westinghouse 17 x 17 fuel array and the neutron shielding pads. Hence the vibrational tests on the Trojan plant will provide the full-scale verification of the modified internals design, which is similar to that of the SNUPPS reactor internals. These tests are scheduled to take place before any of the SNUPPS plants become operational. If, however, the tests can not be performed on a suitable time scale at the Trojan Plant, or at any other applicable plant, before any SNUPPS plant becomes operational, the applicants have committed to implement a vibrational test program at their facility. We find this commitment acceptable.

The applicants have proposed additional confirmatory vibration testing and subsequent visual inspection as part of the preoperational tests to provide added confirmation of the capability of the structural elements of the reactor internals to sustain flow-induced vibrations. The proposed program is consistent with Regulatory Guide 1.20, "Vibration Measurements on Reactor Internals."

We have reviewed the preoperational vibration test program proposed by the applicants for verifying the design adequacy of the reactor internals under loading conditions that will be comperable to those experienced during operation. The combination of tests, conducted at Indian Point Unit 2 and planned for the Trojan plant, predictive analysis, and post-test inspection will provide adequate assurance that the reactor internals can be expected to withstand flow-induced vibrations without loss of structural integrity during their service lifetime. We conclude that the proposed preoperational vibration test program constitutes an acceptable basis for demonstrating the design adequacy of the reactor internals in satisfying the applicable requirements of Criteria 2 and 14 of the General Design Criteria for other than prototype reactors.

4.2.3 Design Analysis for Loss-of-Coolant Accident Loadings

The applicants will perform a dynamic system analysis of the reactor internals to provide an acceptable basis for confirming the structural design adequacy of the reactor internals to withstand the combined dynamic effects of the postulated occurrence of a loss-of-coolant accident and a safe shutdown earthquake.

We have reviewed the analytical methods and find (1) that the analysis will provide adequate assurance that the combined stresses and strains on reactor internals will not exceed the allowable design stress and strain limits for the materials of construction as specified in Appendix F to Section III of the ASME Boiler and Pressure Vessel Code, and (2) that the resulting deflections or displacements of any structural elements of the reactor internals will not distort the reactor internals geometry to the extent that core cooling can be impaired.

The assurance of structural integrity of the reactor internals under the postulated safe shutdown earthquake and the most severe loss-of-coolant accident conditions provides added confidence that the design can be expected to withstand a spectrum of lesser pipe breaks and seismic loading combinations.

We conclude that the use of the proposed analytical techniques will result in an acceptable structural design for the reactor internals.

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4.3 Nuclear Design

The nuclear design of the SNUPPS units is the same as that of Comanche Peak, Units 1 and 2 and Millstone, Unit 3. The Preliminary Safety Analysis Reports for the Comanche Peak and Millstone, Unit 3 plants reference RESAR-3 Consolidated Version for the nuclear design section.

The design bases presented for the nuclear design of the fuel and reactivity control systems comply with all applicable requirements of the General Design Criteria and are acceptable. Descriptions of the fuel assembly enrichments, physics of the fuel burnout process, burnable poison distributions, soluble boron concentrations, delayed neutron fraction, and neutron lifetimes have been provided. The values presented for these parameters meet the design bases and are acceptable.

4.3.1 Power Distributions

We have reviewed the expected power distributions, both typical and limiting, which include the radial, assembly, local rod, and axial power distributions. The associated peaking factors are also discussed. For the SNUPPS units, the 17 x 17 fuel assembly furnishes a lower power density than did the 15 x 15 fuel assembly used in some previous plant designs. The SNUPPS core-average linear power density at full power is 5.43 kilowatts per foot, the same as for Comanche Peak, Units 1 and 2 and Millstone, Unit 3. A design limit overall peaking factor of 2.33 has been established for the SNUPPS units. The applicants have calculated the power distributions expected during both steady-state and typical load-follow operations to show that the actual peaking factor can be maintained below the design value. An allowance for a calculational error of five percent in the expected peaking factors was determined by Westinghouse from comparisons between measured and calculated power distributions. The comparison between expected and design peaking factors demonstrates that the plant can be operated below the design values. Thus, the design peaking factors are appropriate for use in the accident analyses.

We have reviewed the nuclear instrumentation which will be provided to measure the core power distribution. The reactor will be provided with two types of monitoring systems: a system of movable incore fission chamber detectors and a system of fixed ion chambers located symmetrically around the core outside the reactor pressure vessel. The movable incore detectors will be capable of measuring the fuel rod peaking factor to within five percent and will be used to make periodic incore maps of the power distribution. The ion chambers located outside the reactor pressure vessel will provide total power as measured by neutron flux, relative power in each quadrant of the core, and the relative power in the top and bottom of the core. The axial power offset, as measured from the relative power in the top and bottom of the core, and the radial tilt will be used to maintain the overall peaking factor below 2.33.

The proposed power distribution monitoring procedure is constant axial offset control using excore detectors. This is identical to the procedure currently in use at operating Westinghouse reactors and which has been reviewed and approved in several recent operating license cases. The intent of the constant axial offset control is to maintain the axial power distribution and, therefore, the axial xenon distribution, constant as a function of power level. This limits the magnitude of axial xenon transient effects on the overall peaking factor. This is achieved by restricting operation to a $\pm$ 5 percent band about a value of flux difference (between the readings of the upper and lower excore detectors) associated with full power operation of an essentially unrodded core. Above 90 percent of full power, the flux difference must be maintained within the operating band. Between 50 percent and 90 percent of full power, the flux difference may be out of this band no longer than one hour in any 24 hour period.

Westinghouse has informed us that the use of part length control rods, which are included in the SNUPPS core design, in the constant axial offset control procedure has not been fully analyzed. Therefore, part length control rods are not currently being used in operating Westinghouse plants. However, we expect that the use of part length control rods in the constant axial offset control procedure will be fully analyzed before any SNUPPS unit applies for an operating license. Until this item is resolved, however, the use of part length control rods in the SNUPPS plant is prohibited.

4.3.2 Reactivity Coefficients

The reactivity coefficients reflect the changes in the neutron multiplication due to varying core conditions such as power, temperature, pressure, and void changes. These coefficients vary with fuel burnup. The applicants have presented the values for these coefficients that have been employed in the analyses of both normal and transient reactor operation. These coefficients are also used in the accident analyses presented in Section 15.0 of the PSAR. The total power coefficient is strongly negative for all reactor conditions throughout core life. Therefore, the requirements of Criterion 11 of the General Design Criteria are satisfied. The applicants have agreed to measure the moderator temperature coefficient and the power coefficient during startup tests to check the calculated values and to assure that conservative coefficient values were used in the accident analysis. This practice has been used in previously licensed plants and is acceptable.

4.3.3 Control and Control Requirements

To allow for changes of reactivity due to reactor heatup, changes in operating conditions, fuel burnup and fission product buildup, a significant amount of excess reactivity is built into the core. The applicants have provided sufficient information relating to core reactivity balance for the first core, and have shown that means are incorporated into the design to control excess reactivity at all times. Control is achieved with movable control rods and through the variation of boron concentration in the reactor coolant.

The plant will be operated at steady-state full power with most of the full-length control rods withdrawn. Limited insertion of the full-length control rods will permit compensating for fast reactivity changes (e.g., the effects of minor variations in moderator temperature and boron concentrations) without impairing shutdown capability. Part-length control rods are provided to control the axial power distributions. Restrictions upon their use were previously discussed in Section 4.3.1. Sufficient additional control rod worth has been provided to accommodate the reactivity effects of the most limiting accident (steam line break) at any time during the core life with an allowance for the most reactive control rod assembly stuck in the fully withdrawn position.

Soluble boron poison will be used to compensate for slow reactivity changes including those associated with fuel burnup, changes in xenon and samarium concentration, buildup of long-life fission products, burnable poison rod depletion, and the large moderator temperature change from cold shutdown to hot standby. The soluble boron poison system provides the capability to take the reactor to at least 10 percent subcritical and maintain it in the cold shutdown condition at any time during the core life.

This combination of control system satisfies the requirements to Criterion 26 of the General Design Criteria. On the basis of our review, we have concluded that the applicants' assessment of reactivity control requirements is suitably conservative, and that adequate negative reactivity worth has been provided by the control system to assure shutdown capability.

4.3.4 Control Rod Patterns and Reactivity Worths

The full-length control rod assemblies are used for control and shutdown. Load changes will be made with the control rods and/or the soluble poison system. Rod insertion will be controlled by the power-dependent insertion limits that will be defined in the technical specifications. These limits will (1) assure that there is sufficient negative reactivity available to permit the rapid shutdown of the reactor with ample margin, (2) assure that the worths of control rods that might be ejected in the unlikely event of an ejected rod accident will be no worse than that assumed in the accident analyses, and (3) along with the power distribution control procedure, assure that the axial peaking factor does not exceed the limiting value used for the accident analyses. The shutdown rods, which are never inserted during operation, are required to assure a rapid reactor shutdown.

We have reviewed the calculated rod worths and the uncertainties in these worths, and conclude that rapid shutdown capability exists at all times in core life assuming the most reactive control rod assembly is stuck in the fully withdrawn position. The estimate of uncertainties is based upon appropriate comparison of calculations with experiments.

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4.3.5 Stability

We have reviewed the stability of the reactor to xenon-induced power distribution oscillations and the proposed control of such transients. Due to the negative power coefficient, the reactor is inherently stable to oscillations in reactor power. Also, the control and protection system, described in Section 7.7 of RESAR-3, Consolidated Version, provides adequate protection against total power instabilities. The core is calculated to be stable against X-Y xenon oscillations throughout core life. Westinghouse has agreed to verify this stability in a startup physics test for a 193 fuel assembly core similar to SNUPPS. The core is stable to axial xenon oscillations until a core exposure of 12,000 megawatts days per ton is reached. We conclude that sufficient information has been provided to show that axial oscillations will be detected and controlled before any safety limits are reached, thus preventing any fuel damage.

4.3.6 Analytical Methods

The applicants have described the computer programs and calculational techniques used to calculate the nuclear characteristics of the reactor design and have provided examples to demonstrate the ability of these methods to predict experimental results. On the basis of our review, we conclude that the information presented adequately demonstrates the ability of these analytical methods to calculate the reactor physics characteristics of the SNUPPS core.

4.4 Thermal and Hydraulic Design

The SNUPPS reactors will be designed to operate at core power levels of up to 3411 megawatts thermal, which corresponds to a net electrical output of about 1120 megawatts electric. We have evaluated the thermal-hydraulics on the basis of 3411 megawatts thermal. The SNUPPS units will utilize a 17 x 17 fuel assembly, similar to the Comanche Peak plant and the thermal and hydraulic design parameters for the two plants are very similar. A comparison of the thermal and hydraulic design parameters for the SNUPPS, Comanche Peak and Trojan plants is presented in Table 4.3 of this report.

The principal criterion for the thermal-hydraulic design of a reactor is the prevention of fuel rod damage by providing adequate heat transfer for the various core heat generation patterns occurring during normal operation and operational transients, and transient conditions resulting from faults of moderate frequency.

Operating conditions are selected to assure hydraulic stability within the core, thus preventing premature departure from nucleate boiling. Preservation of nucleate boiling as the mode of heat transfer between the hot spot of the fuel cladding and the coolant not only assures that the cladding temperature is only slightly greater than that of the coolant, but that the fuel center line temperature will not reach the melting temperature. The applicants' criterion for overpower protection requires that the maximum fuel center line temperature be less than that of the fuel melting temperature at a peak core power generation rate of 21.1 kilowatts per foot during all modes of operation. To demonstrate the fulfillment of this objective, the applicants

TABLE 4.3

COMPARISON OF THERMAL HYDRAULIC DESIGN PARAMETERS FOR THE SNUPPS, COMANCHE PEAK, AND TROJAN PLANTS

	SNUPPS	COMANCHE PEAK	TROJAN (17 X 17) with densification
Reactor Core Heat Output (MWt)	3411	3411	3411
System Pressure, Nominal (psia)	2250	2250	2250
Minimum DNBR for Design Transients	>1.30	>1.30	>1.30
Total Thermal Flow Rate (lb/hr)	142.2×10^6	142.2×10^6	132.7×10^6
Effective Flow Rate for Heat Transfer	135.8×10^6	135.8×10^6	126.7×10^6
Average Velocity Along Fuel Rods (ft/sec)	16.8	16.8	15.7
Average Mass Velocity (lb/hr-ft^2)	2.66×10^6	2.66×10^6	2.48×10^6
Coolant Temperature (°F)			
Design Nominal Inlet	557.3	557.3	552.5
Average Rise In Core	62.3	62.3	66.9
Active Heat Transfer Surface Area (ft^2)	59,900	59,900	59,700
Average Heat Flux (Btu/hr-ft^2)	189,400	189,400	189,800
Maximum Heat Flux (Btu/hr-ft^2)	454,600	454,600	474,500
Maximum Thermal Output for Normal Operation (KW/ft)	13.0	13.0	13.6
Fuel Central Temperature at Beginning of Life, Maximum at 100% Power (°F)	3250	3250	3400
Fuel Central Temperature at Beginning of Life and Maximum Thermal Output of 21.1 KW/ft (°F)	4400	4400	(not available)

Where:

MWt = megawatts, thermal

psia = pounds per square inch absolute

DNBR = departure from nucleate boiling ratio

lb/hr = pounds per hour

ft/sec = feet per second

lb/hr-ft^2 = pounds per hour per square foot

°F = degrees Fahrenheit

ft^2 = square feet

Btu/hr-ft^2 = British thermal units per hour per square foot

KW/ft = kilowatts per foot

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have calculated a fuel center line temperature of 4400 degrees Fahrenheit for the beginning of life conditions at 21.1 kilowatts per foot compared to a fuel melt temperature of 5080 degrees Fahrenheit (unirradiated).

For recently reviewed Westinghouse designed reactors, the THINC computer code has been used to calculate core thermal-hydraulic performance characteristics. The code considers cross-flow between adjacent assemblies in the core and thermal diffusion between adjacent subchannels in the assemblies. The effect of local power distributions is considered. As a result of these considerations, the THINC code permits the computation of more realistic power shapes than those that had been available from previously used computer codes. These power shapes are especially important at the design conditions.

We will require, as we have in other instances wherein the design is based on this type of analysis, that the acceptability of the thermal, hydraulic, and nuclear feedback calculations be verified by confirmatory tests. We will review the results of the tests and analyses as they become available. Should the test results fail to cover the anticipated range of conditions predicted, the applicants will be required to perform the necessary additional tests during the startup of the Wolf Creek plant.

On the basis of our review of the analytical techniques applied to the previously reviewed and approved 15 x 15 core designs, we have concluded that for the 17 x 17 core design, there is reasonable assurance that (1) the proposed thermal-hydraulic design accounts for departure from nucleate boiling and fuel center line temperature limitations in a satisfactory manner, and (2) the conservatism in the thermal-hydraulic design procedures can be verified. Therefore, we have concluded that the thermal-hydraulic design of the SNUPPS reactors is acceptable.

In the event that sufficient verification cannot be obtained from the test program or that the analytical methods are not conservative, appropriate restrictions on operations can be established at the operating license stage of review.

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5.0 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.1 Summary Description

The reactor coolant system for the Wolf Creek plant is described in Section 5.0 of RESAR-3, Consolidated Version, which the applicants incorporate by reference, and will be the same for all four SNUPPS plants. The reactor coolant system will include a reactor vessel and four coolant loops connected in parallel to the vessel. Each loop will be equipped with a coolant pump and a steam generator. A pressurizer will be connected to one of the loops. The system is very similar to those systems previously approved for Millstone, Unit 3 and the Comanche Peak, Units 1 and 2 with the exception that the Millstone unit will have loop stop valves.

5.2 Reactor Coolant Pressure Boundary

5.2.1 Component Design

Components of the reactor coolant pressure boundary, as defined by the rules of Section 50.55a of 10 CFR Part 50, have been properly identified and classified as Class I components in accordance with Section III of the American Society of Mechanical Engineers, "ASME Boiler and Pressure Vessel Code." These components within the reactor coolant pressure boundary will be constructed in accordance with the requirements of the applicable codes and addenda as specified by the rules of Section 50.55a of 10 CFR Part 50. We conclude that construction of the components of the reactor coolant pressure boundary in conformance with the Commission's regulations provides reasonable assurance that the resulting quality standards are commensurate with the importance of the safety function of the reactor coolant pressure boundary and, therefore, is acceptable.

The applicants have made a commitment that no ASME code cases considered unacceptable to the Commission will be applied in the construction of pressure-retaining ASME Code Class 1 components within the reactor coolant pressure boundary (Quality Group Classif cation A). The applicants have also stated their intent to comply with Regulatory Guides 1.84, "Code Case Acceptability - ASME Section III Design and Fabrication," and 1.85, "Code Case Acceptability - ASME Section III Materials." In the event the use of new code cases approved by the ASME Council are planned, the applicants will request Commission authorization prior to their application in the construction of ASME Code Class I components. We conclude that compliance with the Commission's regulations in the use of approved code cases will result in a component quality level commensurate with the importance of the safety function of the reactor coolant pressure boundary and, therefore, is acceptable.

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The design loading combinations specified for ASME Code Class 1 reactor coolant pressure boundary components have been appropriately categorized with respect to the plant condition identified as normal, upset, emergency or faulted. The non axisymmetric pressure loads (reactor internals response and cavity pressurization) that would result from a postulated pipe rupture at the reactor pressure vessel cold leg nozzle will be combined with other resulting mechanical hydraulic loads to assure a conservative design basis for the reactor vessel support system. The design limits proposed by the applicants for these plant conditions are consistent with the criteria recommended in Regulatory Guide 1.48. Use of these criteria for the design of the reactor coolant pressure boundary components will provide reasonable assurance that in the event an earthquake should occur at the site, or other system upset, emergency or faulted conditions should develop, the resulting combined stresses imposed on the system components will not exceed the allowable design stresses and strain limits for the materials of construction. Limiting the stresses and strains under such loading combinations provides a basis for the design of the system components for the most adverse loadings postulated to occur during the service lifetime without loss of the system's structural integrity.

The design load combinations and associated stress and deformation limits specified in ASME Code Class 1 components constitute an acceptable basis for design in satisfying the related requirements of Criteria 1, 2 and 4 of the General Design Criteria.

The applicants have identified the active components within the reactor coolant pressure boundary for which operation is required to safely shut down the plant and maintain it in a safe condition in the event of a safe shutdown earthquake or design basis accident.

The applicants have agreed to utilize an operability assurance program, in addition to stress and deformation limits, to qualify active valves. Such a program will include valve testing, or a combination of tests and predictive analysis, supplemented by seismic qualification testing of valve operator systems to provide assurance that active components will withstand the imposed loads associated with normal, upset, emergency and faulted plant conditions without loss of structural integrity, and will perform the active function under conditions comparable to those expected when safe plant operation or shutdown is to be effected, or the consequences of a seismic transient or of an accident are to be mitigated.

A commitment to develop and utilize a component operability assurance program satisfactory to the staff constitutes an acceptable basis for implementing the requirements of Criterion 1 of the General Design Criteria as related to the operability of ASME Code Class 1 active valves.

5.2.2 Overpressurization Protection

Protection of the primary system against overpressurization will be provided by two power operated pressure relief valves and three safety valves. The three safety valves, in conjunction with the steam generator safety valves, will protect the reactor coolant system against overpressure. The relief valves will be designed to limit the pressurizer pressure to a value below the high pressure trip set point for all design transients up to and including the design percentage step load decrease of 95 percent with steam dump but without reactor trip.

The required capacity of the pressurizer safety valves will be determined from consideration of the complete loss of steam flow to the turbine with credit taken for steam generator safety valve operation and maintenance of the main feedwater flow, but with no credit taken for reactor trip. The peak reactor coolant system pressure will be limited to 110 percent of the design value. No credit will be taken for operation of pressurizer relief valves, steam line relief valves, steam dump system, reactor control system, pressurizer level control system, and pressurizer spray. In determining the valve capacity, the plant will be assumed to be operating at a thermal power level of 3565 megawatts.

A loss of load transient will also be analyzed for the case where the main feedwater flow is lost at the same time that steam flow to the turbine is lost. For this transient, the system will be protected against overpressurization by the pressurizer safety valves in conjunction with the reactor protection system.

The overpressure protection devices are the same as those to be used and which were accepted on Millstone, Unit 3 and the Comanche Peak, Units 1 and 2. The overpressure protection system will limit the reactor coolant system pressure to below 110 percent of design pressure which is in accordance with the ASME Code limit and is, therefore, acceptable.

The criteria to be used in developing the design and mounting of the safety and relief valves of ASME Code Class 1 systems will provide adequate assurance that, under discharging conditions, the resulting stresses will not exceed the allowable design stress and strain limits for the materials of construction. Limiting the stresses under the loading combinations associated with the actuation of these pressure relief devices provides a conservative basis for the design of the system components to withstand these loads without loss of structural integrity and impairment of the overpressure protection function.

We conclude that the criteria to be used for the design and installation of overpressure relief devices in ASME Code Class 1 systems comply with Regulatory Guide 1.67, "Installation of Overpressure Protection Devices," ASME Code Case 1569 and Subarticle NB-3600 of Section III of the ASME Code, and constitute an acceptable design basis in meeting the applicable requirements of Criteria 1, 2, 4, 14 and 15 of the General Design Criteria.

5.2.3 Materials Considerations

We have reviewed the proposed materials of construction for the reactor coolant pressure boundary to assure that the possibility of serious corrosion or stress corrosion is minimized. All materials used are compatible with the expected environment, as proven by extensive testing and satsifactory service performance.

Further protection against corrosion problems will be provided by control of the chemical environment. The composition of the reactor coolant will be controlled, and the proposed maximum containment levels, as well as the proposed pH, hydrogen overpressure, and boric acid concentrations, have been shown by tests and service experience to be adequate to protect against corrosion and stress corrosion problems.

We have evaluated the requirements for the external insulation to be used on austenitic stainless steel components and conclude that they are in conformance with Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel."

The possibility that serious corrosion or stress corrosion problems would occur, in the unlikely event that the emergency core cooling system or containment spray system operataion is required, will be minimized because the pH of the circulating coolant will be maintained within acceptable limits by sodium hydroxide additions.

The controls on chemical composition that will be imposed on the reactor coolant and the emergency core cooling water, and the use of external thermal insulation in conformance with Regulatory Guide 1.36 provide reasonable assurance that the reactor coolant pressure boundary materials will be acceptably protected from conditions that would lead to loss of integrity from stress corrosion.

5.2.4 Fracture Toughness

We have reviewed the materials selection, toughness requirements, and extent of materials testing proposed by the applicants to provide assurance that the ferritic materials used for pressure retaining components of the reactor coolant pressure boundary will have adequate toughness under test, normal operation, and transient conditions. The ferritic materials are specified to meet the toughness requirements of the ASME Boiler and Pressure Vessel Code, Section III, Summer 1972 Addenda. In addition, materials for the reactor vessel are specified to meet the additional test requirements and acceptance criteria of Appendix G to 10 CFR Part 50.

The fracture toughness tests and procedures required for the reactor vessel by Section III of the ASME Code, as augmented by Appendix G to 10 CFR Part 50, provide reasonable assurance that adequate safety margins against the possibility of nonductile behavior or rapidly propagating fracture can be established for all pressure retaining components of the reactor coolant pressure boundary.

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The reactor will be operated in accordance with the ASME Code, Section III, Summer 1972 Addenda, and Appendix G to 10 CFR Part 50. This will minimize the possibility of failure due to a rapidly propagating crack. Additional conservatism in the pressure-temperature limits used for heatup, cooldown, testing, and core operation will be provided because the pressure-temperature limits will be determined assuming that the beltline region of the reactor vessel has already been irradiated.

The use of Appendix G of the ASME Code as a guide in establishing safe operating limitations, using results of the fracture toughness tests performed in accordance with the ASME Code and Commission regulations, will assure adequate safety margins during operating, testing, and maintenance, and postulated accident conditions. Compliance with these ASME Code provisions and Commission regulations constitutes an acceptable basis for satisfying the requirements of Criterion 31 of the General Design Criteria.

The toughness properties of the reactor vessel beltline material will be monitored throughout service life with a material surveillance program that will meet the requirements of the American Society for Testing Materials Specification (ASTM) E 185-73, "Surveillance Tests on Structural Materials in Nuclear Reactors," and comply with Appendix H to 10 CFR Part 50.

Changes in the fracture toughness of material in the reactor vessel beltline caused by exposure to neutron radiation will be assessed properly and adequate safety margins against the possibility of vessel failure can be provided if the material requirements of the above documents are met. Compliance with these documents will assure that the surveillance program constitutes an acceptable basis for monitoring radiation induced changes in the fracture toughness of the reactor vessel material, and will satisfy the requirements of Criterion 31 of the General Design Criteria.

Although the use of controlled composition material for the reactor vessel beltline will minimize the possibility that radiation will cause serious degradation of the toughness properties, the applicants have stated that should results of tests indicate that the toughness is not adequate, the reactor vessel can be annealed to restore the toughness to acceptable levels.

5.2.5 Austenitic Stainless Steel

The applicants have stated that the possibility of intergranular stress corrosion in austenitic stainless steel used for components of the reactor coolant pressure boundary will be minimized because sensitization will be avoided, and adequate precautions will be taken to prevent contamination during manufacture, shipping, storage, and construction. The applicants' proposed plans to avoid sensitization are in conformance with Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," and include controls on compositions and heat treatments, including process and cooling rates.

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The use of materials with satisfactory service experience and conformance with Regulatory Guide 1.44 provide reasonable assurance that austenitic stainless steel components will be compatible with the expected service environments, and the probability of loss of structural integrity is minimized.

The applicants have described the measures that will be taken to assure the deleterious hot cracking of austenitic steel welds is prevented. All weld filler metal will be of selected composition and welding processes will be controlled to produce welds with at least five percent delta ferrite in conformance with the recommendations in Regulatory Guide 1.31, "Control of Stainless Steel Welding," and the requirements of the ASME Code.

We conclude that following these recommendations and requirements will provide reasonable assurance that no deleterious hot cracking will be present that could contribute to loss of integrity of austenitic stainless steel welds in the reactor coolant pressure boundary.

5.2.6 Pump Flywheel Integrity

The probability of a loss of pump flywheel integrity, which could result in high energy missiles and excessive vibration of the reactor coolant pump assembly, will be minimized by the use of suitable material, and adequate design and inservice inspection.

The specifications for materials, design, fabrication, and preservice and inservice inspection for the pump will be in accordance with Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity."

The use of suitable material, and adequate design and inservice inspection for the flywheels of reactor coolant pump motors, provide reasonable assurance that the structural integrity of the flywheels is adequate to withstand the forces imposed in the event of pump design overspeed transients without loss of function, and that the integrity of the flywheels will be verified periodically, in service, to assure that the required level of soundness of the flywheel material is adequate to preclude failure. We conclude that compliance with the recommendations of Regulatory Guide 1.14 constitutes an acceptable basis for satisfying the applicable portions of Criterion 4 of the General Design Criteria.

The potential for the reactor coolant pump flywheel to become a missile in the event of a rupture in the pump suction or discharge sections of reactor coolant system piping, is under generic study by Westinghouse and the staff. The Electrical Power Research Institute has contracted Combustion Engineering, Incorporated to perform a 1/5 scale reactor coolant pump research program. The objective of the program will

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be, in part, to obtain empirical data to substantiate or modify current mathematical models used in predicting pump performance during a postulated loss-of-coolant accident. Results from the program are expected by the fall of 1976. We will be following the development and performance of this program as well as other industry analytical and experimental programs on a generic basis.

If the results of the generic investigations of this matter indicate that additional protective measures are warranted to prevent excessive pump overspeed or to limit potential consequences to safety-related eqiupment, we will determine what modifications, if any, are necessary to assure that an acceptable level of safety is maintained. If modifications are necessary, we will require the applicants to make them.

5.2.7 Leakage Detection Systems

We have reviewed the systems proposed by the applicants to detect leakage from components and piping of the reactor coolant pressure boundary into the containment. The leakage detection systems proposed will include diverse leak detection methods, have sufficient sensitivity to measure small leaks, identify the leakage source to the extent practical, and be provided with suitable control room alarms and readouts. The major components of the systems are the sump level and flow monitors, air particulate and condensate monitors, and containment radioactive gas and humidity monitors for indirect indication of leakage.

The leakage detection systems proposed to detect leakage from components and piping of the reactor coolant pressure boundary are in conformance with Regulatory Guide 1.45, "Reactor Coolant Boundary Leakage Detection Systems," and thus provide reasonable assurance that any structural degradation resulting in leakage during service will be detected in time to permit corrective actions.

We conclude that compliance with the recommendations of Regulatory Guide 1.45 constitutes an acceptable basis for satsifying the requirements of Criterion 30 of the General Design Criteria and that the proposed system is acceptable.

5.2.8 Inservice Inspection Program

To assure that no deleterious defects develop during service, selected welds and weld heat-affected zones in the reactor coolant pressure boundary will be inspected periodically. The applicants have stated that the design of the reactor coolant system incorporates provisions for access for inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, and that suitable equipment will be developed to facilitate the remote inspection of these areas of the reactor vessel not readily accessible to inspection personnel.

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We conclude that the conduct of periodic inspections and hydrostatic testing of pressure retaining components in the reactor coolant pressure boundary, in accordance with the requirements of Section XI of the ASME Code, will provide reasonable assurance that evidence of structural degradation or loss of leaktight-integrity occurring during service will be detected in time to permit corrective action before the safety function of a component is compromised. Compliance with the inservice inspections required by Section XI of the ASME Code constitutes an acceptable basis for satisfying the requirements of Criterion 32 of the General Design Criteria.

To assure that no deleterious defects develop during service in ASME Code Class 2 system components in connected systems to the reactor coolant pressure boundary, such as the residual heat removal system, portions of the chemical and volume control system, and engineered safety features not part of ASME Code Class 1 systems, the applicants state that selected welds and weld heat-affected zones will be inspected prior to reactor startup and periodically throughout the life of the plant. These ASME Code Class 2 systems and ASME Code Class 3 systems, such as the component cooling water systems and portions of the radwaste system, will receive visual inspections while the systems are pressurized in order to detect leakage, signs of mechanical or structural distress, and corrosion.

The applicants have stated that the ASME Code Class 2 systems will meet the inspection requirements of Section XI of the ASME Code. The inservice requirements of the ASME Code Class 2 systems and ASME Code Class 3 systems are in conformance with the recommendations of Regulatory Guide 1.51. We conclude that compliance with the inservice inspections required by the ASME Code and Regulatory Guide 1.51, "Inservice Inspection of ASME Code Class 2 and 3 Nuclear Power Plant Components," constitutes an acceptable basis for satisfying Criteria 36, 29, 42, and 45 of the General Design Criteria for these systems.

5.3 Reactor Vessel Integrity

We have reviewed all factors contributing to the structural integrity of the reactor vessel and we conclude there are no special considerations (Commission Memorandum and Order in the Matter of Consolidated Edison Company of New York, Indian Point Unit No. 2, Docket No. 50-247, October 26, 1972) that make it necessary to consider potential vessel failure for the SNUPPS plants.

The bases for our conclusion are that the design, material, fabrication, inspection, and quality assurance requirements will conform to the rules of the ASME Boiler and Pressure Vessel Code, Section III (1971 Edition), including Winter 1972 Addenda and the applicable code cases.

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The fracture toughness requirements of the ASME Boiler and Pressure Vessel Code, Section III, Summer 1972 Addenda will be met. The operating limitations on temperature and pressure will be established for this plant in accordance with Appendix G of the Summer 1972 Addenda and Appendix G to 10 CFR Part 50.

The integrity of the reactor vessel is assured because the vessel:

(1) Will be designed and fabricated to the high standards of quality required by the ASME Boiler and Pressure Vessel Code and pertinent code cases.

(2) Will be made from materials of controlled and demonstrated high quality.

(3) Will be extensively inspected and tested to provide substantial assurance that the vessel will not fail because of material or fabrication deficiencies.

(4) Will be operated under conditions and procedures and with protective devices that provide assurance that the reactor vessel design conditions will not be exceeded during normal reactor operation or during most upsets in operation, and that the vessel will not fail under the conditions of any of the postulated accidents.

(5) Will be subjected to monitoring and periodic inspection to demonstrate that the high initial quality of the reactor vessel has not deteriorated significantly under the service conditions.

(6) May be annealed to restore the material toughness properties if this becomes necessary.

5.4 Loose Parts Monitoring

Occasionally, miscellaneous items such as nuts, bolts, and other small items have become loose parts within reactor coolant systems. In addition to causing operational inconvenience, such loose parts can damage other components within the system or be an indication of undue wear or vibration. For such reasons, we have encouraged applicants over the past several years to participate in programs designed to develop an effective on-line loose parts monitoring system.

For the past few years we have required many applicants to initiate a program, or to participate in an ongoing program, the objective of which was the development of a functional, loose parts monitoring system within a reasonable period of time. Recently, prototype loose parts monitoring systems have been developed and are presently in operation or being installed at several plants.

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As a result of our review, we required the applicants to commit to install an appropriate available loose parts monitoring system. The applicants have made a commitment to install a loose parts monitoring system. We find this commitment acceptable.

5.5 Gross Failed-Fuel Monitor

For some time we have been requiring that nuclear power plants include a system to permit detection of any potential gross fuel failures in the core. The purpose for such a system is that it would allow for corrective action following a postulated gross fuel failure to prevent further damage to the core.

During the review of the proposed SNUPPS plants, we noted that the design did not include provisions for gross failed-fuel monitoring. Therefore, we required the applicants to commit to install a gross failed-fuel monitoring system. The applicants have committed to include a system to permit detection of potential gross fuel failures and will revise the PSAR to reflect the design modifications resulting from this commitment. We find this commitment acceptable.

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6.0 ENGINEERED SAFETY FEATURES

6.1 Summary Description

The engineered safety features for the Wolf Creek plant are described in Section 6.0 of the SNUPPS PSAR and are the same for all four SNUPPS plants.

The purpose of the various engineered safety features is to provide a complete and consistent means of assuring that the plant personnel and the public will be protected from excessive exposure to radioactive materials should a major accident occur in the plant. In this section we discuss the engineered safety feature systems proposed for the SNUPPS plants. Certain of these systems or parts of these systems will have functions for normal plant operation as well as serving as engineered safety features.

We have reviewed the proposed systems and components designated as engineered safety features. These systems and components will be designed to be capable of assuring safe shutdown of the reactor under the adverse conditions of the various postulated design basis accidents described in Section 15.0 of this report. They will be designed, therefore, as seismic Category I systems and must function even with complete loss of offsite power.

Components and systems will be provided in sufficient redundancy so that a single failure of any component or system will not result in the loss of the capability to achieve safe shutdown of the reactor. These design requirements are in accordance with the General Design Criteria.

We have reviewed the selection of materials proposed for the containment heat removal and the emergency core cooling systems, in conjunction with the expected chemistry of the cooling and containment spray system water. The applicants have shown that the use of sensitized stainless steel will be avoided, and the pH of the containment spray and the circulating coolant will be controlled. We have concluded that the proposed controls on materials and cooling water chemistry will provide assurance that the integrity of components of these systems will not be impaired by corrosion or stress corrosion.

Further, the applicants have stated that welding of austenitic stainless steel for components of these systems will be controlled to prevent deleterious hot cracking. The proposed control of weld metal composition and welding procedures are in conformance with the recommendations of Regulatory Guide 1.31 and will provide assurance that loss of function will not result from hot cracking of welds.

6.2 Containment Systems

The proposed containment system will include the containment structure and associated systems, such as the containment heat removal system, the containment isolation system, and the containment combustible gas control system and provisions for containment leakage testing.

6.2.1 Containment Functional Design

The primary containment will be a steel-lined, prestressed, post-tensioned concrete structure with a net free volume of about 2,500,000 cubic feet. The containment structure will house the nuclear steam supply system, including the reactor, steam generators, reactor coolant pumps, and pressurizer, as well as certain components of the engineered safety feature systems. The containment will be designed for an internal pressure of 60 pounds per square inch gauge and a temperature of 320 degrees Fahrenheit, and will be tested at 115 percent of design pressure.

The applicants have described the results and methods used to analyze the containment pressure response for a number of design basis loss-of-coolant accidents. Various break locations and sizes were evaluated to determine that the double-ended rupture at the pump suction of the reactor coolant system results in the highest containment pressure. Minimum containment cooling, assumed for these analyses, is provided with one containment spray train and two containment air cooling units (representing about 50 percent of the total cooling capacity), which satisfies the single active failure criterion.

The applicants have analyzed the containment pressure response from postulated loss-of-coolant accidents in the following manner. Mass and energy release rates of primary coolant from postulated breaks were first calculated for the various phases of the accident (blowdown, reflood and post-reflood), using the SATAN V, LOCTA, REFLOOD, and FROTH Westinghouse computer codes, These mass and energy release rates were then used as inputs to the Bechtel COPATTA code to calculate the containment pressure response.

Mass and energy release rates to the containment during the blowdown phase of the postulated accident were calculated by the applicants using the SATAN V computer code, which is a multinode thermal hydraulic systems program. The applicants increased the energy rate to the containment during the blowdown phase by extending the time the core remains in nucleate boiling, during which time the energy release rate from the core is highest. This energy was determined using the LOCTA computer program which takes core flow input from the SATAN V code and performs a detailed thermal hydraulic analysis for the core heat release. By using this method, the core conservatively transfers more heat to the containment for containment analysis studies. We find both the SATAN V computer code and the LOCTA computer code acceptable for calculating energy and mass release during the blowdown phase of a postulated loss-of-coolant accident.

The time delay due to lower plenum refill and subsequent decrease in mass and energy releases to the containment has not been considered by the applicants for containment analyses. The applicants have conservatively assumed that the bottom of the core is recovered immediately after the end of blowdown.

The mass and energy release rates to the containment during the reflood phase of the postulated accident (i.e., following blowdown and refill) were calculated using the REFLOOD computer code. The analysis of the reflood phase of the postulated accident is important with regard to pipe ruptures in the reactor coolant system cold legs because the steam and entrained liquid leaving the core will pass through the steam generators and will be evaporated and/or superheated to the temperature of the steam generator secondary fluid, principally in the loop with the ruptured line. Thus, additional energy beyond that released during blowdown becomes available. Results of the Full Length Emergency Cooling Heat Transfer (FLECHT) experiments indicate that the carryout fraction for fluid leaving the core during the reflood phase is about 80 percent of the incoming flow to the core. The rate of energy release to the containment during this phase becomes proportional to the flow rate into the core. The applicants assume that the 80 percent carryout fraction takes place until the core is reflooded to the 10-foot level. The rupture of the cold leg at the pump suction results in the highest mass flow through the core and therefore, through the steam generators. As a result, the containment analysis includes the effect of steam generator stored energy.

To confirm the validity of the mass and energy release to the containment during the reflood phase of the accident, we have compared the results using the REFLOOD computer program to the results using our FLOOD code. These comparisons indicate equivalent predictions of energy release. Therefore, we conclude that the REFLOOD program provides an acceptable method of computing the mass and energy release rates during the reflood phase of a postulated loss-of-coolant accident.

After the core has been completely covered with water (post-reflood phase), decay heat generation will continue to produce boiling in the core and result in a two-phase, low quality mixture of steam and water. This two-phase mixture is assumed to rise above the core and enter the steam generators due to expansion of the fluid. By this process the remainder of the available steam generator energy is removed by boiling of the water entrained in the two-phase mixture and carried into the containment as steam. In calculating the rate of energy removed from the steam generators, the applicants have used the maximum steam flow permitted by the hydraulic resistance of the system. A portion of the steam that flows through the unbroken loops through the emergency core cooling system injection points is assumed to be quenched before exiting to the containment. The applicants have performed calculations to determine the mass and energy release from the primary loop to the containment for the post-reflood phase of the accident for both the maximum and minimum engineered safety features cases. There was little difference in the containment pressure response for the two cases because for the maximum engineered safety features case with the highest safety

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injection flow, the additional available containment cooling more than compensates for the increased blowdown rate. We have reviewed the applicants' calculational methods for the post-reflood phase of the postulated loss-of-coolant accident and conclude that the resultant mass and energy release values are conservative.

The applicants have used the COPATTA code to calculate the containment pressure response to a postulated loss-of-coolant accident. The peak calculated containment pressure of 50.7 pounds per square inch gauge occurred for the cold leg, pump suction doubled-ended rupture. The analysis was based on the energy sources discussed above and partial operation of the containment spray and fan cooler systems.

We have performed confirmatory containment pressure response calculations for the design basis accident using our CONTEMPT computer code. Conservative condensing heat transfer coefficients to structures inside the containment were used in our calculations. Our results show a peak containment pressure of 51.6 pounds per square inch gauge which provides about 14 percent margin between the peak calculated pressure and the design pressure for containment.

The applicants also analyzed the containment pressure response for postulated main steam line failures inside the containment. A 1.4 square foot steam line break was considered the limiting break size, which corresponds to the size of the flow restrictor located in the nozzle of the steam generator. For this case, a maximum containment pressure of 48.2 pounds per square inch gauge was calculated for the limiting single failure, i.e., a diesel generator failure. The information was insufficient for us to complete our review. Therefore, we required the applicants to provide, and have received, additional information on containment pressure and temperature responses resulting from postulated main steam line breaks. We are currently reviewing this information and will report our evaluation of the postulated steam line break accidents in a supplement to the Safety Evaluation Report.

The applicants have provided the containment pressure response corresponding to inadvertent actuation of the containment spray system for limiting initial conditions. Limiting summer and winter conditions were selected so as to yield the largest differentials between containment ambient and spray water temperatures. The applicants calculate a peak reverse containment pressure differential of 2.72 pounds per square inch differential for the summer conditions as compared to the design external pressure load of three pounds per square inch differential. Therefore, a margin of approximately ten percent exists between the design external pressure load and the limiting case analyzed. We have performed confirmatory containment pressure response calculations for the spray actuation accident using the CONTEMPT code and find our results are in good agreement with those of the applicants.

The emergency core cooling system containment backpressure calculations were supplied for the containment design. Appendix K to 10 CFR Part 50 requires that the effect of operation of all the installed pressure reducing systems and processes be included in the emergency core cooling system evaluation. For this evaluation, it is conservative to minimize the containment pressure since this will increase the resistance to steam flow in the reactor coolant loops and reduce the reflood rate in the core.

The assumptions made by the applicants for the containment net free volume, passive heat sinks, and operation of the containment heat removal systems were conservatively selected for the emergency core cooling system analysis. These parameters were selected to minimize containment backpressure. Data for the passive heat sinks were based on the recommendations in our staff Technical Position CSB 6-1, "Minimum Containment Pressure Model for PWR ECCS Performance Evaluation," which is part of Standard Review Plan 6.2.1.5, "Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies." The passive heat sink data are based on measurements within the containments of similar nuclear plants. At the operating license stage, we will review the input data used in the analysis and compare it to the data for the as-built plant.

We conclude that the plant-dependent information used for the emergency core cooling system containment pressure analysis for the SNUPPS plants is conservative, and that the calculated containment pressure is in accordance with Appendix K to 10 CFR Part 50. This evaluation is further discussed in Section 6.3 of this report.

The applicants have analyzed the pressure response within the containment interior compartments, such as the reactor cavity and the steam generator compartments, the pressurizer vault and the pressurizer surge line compartment, during a postulated loss-of-coolant accident. The applicants have incorporated a 40 percent design margin above the peak calculated differential pressure for each compartment. We have performed confirmatory subcompartment calculations and find our results are in good agreement with those of the applicants. We conclude, therefore, that the subcompartment design is acceptable.

Subject to the satisfactory resolution of the containment pressure and temperature response to a spectrum of postulated main steam line breaks, we will be able to conclude that the containment system design and subcompartment design will be acceptable.

6.2.2 Containment Heat Removal Systems

The proposed containment spray system and the containment fan cooling system will be provided to remove heat from the containment following a postulated loss-of-coolant accident. The spray system will consist of two spray trains and the containment cooling system will consist of four fan cooling units. The systems will be designed to accommodate any single failure and still be capable of supplying 100 percent of the cooling requirements following a postulated loss-of-coolant accident to maintain a peak containment pressure below the design pressure.

The containment spray system serves only as an engineered safety feature and will perform no normal operating function. It will be designed as a seismic Category I system which will consist of redundant piping, valves, pumps and spray headers. All active components of the containment spray system will be located outside of the containment building. Missile protection will be provided by direct shielding or physical separation of equipment.

The containment spray pump recirculation intakes in the containment emergency sump will be enclosed by a screen assembly to prevent the entry of debris which could clog the spray nozzles. We find that the containment emergency sump design is consistent with the guidelines of Regulatory Guide 1.82, "Sumps for Emergency Core Cooling and Containment Spray Systems," and, therefore, is acceptable.

A high-high reactor building pressure signal from the engineered safety features actuation system will automatically actuate the containment spray system. The spray pumps will initially take suction from the refueling water storage tank. When the water in the tank reaches a low level, the spray pump suction will be manually transferred to the reactor building sump. The applicants have performed an analysis, consistent with the guidelines of Regulatory Guide 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps," to demonstrate tnat sufficient net positive suction head will be available for both the injection or recirculation modes of operation.

The containment cooling system, consisting of four equal capacity fan cooling units, will be used during both normal and accident conditions. Each of the cooling units will contain cooling coils and a fan motor assembly. Cooling water will be supplied by the essential service water system. The containment cooling system will be designed as seismic Category I. The proposed system arrangement will minimize the possibility of missile damage to the system components.

We conclude that the containment heat removal systems, will be designed to meet the requirements of Criteria 38, 39 and 40 of the General Design Criteria and Regulatory Guide 1.1, and therefore are acceptable.

6.2.3 Containment Air Purification and Cleanup Systems

The containment spray system will be used for iodine removal from the containment atmosphere following a postulated loss-of-coolant accident. Sodium hydroxide will be added to the containment spray solution to enchance the iodine scrubbing function of the system. The system will be designed to raise the spray pH to 9.5 during the injection phase of operation of the spray system. Sodium hydroxide addition will be continued during the recirculation phase until the pH of the solution in the containment sump reaches 8.5.

We calculated first order removal coefficients of 10 inverse hours for elemental iodine and 0.45 inverse hours for particulate iodine in an estimated effective containment volume of 2,000,000 cubic feet. Since the elemental iodine removal effectiveness diminishes after a decontamination factor of 100 has been achieved in the containment atmosphere, we assumed a decontamination factor limit of 100. The minimum long term sump pH of 8.5 is considered adequate to maintain the decontamination factor of 100 for the elemental iodine. We have evaluated the containment spray system and conclude it is effective for removal of elemental iodine, and iodine absorbed on airborne particulate matter.

The engineered safety feature air filtration systems proposed for the plant are the fuel building exhaust system, the control room filtration system, and the control room pressurization system. We have credited these air filtration systems with adsorption efficiencies of 95 percent for elemental and organic iodine and 99 percent for particulate iodine in our calculations of the radiological consequences of accidents. We have evaluated these systems and find that they conform with the guideline positions stated in Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants." Therefore, we conclude that they are acceptable.

Section 15.5 of this report presents our evaluation of the consequences of a postulated loss-of-coolant accident. In this evaluation, we have determined that during this postulated accident, potential leakage of sump water outside of containment into the emergency core cooling system equipment area of the auxiliary building, would result in doses at the exclusion area boundary which could exceed 10 CFR Part 100 guidelines if engineered safety feature filters, effective in iodine removal, are not used in the auxiliary building. Therefore, we require that the emergency core cooling system equipment area in the auxiliary building also be served by an engineered safety feature filter system to mitigate these consequences. We required the applicants to commit to this staff position. The applicants have committed to this position and we find the commitment acceptable.

6.2.4 Containment Isolation System

The proposed containment isolation system will be designed to automatically isolate piping systems that penetrate the containment to prevent outleakage of the containment atmosphere following a postulated loss-of-coolant accident. Double barrier protection, in the form of closed systems and isolation valves, will be provided to assure that no single active failure will result in the loss of containment integrity.

Containment isolation will automatically occur upon receipt of containment high pressure signals or reactor coolant system low pressure signals from the safety features actuation system. High radiation signals will also be used to isolate the containment vessel purge system lines.

We have reviewed the containment isolation system for conformance to Regulatory Guide 1.11, "Instrument Lines Penetrating Primary Reactor Containment," and to Criteria 54, 55, 56 and 57 of the General Design Criteria. We conclude that the system design conforms to these criteria and is acceptable.

6.2.5 Combustible Gas Control Systems

Following a postulated loss-of-coolant accident, hydrogen may accumulate inside the containment. The major sources of hydrogen generation include (1) a chemical reaction between the fuel rod cladding and the steam resulting from vaporization of emergency core cooling water, (2) corrosion of metals and paints by the spray solution, and (3) radiolytic decomposition of the cooling water in the reactor core and the containment sump.

The applicants' analysis of hydrogen generation following a postulated loss-of-coolant accident is consistent with the guidelines of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss of Coolant Accident," and indicates that the concentration in the containment would not reach the lower flammability limits of four volume percent until about 20 days after the accident.

Two 100 percent capacity electric recombiners, each with a capacity of 100 standard cubic feet per minute, will be located inside containment for post-accident hydrogen control. The proposed recombiner system will incorporate several design features that are intended to assure the capability of the system to be operable in the event of an accident. Among these are its (1) seismic Category I design, (2) protection from missiles and jet impingement and (3) redundancy to the extent that no single component failure disables both recombiners.

A hydrogen purge system will be provided in addition to the recombiners to serve as a backup to the redundant hydrogen recombiner units.

The applicants calculate that the hydrogen concentration inside containment after a postulated loss-of-coolant accident will be limited to two volume percent with operation of a single recombiner started one day after the accident. We have performed a similar analysis of hydrogen generation and hydrogen concentration in the containment following a postulated loss-of-coolant accident and our results are in agreement with the applicants' results.

A hydrogen mixing system will be included in the containment. This system will consist of four fans to maintain a uniformly mixed containment atmosphere following a postulated loss-of-coolant accident. Air will be drawn from the steam generator compartments by the locally mounted fans and will be discharged toward the upper regions of the containment. The use of this system will complement the air patterns established by the containment air coolers, which will take suction from the operating floor level and discharge to the lower regions of the containment, and the containment sprays which will cool the air and cause it to drop to lower elevations.

A redundant hydrogen monitoring system will also be included in the containment. Each hydrogen monitoring subsystem will consist of a hydrogen analyzer and two associated sample lines.

Based on our review of the systems to be provided for combustible gas control following a postulated loss-of-coolant accident, we conclude that the systems will conform to the guidelines of Regulatory Guide 1.7 and meet the requirements of Criteria 41, 42 and 43 of the General Design Criteria and are, therefore, acceptable.

6.2.6 Containment Leakage Testing Program

The proposed containment design includes the provisions and features necessary to satisfy the testing requirements of Appendix J to 10 CFR Part 50. The design of the containment penetrations, including gasketed seals and electrical penetrations, and isolation valves will permit individual periodic leakage rate testing at the pressure specified in Appendix J to 10 CFR Part 50.

The proposed reactor containment leakage testing program complies with the requirements of Appendix J to 10 CFR Part 50. Such compliance provides adequate assurance that containment leaktight integrity can be verified throughout the service lifetime of the plant and that the leakage rates will be periodically checked during service on a timely basis to maintain such leakages within the specified limits.

Maintaining containment leakage rates within such limits provides reasonable assurance that, in the event of any radioactivity releases within the containment, the loss of the containment atmosphere through leak paths will not be in excess of acceptable limits specified for the site, i.e., the resultant doses will be well within 10 CFR Part 100 limits.

We conclude that compliance with the requirements of Appendix J to 10 CFR Part 50 constitutes an acceptable basis for satisfying the requirements of Criteria 52, 53, and 54 of the General Design Criteria.

6.3 Emergency Core Cooling System

6.3.1 Design Bases

The applicants have stated that the emergency core cooling system will be designed to provide emergency core cooling during those postulated accident conditions where it is assumed that mechanical failures occur in the reactor coolant system piping resulting in loss of coolant from the reactor vessel greater than the available coolant makeup capacity using normal operating equipment. The emergency core cooling system will also be designed to protect against steam leak break consequences. The system will be similar in design, size, and capacity to that of the Comanche Peak plant, which will also be designed for a core thermal output of 3411 megawatts. The emergency core cooling system is described in Section 6.3 of RESAR 3, Consolidated Version, which the applicants have incorporated by reference.

The design bases are to prevent fuel and cladding damage that would interfere with adequate emergency core cooling and to mitigate the amount of clad-water reaction for any size break up to and including a double-ended rupture of the largest primary coolant line. The applicants have stated that these requirements will be met even with minimum engineered safeguards available, such as the loss of one emergency power bus together with the unavailability of offsite power.

The emergency core cooling system to be provided will have the required number, diversity, reliability, and redundancy such that no single failure of the emergency core cooling system equipment, occurring during a loss-of-coolant accident, will result in inadequate cooling of the reactor core. Each of the proposed subsystems will be designed to function over a specific range of reactor coolant piping system break sizes, up to and including the flow area associated with a postulated double-ended break in the largest reactor coolant pipe (9.14 square feet is the double-ended area).

The reactor coolant system and the above described emergency core cooling subsystems and components as proposed are similar to those reviewed and accepted for the Comanche Peak plant.

6.3.2 System Design

The proposed emergency core cooling system will consist of four accumulator tanks, two high pressure injection systems and a low pressure injection system, with provisions for recirculation of the borated coolant after the end of the injection phase. Various combinations of these systems will assure core cooling for the complete range of postulated break sizes.

Following a postulated loss-of-coolant accident, the emergency core cooling system will operate initially in the active high pressure injection mode and the passive accumulator injection mode, then in the active low pressure injection mode, and subsequently in the recirculation mode.

The high pressure injection mode of operation, upon actuation of a safety injection signal, will consist of the operation of two centrifugal charging pumps (rated at 150 gallons per minute each at a design head of 5800 feet) which provide high pressure injection of boric acid solution (via the boron injection tank maintained at 21,000 parts per million boron concentration) into the reactor coolant system. Also designed to operate during the high pressure injection mode are two safety injection pumps (rated at 425 gallons per minute each at a design head of 2500 feet), which will take their suction from the refueling water storage tank in which the borated water will have a boron concentration of 2000 parts per million.

Low pressure injection will be provided by two residual heat removal pumps (rated at 3000 gallons per minute each at a design head of 375 feet) which will take their suction from the refueling water storage tank.

When a predetermined amount of water in the refueling water storage tank has been injected or on actuation of the low-level alarm from the refueling water storage tank, suction will be transferred manually to the containment sump for the recirculation mode of operation. Then the emergency core cooling system will provide the long-term core cooling requirements by recirculating the spilled reactor coolant (from the ruptured pipe) collected in the containment sump, back to the reactor vessel through the reactor coolant cold legs.

Each of the four accumulator tanks will have a total volume of 1350 cubic feet with a minimum borated water volume of 850 cubic feet and 500 cubic feet of nitrogen gas at a normal operating pressure of 660 pounds per square inch gauge. Each tank will be connected to one of the cold legs of the reactor coolant system by a line with two check valves in series. A normally open gate valve will also be located in the lines between each accumulator and the cold leg piping. These valves will be provided with appropriate interlocks to assure that the valves will be open during power operation when availability of the accumulators is required.

6.3.3 Performance Evaluation

The emergency core cooling system has been designed to deliver fluid to the reactor coolant system in order to control the predicted cladding temperature transient following a postulated pipe break and for removing decay heat in the long-term, recirculation mode.

On June 29, 1971, the Commission issued an Interim Policy Statement containing Interim Acceptance Criteria for the performance of the emergency core cooling systems for light-water cooled nuclear power reactors. The Interim Policy Statement includes a set of conservative assumptions and procedures to be used in conjunction with computer codes to analyze and evaluate the emergency core cooling system function for a pressurized water reactor incorporating a dry containment. A public rulemaking hearing on the Interim Acceptance Criteria for the emergency core cooling systems for light-water cooled nuclear power reactors has been conducted.

On January 4, 1974, the Commission published its decision in the rulemaking proceeding (Docket No. RM-50-1) concerning acceptance criteria for emergency core cooling systems for light-water cooled nuclear power reactors. This decision included the new amendments of 10 CFR Part 50 which incorporates the ruling. The new subparagraphs (a)(4) of paragraph 50.34 and (a)(1) of paragraph 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors," of 10 CFR Part 50, state in part:

50.34(a)(4), "Analysis and evaluation of ECCS cooling performance following postulated loss-of-coolant accidents shall be performed in accordance with the requirements of §50.46 for facilities for which construction permits may be issued after December 28, 1974."

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50.46(a)(1), "...each boiling and pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical Zircaloy cladding shall be provided with an emergency core cooling system (ECCS) which shall be designed such that its calculated cooling performance following postulated loss-of-coolant accidents conforms to the criteria set forth in paragraph (b) of this section. ECCS cooling performance shall be calculated in accordance with an acceptable evaluation model, and shall be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the entire spectrum of postulated loss-of-coolant accidents is covered. Appendix K, ECCS Evaluation Models, sets forth certain required and acceptable features of evaluation models."

These latest provisions are applicable to the SNUPPS plants since the issuance of construction permits can occur only after December 28, 1974. The applicants have committed to provide an analysis of a spectrum of breaks satisfying the requirements of these criteria. The applicants have already submitted a partial analysis. This is under review at the present time. The applicants are scheduled to submit additional information in September 1975 to complete their response on the emergency core cooling system analysis to demonstrate compliance with the requirements of Appendix K to 10 CFR Part 50. We will review all the analyses and the results of our evaluation will be reported in a supplement to this Safety Evaluation Report.

An evaluation of the emergency core cooling system in accordance with the requirements of the Interim Acceptance Criteria had been previously performed in our evaluations of other plants, such as Comanche Peak, which have incorporated the emergency core cooling system design described in Section 6.3 of RESAR 3, Consolidated Version. These evaluations demonstrated system compliance with the Interim Acceptance Criteria.

6.3.4 Tests and Inspections

The applicants will demonstrate the operability of the emergency core cooling system by subjecting all components to preoperational tests, periodic testing, and in-service testing and inspections.

The preoperational tests to be performed fall into three categories:

(1) System actuation tests to verify (a) the operability of all emergency core cooling system valves initiated by the safety injection signal, the phase A containment isolation signal, and the phase B containment isolation signal; and (b) the operability of all safeguard pump circuitry down through the pump breaker control circuits and the proper operation of all valve interlocks.

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(2) Accumulator injection tests to check the accumulator system and injection line to verify that the lines are free of obstructions and that the accumulator check valves and isolation valves operate correctly. The applicants will perform a low pressure blowdown of each accumulator to meet the test objectives.

(3) Operational test of all the major pumps. These pumps consist of the charging pumps, the residual heat removal pumps, and the safety injection pumps. The applicants will use the results of these tests to evaluate the hydraulic and mechanical performance of these pumps delivering through the flow paths for emergency core cooling. These pumps will operate under both miniflow (through test lines) and full flow (through the actual piping) conditions.

By measuring the flow in each pipe, the applicants will make the adjustments necessary to assure that no one branch has an unacceptable low or high resistance. System checks will be made to ascertain that total line resistances are sufficient to prevent excessive runout of the pump. The applicants will be required to show during preoperational tests that the minimum acceptable flows, as determined for the Final Safety Analysis Report analysis, are met by the measured total pump flow and relative flow between the branch lines.

The systems will be accepted only after demonstration of proper actuation of all components and after demonstration of flow delivery of all components within design requirements.

The applicants will perform routine periodic testing of the emergency core cooling system components and all necessary support systems with the plant at power. Valves that are required to operate after a loss-of-coolant accident will be operated through a complete cycle and pumps will be operated individually in these tests on their miniflow lines except for the charging pumps which will be tested by their normal charging function.

The applicants will use test circuits to periodically check for leakage of reactor coolant back through the accumulator discharge line check valves to ascertain that these valves seat whenever the reactor coolant system pressure is above a pre-set value. The periodic testing of the emergency core cooling system will also include a visual inspection of pump seals, valve packings, flanged connections, and relief valves to detect leakage.

The applicants have stated that the emergency core cooling system components will be designed and fabricated to permit inspection and in-service tests in accordance with Section XI of the ASME Code.

6.3.5 Conclusions

As stated in Section 6.3.3 of this report, the applicants will be required to demonstrate compliance with the emergency core cooling system criteria published in the

Federal Register on January 4, 1974. We are continuing our review of the applicants' analysis of the emergency core cooling system, which will include additional information to be supplied by the applicants, and our evaluation will be included in a supplement to this Safety Evaluation Report. System compliance with the Interim Acceptance Criteria was demonstrated in our evaluation of other plants, such as Comanche Peak, which will use the emergency core cooling system design described in RESAR-3, Consolidated Version.

6.4 Control Room Habitability Systems

The emergency protective provisions of the control room related to the accidental release of radioactivity and toxic gases are evaluated in this section. While relevant portions of the control room ventilation system are described here, a more complete description and evaluation of the control room ventilation system is given in Section 9.5.1 of this report.

The applicants propose to meet Criterion 19 of the General Design Criteria by use of concrete shielding and by the installation of redundant charcoal filter trains. A pressurization charcoal filter of 2000 cubic feet per minute capacity will be used to filter control building pressurization air. A portion of this air (400 cubic feet per minute) will be mixed with 1600 cubic feet per minute of control room return air, filtered through a 2000 cubic feet per minute charcoal filter and supplied to the control room. The 400 cubic feet per minute of pressurized air will be supplied to prevent infiltration of unprocessed air into the control room. This emergency mode of ventilation will be automatically activated upon the receipt of a high radiation signal from the radiation detectors in the fresh air inlets, receipt of a safety injection signal, receipt of a spent fuel pool high-radiation signal, or by detection of chlorine by the remote chlorine sensors or the chlorine detectors to be located in the control building fresh air inlet duct.

We have calculated operator doses assuming a postulated loss-of-coolant accident for the ventilation system as described above. The resultant doses were found to meet the guidelines of Criterion 19 of the General Design Criteria. Therefore, in terms of mitigating the radiological consequences in the control room due to a postulated loss-of-coolant accident, we find the system acceptable.

Control room habitability, following a postulated toxic gas release, is required to assure that operators can continue to operate the plant. Chlorine has been identified as the only material that, if released, would pose a potential operator hazard. Provisions, such as quick-acting chlorine detectors, charcoal filters, and self-contained breathing apparatus, will be provided to protect the operators against a chlorine release. We have reviewed these provisions with respect to the guidelines of Regulatory Guides 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," and 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine

Release," and had found them to be acceptable with the exception of breathing apparatus capacity. We require that the control room area be supplied with at least five self-contained breathing devices having a total of 30 man-hours of breathing air, as recommended by Items 4c and 4d of Regulatory Guide 1.95. Therefore, we required the applicants to either commit to this staff position or justify other acceptable means of satisfying the objectives of the guide. The applicants have committed to the above staff position.

Based on this commitment, we now conclude that the proposed design of the control room habitability systems is acceptable.

6.5 Auxiliary Feedwater System

The auxiliary feedwater system will be used to maintain water level in the steam generators during periods when the main feedwater system is not in operation and the reactor coolant temperature is greater than 350 degrees Fahrenheit. The auxiliary feedwater system will provide feedwater for the removal of reactor core decay heat when feedwater is not available from the condensate and main feedwater system.

The auxiliary feedwater system will not be required to operate during normal power operation. It will be on standby to deliver auxiliary feedwater flow during normal cooldown or emergency conditions.

The auxiliary feedwater system will consist of two motor-driven pumps, one steam turbine-driven pump, associated piping, valves, instruments and controls. Normal auxiliary feedwater flow will be from the nonsafety-related condensate storage tank through a motor-operated isolation valve and a nonreturn valve, to the auxiliary feedwater pumps. Two backup sources of water, one from each redundant and separate safety-related essential service water header, will be provided for the turbine-driven pump. A separate backup source of water will also be provided to each motor-driven pump from the safety-related essential service water system.

The condensate storage tank will have sufficient capacity to maintain the plant at hot shutdown conditions for 24 hours, and then to cool down the primary system at an average rate of 50 degrees Fahrenheit per hour to a temperature of 350 degrees Fahrenheit, at which point the residual heat removal system can operate.

Each of the two motor-driven pumps, and each of the two headers from the turbine-driven pump discharge, will be able to feed two steam generators through their individual control valve stations. Power to each pump motor will be supplied from a physically separated Class IE bus. The turbine-driven pump will be supplied by steam from two main steam headers. Controls and valve operators on each auxiliary feedwater train, which will be supplied by either a motor-driven pump or by the turbine-driven pump, will be fed from an independent Class IE power system. The power to each train will have a diverse power source. Auxiliary feedwater control will normally be done

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at the control room, but instrumentation will be provided for local operation in the unlikely event that evacuation of the control room is required.

The auxiliary feedwater system, including the two separate sources from the essential service water system, will be designed as seismic Category I. The auxiliary feedwater pumps will be located in separate watertight compartments to protect them from flooding or steam release following equipment failure. Either of the two motor-driven pumps will supply 100 percent of the feedwater flow required for decay heat removal. The turbine-driven auxiliary feedwater pump will supply 200 percent of the feedwater flow required for decay heat removal and its discharge can be directed to either, or both, motor-driven pump discharge headers.

We have reviewed the design criteria and bases, and the single-failure analysis provided by the applicants, and conclude that the auxiliary feedwater system is acceptable.

7.0 INSTRUMENTATION AND CONTROLS

7.1 General

The proposed design for the instrumentation and control systems for the Wolf Creek plant is described in Section 7.0 of the SNUPPS PSAR, which incorporates RESAR-3, Consolidated Version. The instrumentation and control systems design for the Wolf Creek plant is identical to that for the other SNUPPS units.

The proposed design for the protection and control systems for the Wolf Creek facility was reviewed utilizing the Commission's General Design Criteria, applicable Institute of Electrical and Electronic Engineers (IEEE) Standards, including IEEE Standard 279-1971, ""Criteria for Protection Systems for Nuclear Power Generating Stations," and applicable Regulatory Guides for light water reactors.

The design of those portions of the protection and control systems of the Wolf Creek plant which fall within the scope of the nuclear steam supply system contractor is described in RESAR-3, Consolidated Version. Our review of the protection and control systems concentrated on those aspects of the design which are different than those described in RESAR-3, Consolidated Version. Earlier editions of RESAR-3 were evaluated during our review of other plants using the Westinghouse nuclear steam supply system. The results of these reviews were factored into RESAR-3, Consolidated Version. Our review of the Wolf Creek facility consisted primarily of evaluation of the exceptions to RESAR-3, Consolidated Version, which were identified by the applicants.

The balance-of-plant portions of the protection and control systems, which do not fall within the scope of the nuclear steam supply system, are described in the SNUPPS PSAR. The architect-engineer services for the portion of the balance-of-plant which is described in the SNUPPS PSAR will be provided by the Bechtel Power Corporation. The review of the portions of the protection and control systems which fall within the Bechtel scope of supply was accomplished by comparing the design proposed for the SNUPPS units with the approved designs of the Trojan plant and Millstone, Unit 3.

We note that incidents involving the inadvertent disabling of a component by racking out the circuit breaker for a different component have occurred in operating nuclear power plants. It has been determined that in such cases the racked-out position of breakers had not been included in the failure mode analysis of those control circuits. In response to our concern, the applicants have revised the PSAR to include the criterion that in the standardized (SNUPPS) portion of the design there are no interlocks, sequencing controls, or common control elements between redundant systems. The applicants have committed to perform a failure mode and effects analysis to verify the independence of the protection systems and will submit this analysis in the Final Safety Analysis Report. We find this commitment acceptable.

The applicants have stated that certain reactor trip and engineered safety features system channel response times, which are considered to be significant in the analysis of postulated accidents, will be verified during plant preoperational testing prior to initial criticality. They will thereafter be tested periodically. We find the proposed test program acceptable and will review the details of the response time test program at the operating license stage of the review.

7.2 Reactor Trip System

The applicants have stated that the reactor trip system will be identical to that presented in RESAR-3, Consolidated Version. The system will utilize the Westinghouse solid state logic design described in WCAP-7488-L, "Solid State Logic Protection System" and first introduced in the now operational D. C. Cook facility. This system automatically initiates a reactor trip whenever any monitored variable or combination of monitored variables exceeds its safe operating range, as defined by the reactor trip setpoints stated in the technical specifications.

The following is a list of reactor trips provided:

(1) Power range high neutron flux

(2) Intermediate range high neutron flux

(3) Source range high neutron flux

(4) Power range high positive neutron flux rate

(5) Power range high negative neutron flux rate

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(6) Core overtemperature ΔT

(7) Core overpower ΔT

(8) High pressurizer pressure

(9) Low pressurizer pressure

(10) High pressurizer water level

(11) Low reactor coolant flow

(12) Reactor coolant pump breaker open

(13) Reactor coolant pump bus undervoltage

(14) Reactor coolant pump bus underfrequency

(15) Low feedwater flow

(16) Low-low steam generator water level

(17) Turbine trip

(18) Safety injection signal actuation

(19) Manual

The reactor trip system will be periodically tested for proper operation in accordance with the recommendations of Regulatory Guide 1.22, "Periodic Testing of Protection System Actuation Functions."

This system is essentially the same as those systems previously reviewed and approved for similar plants such as Comanche Peak, Units 1 and 2 and Millstone, Unit 3. On this basis, we have concluded that the design of the reactor trip system meets the requirements of the applicable criteria, standards, guides and positions adopted by the Commission and is acceptable with the following notations and exceptions:

(1) Our requirements with respect to anticipated transients without scram are provided in WASH-1270, "Anticipated Transients Without Scram for Water-Cooled Power Reactors." The applicants have stated that they believe the Wolf Creek design already satisfied the requirements of WASH-1270 and that no hardware modifications are required to mitigate the consequences of the anticipated transients without scram. The applicants reference Westinghouse report, WCAP-8330, "Westinghouse Anticipated Transient Without Trip Analysis," as the basis for this conclusion. Report WCAP-8330 is currently under generic review by the staff. We will require that any design changes that are required as a result of our review, when it is completed, be implemented in the design of the SNUPPS plants.

(2) The design of the anticipatory trips portion of the reactor trip system was not acceptable since no clear commitment of conformance with the requirements of IEEE Standard 279-1971 was provided, particularly with regard to qualification and testability of the sensors. These anticipatory trips are (a) turbine autostop oil pressure low, (b) turbine stop valves close, (c) undervoltage on reactor coolant pump power supply buses, (d) underfrequency on reactor coolant pump power supply buses, and (e) reactor coolant pump circuit breakers open. Our position is that there are no other classes of trip circuits, thus all input to

the reactor trip system must be Class IE, seismically and environmentally qualified and conform to the requirements of IEEE Standard 279-1971.

In response to our concern, the applicants have fully committed to meet this requirement for trip items (c) and (d). The applicants have also committed to meet this requirement for trip items (a) and (b) with an exception to the seismic criteria regarding mounting and location for that portion of the trip system located within non-seismic Category I structures. We consider that a design which meets all of the requirements for Class IE circuits is acceptable and that allowing the above exception is an acceptable method of meeting our requirement.

For trip item (e), the applicants have deleted the reactor coolant pump circuit breakers open as an anticipatory trip and reference Westinghouse Topical Report WCAP-8424, Revision 1, "An Evaluation of Flow Accidents Caused by Power System Frequency Transients in Westinghouse PWRs", for justification. We are currently reviewing this topical report on a generic basis. In the event that the generic resolution of WCAP-8424 does not result in the deletion of this trip, the applicants have committed to qualify the circuits for the reactor coolant pump circuit breakers in the same manner as trip items (a) and (b) above. We will evaluate the applicants' requirements for trip item (e) at the conclusion of our generic review WCAP-8424, if required.

(3) A number of concerns regarding physical separation and isolation of safety and non-safety circuits within the solid state logic and the process analog racks of the protection system have been identified. These include (a) qualification of the photodiode isolators in the solid state logic racks, (b) inadequate physical separation between the input and output wiring of the isolation board of the solid state logic racks, and (c) inadequate physical separation between the protection and control system. These concerns are being resolved generically with Westinghouse. We will verify the resolution of these concerns during the operating license stage of our review.

7.3 Engineered Safety Features Initiation and Actuation Systems

The designs for the instrumentation systems that are to be used to initiate and control the operation of the engineered safety features systems are described in RESAR-3, Consolidated Version. The Bechtel Power Corporation is responsible for the design of the containment including the various containment systems that will be engineered safety features or supporting systems. The instrumentation associated with these systems have been described by the applicants, including interfacing with the Westinghouse-supplied instrumentation. The applicants have stated that the periodic testing of the engineered safety features systems has been described in RESAR-3, Consolidated Version and will conform to the recommendations of Regulatory Guide 1.22.

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The engineered safety features actuation system is composed of an analog portion consisting of three to four redundant channels per plant parameter monitored and a digital portion consisting of two redundant logic trains which receive inputs from the analog protection channels.

Functions initiated by the engineered safety features actuation system include:

(1) Reactor trip (if a trip has not already been initiated by the reactor trip system).

(2) Phase A containment isolation (whose function is to prevent fission product release).

(3) Steam line isolation (to prevent the continuous, uncontrolled blowdown of more than one steam generator).

(4) Start of emergency diesels.

(5) Isolation of control room intake ducts.

(6) Containment spray actuation.

On the basis of our review of this and similar plants, we conclude that the design of the initiation and actuation systems for the SNUPPS design is acceptable with the following notations and exceptions:

(1) The test program in RESAR-3, Consolidated Version references Westinghouse Topical Report WCAP-7705, "Engineered Safeguards Final Device Actuator Testing." The applicants had committed to the generic resolution of this report. However, WCAP-7705 has recently been found unacceptable by the staff, and both Westinghouse and the staff have indicated that early resolution of this matter on a generic basis does not seem possible. We have requested the applicants to address, for their application, their resolution of the deficiencies of WCAP-7705 identified in our letter to Westinghouse, dated July 14, 1975. We will review the applicants' proposed resolution and report our evaluation in a supplement to the Safety Evaluation Report.

(2) The auxiliary feedwater system is comprised of two subsystems, one that utilizes a steam-driven pump and Class IE direct-current power for controls, and one which utilizes Class IE alternating-current power for two redundant motor driven pumps and controls. Either subsystem can provide required flow.

We have reviewed the design of the auxiliary feedwater system and find that the design provides sufficient redundancy and diversity such that there is not complete reliance upon any one source of energy. This diversity includes not only the pump drives, but all instrumentation and controls, control circuitry, and motive power to all valve operators that are required for operation of the system. The electrical power, instrumentation and controls for the auxiliary feedwater system satisfy the requirements set forth in IEEE Standard 279-1971 and IEEE Standard 308-1971. We conclude that this design is acceptable.

(3) Regarding the application of the single failure criterion to manually-controlled electrically-operated valves, the applicants state that, "All safety-related systems having electrically-operated, manually controlled valves within the Bechtel scope of supply are designed to meet the single electrical component failure criterion. This is accomplished by the use of redundant valves." We conclude that this design is acceptable for the Bechtel scope of supply.

For the Westinghouse scope of supply, the applicants reference RESAR-3, Consolidated Version. We had reviewed this design and concluded that it is not acceptable. We required that the design be revised to conform fully to the guidance of our staff position, attached as Appendix C. In response to our position, the applicants have submitted their proposed resolution for meeting the single failure criterion by locking out power to the controller for these valves. We are currently reviewing the proposed design change and will report our evaluation in a supplement to the Safety Evaluation Report.

(4) The instrumentation and controls for the actuation of the containment spray system are redundant and immune to single failures. Since inadvertent actuation of the containment spray would be operationally undesirable, the design also provides that no single failure or single operator action could inadvertently initiate the containment spray system. We have reviewed the electrical power, controls and instrumentation for the containment spray system and find the design to be acceptable.

7.4 Systems Required for Safe Shutdown

Our review of the systems being provided for safe shutdown has shown that these systems are identical to those described in RESAR-3, Consolidated Version, and are therefore acceptable.

The applicants have identified the systems and equipment and their locations that are available for proceeding to hot shutdown from outside the control room, as required by Criterion 19 of the General Design Criteria. The applicants have also committed to providing procedures for attaining cold shutdown from outside the control room in compliance with Criterion 19. We find this commitment to be acceptable.

7.5 Safety-Related Display Instrumentation

7.5.1 Westinghouse Scope of Supply

The design criteria for the safety-related display instrumentation which fall within the Westinghouse scope of supply are presented in RESAR-3, Consolidated Version. We interpret these criteria to meet our position.

The safety-related display instrumentation for post-accident monitoring and safe shutdown should be:

(1) Redundant, with indicators in the control room for both channels, and with at least one channel recorded.

(2) Energized from the onsite emergency power supplies.

(3) Designed in accordance with the requirements of IEEE Standard 279-1971.

(4) Qualified in accordance with the requirements of IEEE Standard 323-1974 and IEEE Standard 344-1971, as supplemented by applicable staff positions, with the exception that the recorders are not required to function within their required accuracy during the safe shutdown earthquake, but must function within their required accuracy immediately after the ground motion subsides without requiring any maintenance.

On the basis of the above interpretation, we conclude that the design of the safety-related display instrumentation is acceptable for the Westinghouse scope of supply.

7.5.2 Bechtel Scope of Supply

The design criteria for the safety-related display instrumentation for the balance of plant are described in Section 7.5 of the SNUPPS PSAR and in the responses to our requests for additional information. We find the design criteria acceptable.

The applicants had indicated an exception to the requirements of Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems," for portions of the systems required for safe shutdown (See PSAR Section 7.5.3.3.1). It is our position that systems required for safe shutdown must be included with those systems "that perform a function important to safety." (See paragraph 1 of part B of Regulatory Guide 1.47). We require that the display instrumentation be designed to provide conformance with the guidance of Regulatory Guide 1.47, as supplemented by our staff position, attached as Appendix D, for systems required for safe shutdown. The applicants have provided clarifying information which demonstrates conformance with our position.

Therefore, we conclude that the design is acceptable.

7.6 Other Systems Required for Safety

These systems are essentially as presented in RESAR-3, Consolidated Version. We have reviewed these systems and find them acceptable with the following notations and exceptions:

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(1) The turbine overspeed protection system consists of two independent, redundant means of tripping the turbine before design overspeed is approached. The primary trip device is the mechanical-hydraulic overspeed trip with redundancy being provided by the electric-hydraulic backup overspeed trip. The electric-hydraulic backup overspeed trip is generally set to operate at a speed of 1.5 percent greater than the mechanical-hydraulic overspeed trip.

The trip functions can be independently tested during all modes of operation. Lock-out valves are provided to test the mechanical or electrical trip devices while the machine is operating with the opposite function offering protection while testing. Hydraulics can be tested by testing individual turbine valves.

We have concluded that the design meets the redundancy, diversity and testability requirements for the turbine overspeed protection system and is, therefore, acceptable.

(2) In Section 15.3.4 of RESAR-3, Consolidated Version (Complete Loss of Forced Reactor Coolant Flow) credit is taken for reactor coolant pump trip on underfrequency or fault conditions on the pump power source to assure the pump's kinetic energy is available for flow coastdown. It is our position that if such credit is taken, the pump breakers must be qualified in accordance with the requirements of IEEE Standard 279-1971 and IEEE Standard 308-1971. Further, they must be located in a seismic Category I structure. It has been tentatively established that unless it can be demonstrated by analysis that an underfrequency rate of 15 hertz per second will not prevent the pumps from performing their coastdown function, the tripping of the reactor coolant pump breakers will be considered a required safety action.

We are currently reviewing Westinghouse Topical Report WCAP-8424, Revision 1, to resolve this matter. The applicants have committed to the generic resolution of this matter between Westinghouse and the staff. We find this commitment acceptable.

(3) We have reviewed the design of the combustible gas sampling system. We find that it conforms to the requirements of Criteria 13 and 41 of the General Design Criteria and the guidelines of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss of Coolant Accident." We find that the system meets the redundancy, power-source, and instrumentation requirements for an engineered safety feature system. The applicants state that the system will be constructed as seismic Category 1. We conclude that the design is acceptable.

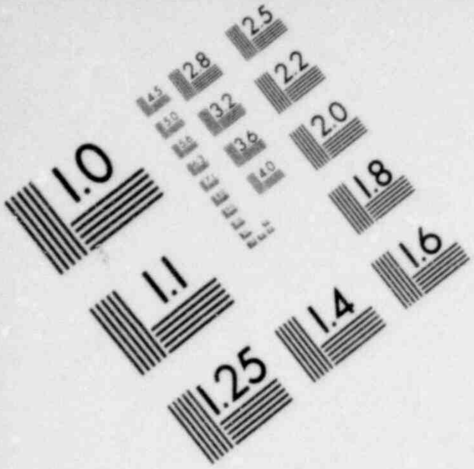
(4) We have reviewed the design and the applicants' program for testing of the containment electrical penetrations and conclude that the requirements of IEEE Standard 336-1971 and IEEE Standard 317-1971, as supplemented by Regulatory

IMAGE EVALUATION
TEST TARGET (MT-3)

6″

IMAGE EVALUATION
TEST TARGET (MT-3)

6"



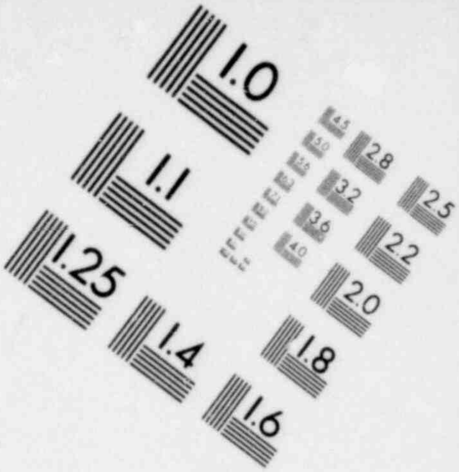
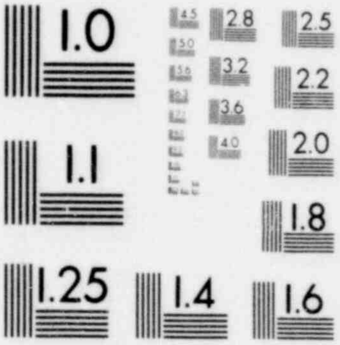
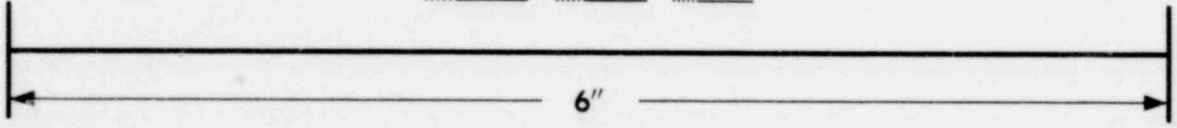


IMAGE EVALUATION TEST TARGET (MT-3)





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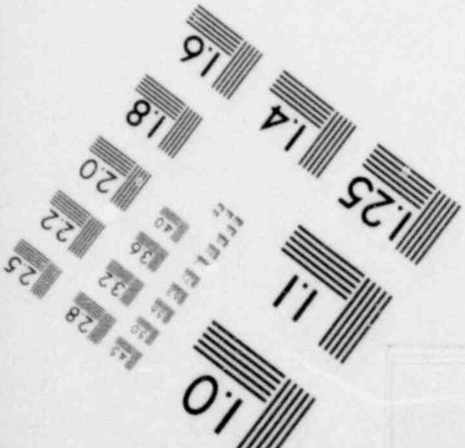

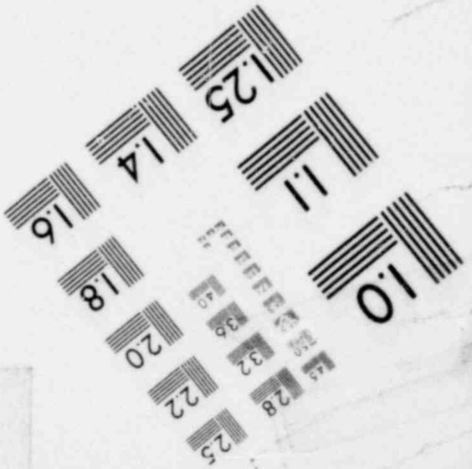

Guide 1.63, "Electric Penetration Assemblies in Containment Structures for Water-Cooled Nuclear Power Plants," are satisfied. Therefore, we conclude that the design and testing program for containment electrical penetrations are acceptable.

7.7 Control Systems Not Required For Safety

The design of the control systems not required for safety is as presented in RESAR-3, Consolidated Version with these notations:

(1) The facility design incorporates the option of a capability for 50 percent net load rejection without inducing reactor trip. We find this to be acceptable.

(2) The control room arrangement for the plant is different from that shown in RESAR-3, Consolidated Version. We conclude that this does not degrade plant safety.

Therefore, we conclude that the design of the control systems not required for safety is acceptable.

7.8 Process Instrumentation (7300 Series)

The applicants have identified the Westinghouse Electric Corporation 7300 series process instrumentation as being part of the SNUPPS instrumentation system design. This equipment performs the same functions as the earlier 7100 series equipment and, therefore, is acceptable as a preliminary design basis.

The 7300 series equipment is to be used in several plants (e.g., North Anna, Units 1 and 2, Docket Nos. 50-338 and 50-339) for which operating license reviews will have been completed prior to the end of construction of the first SNUPPS plant. Hence, we believe that any problem areas which may arise with the 7300 series equipment will be identified and resolved on these other dockets (or directly with Westinghouse on a generic basis) prior to the submittal of the Final Safety Analysis Reports for the SNUPPS plants. We will require that the generic resolution of any problem areas be implemented by the applicants for the SNUPPS design. This will be verified during the operating license stage of review.

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8.0 ELECTRIC POWER SYSTEMS

8.1 Introduction

The criteria for the proposed electric power systems and the onsite power systems for the standard portion of the Wolf Creek plant are described in Section 8.0 of the SNUPPS PSAR. The grid description, the offsite power systems and the few remaining onsite power systems outside the scope of the standard plant design are described in Section 8.0 of the Wolf Creek Site Addendum Report.

The following served as the bases for performing the evaluation of the electric power systems:

(1) Criteria 17 and 18 of the General Design Criteria.

(2) Regulatory Guides 1.6, 1.9, 1.32, 1.41, 1.75, 1.81 and 1.93.

(3) IEEE Standard 308-1971 "Standard Criteria for Class IE Electric Systems for Nuclear Power Generating Stations."

(4) Applicable Staff Positions.

8.2 Offsite Power Systems

Offsite alternating-current power for the Wolf Creek site will be provided by (1) a 345-kilovolt network connected to the site by means of four transmission lines and (2) the Kansas Gas & Electric Company's 69-kilovolt transmission system connected to the site by a separate transmission line. This provides two full-capacity, immediate-access, physically-independent offsite sources of alternating-current power for station startup and shutdown.

Each of the immediate access sources will supply one of the two Class IE buses through an assigned engineered safety feature transformer. The 13.8-kilovolt side of the 69/13.8-kilovolt transformer in the 69-kilovolt switchyard will be connected to the No. 1 engineered safety feature transformer by means of an underground circuit. The other engineered safety feature transformer will be connected to 13.8-kilovolt windings of the 345/13.8-kilovolt startup transformer which is part of the power block of the standard nuclear unit. The startup transformer will be connected to the 345-kilovolt switchyard by an overhead line to be supported on its own individual structures. A second overhead 345-kilovolt circuit will connect the main transformers for the nuclear generating unit to the 345-kilovolt switchyard. These two overhead transmission

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lines will be supported on their own individual structures. Structural design and circuit separation are such as to eliminate the possibility of a structural collapse causing an outage of both 345-kilovolt transmission circuits.

The 345-kilovolt switchyard design will include a breaker-and-a-half arrangement for each circuit, along with breaker failure backup protection. Each 345-kilovolt breaker will have two trip coils on separate isolated direct-current control circuits. The 345-kilovolt and 69-kilovolt circuit breakers can be tested, inspected and maintained without removing the generators, transformers or transmission lines from service.

The applicants have conducted a load-flow and transient stability analysis to demonstrate the following for the offsite power system:

(1) The system can successfully withstand loss of the Wolf Creek plant when fully loaded.

(2) With all 345-kilovolt lines in service and the Wolf Creek plant fully loaded, the system can successfully withstand the loss of any one 345-kilovolt line from the Wolf Creek substation under three-phase fault conditions with the fault cleared in normal clearing time.

(3) With all 345-kilovolt lines in service and the Wolf Creek plant fully loaded, the system can successfully withstand the loss of any two elements caused by a single-phase fault being cleared by back-up breaker operation in back-up clearing time.

(4) Any one 345-kilovolt line, when energized from the remote end, can successfully carry the total engineered safety feature load should it become necessary to do so.

(5) The 69-kilovolt line from the Athens substation to the Wolf Creek site can successfully carry the total engineered safety feature load should it become necessary to do so.

(6) All of the above comments apply on both a transient stability and a steady state basis.

We have reviewed the applicants' plans for testing the offsite power system and find that they meet the requirements of Criterion 18 of the General Design Criteria. We conclude that the design of the offsite emergency power system meets the requirements of Criteria 17 and 18 of the General Design Criteria and is, therefore, acceptable.

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8.3 Onsite Power Systems

The onsite power systems are comprised of the standardized (SNUPPS) portion and the site-related portion. The standardized portion will include the electrical systems relating to the nuclear steam supply system, the turbine generator, the main, unit auxiliary, startup and engineered safety features transformers, and the auxiliary equipment associated with these items. The site-related portion, which is the nonstandardized portion, will include the switchyard and electrical systems relating to the intake structures, water and sewage treatment facilities, and miscellaneous station buildings such as the office, warehouses, and guardhouse. Controls for safety-related parts of the site-related portion will be located in the control room.

8.3.1 Alternating-Current Power System

The onsite alternating-current power system will include a Class IE electric system. The Class IE electric system will include two independent alternating-current load groups, each of which has its own buses, transformers, loads, and associated 125-volt direct-current control power. Either load group will be independently capable of safely shutting down the unit.

Two physically and electrically independent engineered safety features transformers will be provided to supply the Class IE alternating-current electric power system. One independent diesel generator will be provided for each Class IE alternating-current load group. For each protection and control channel, one independent 125-volt direct-current and 115-volt vital alternating-current power source will be provided.

The Class IE system will be designed to adequately perform its safety function in the event of a failure of a single component of the system.

We conclude that the applicants' design of the engineered safety features alternating-current power distribution system is adequate to assure that no single failure will cause the loss of more than one engineered safety feature division. We conclude, therefore, that the design is acceptable.

The onsite alternating-current standby power will be supplied by two diesel generators, either of which is capable of supplying essential loads necessary to reliably and safely shutdown and isolate the reactor. The diesel generators are conservatively rated for continuous operation consistent with essential load requirements. Each diesel generator will be connected exclusively to a 4.16-kilovolt safety features bus of a load group. The plant will have two load groups and the safety-related equipment on both load groups is similar. The load groups are redundant, and one load group will be adequate to satisfy the minimum engineered safety features demand caused by a loss-of-coolant accident and the loss of preferred power supply. The diesel generators will be electrically isolated from each other. There is no provision for automatic

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switching of redundant buses or loads. Further, interlocks will be provided to prevent redundant buses from being paralleled. The starting and operation of any diesel is not conditioned by operation of any other.

Physical separation for fire and missile protection will be provided between the diesel generators as they are housed in a Category I structure with a Category I wall between them. Power and control cables for the diesel generators and associated switchgear will be routed to maintain physical separation. The diesel generators and support equipment necessary for operation will comply with seismic Category I requirements.

The ratings for the diesel generator sets satisfy the requirements set forth in Regulatory Guide 1.9, "Selection of Diesel Generator Set Capacity for Standby Power Supplies."

In the event that the diesel generator sets are of a type or size not previously used or qualified as standby emergency power sources, the applicants have committed to qualify them in accordance with our requirements. We conclude that this commitment is acceptable.

Each standby diesel generator will have a two-hour supply of fuel oil in its day tank. The main fuel oil storage tank for the unit will contain sufficient fuel oil to supply one standby diesel generator for seven days at rated load. Each standby diesel generator will be provided with a fuel transfer pump having a capacity equal to three times the consumption rate of the diesel generator at nameplate rating. Each transfer pump will be powered from the Class IE source associated with the diesel generator it serves. We conclude that this is acceptable.

The applicants have documented that the design of the alternating-current onsite power system will conform to Criteria 17 and 18 of the General Design Criteria, Regulatory Guides 1.6, 1.9 and 1.32, and IEEE Standard 308-1971 and that its safety-related components will meet the seismic and environmental requirements for seismic Category I equipment. On these bases, we conclude that this preliminary design for the alternating-current onsite power system is acceptable.

8.3.2 Direct-Current Power System

The direct-current power system will consist of four independent Class IE 125-volt direct-current subsystems, one non-Class IE 125-volt direct-current system and one non-Class IE 250-volt direct-current system. The direct-current power system will be designed to provide continuous power for controls, instrumentation, inverters and direct-current emergency auxiliaries.

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The Class IE direct-current system will provide direct-current electric power to the Class IE direct-current loads and for control and switching of the Class IE systems. Physical separation, electrical isolation and redundancy will be provided to comply with the requirements of IEEE Standard 308-1971. Each Class IE direct-current power subsystem will consist of one 125-volt battery, one battery charger, one inverter and one distribution bus. The battery chargers for direct-current subsystems will be supplied with 480-volt alternating-current power from different Class IE buses. The inverters will provide four independent 115-volt alternating-current vital instrumentation and control power supplies for the channels of reactor protection and engineered safety features systems. One spare battery charger and one spare inverter will be provided for the standardized portion of the plant.

Each battery will be sized to supply the emergency load requirements for 200 minutes without battery charger support. Each charger will be automatically energized when the diesel generator supplies the bus. The capacity of each battery charger is based on the largest combined demands of the various steady-state loads and the charging capacity to restore the battery from the design minimum charge state to the fully charged state within 12 hours under any plant operating condition.

The batteries, racks, chargers, and the auxiliary distribution equipment (panelboards) are designated as seismic Category I, and will be designed to maintain their functional capability during and after a safe shutdown earthquake. The four batteries will be located in separate temperature controlled rooms with independent room ventilation systems.

We conclude that the design of the direct-current onsite power system conforms to Criteria 17 and 18 of the General Design Criteria, Regulatory Guides 1.6, 1.32 and IEEE Standard 308-1971. On these bases, we conclude that the design criteria and the proposed design for this system are acceptable.

8.4 Physical Independence of Electric Systems

The criteria for electrical isolation and physical separation which the applicant present in the PSAR essentially duplicate those contained in Regulatory Guide 1.75, "Physical Independence of Electric Systems." On this basis, we find the applicants' criteria for meeting our requirements for physical separation to be acceptable. (We have not completed our review of the Westinghouse 7300 series process instrumentation. Possible nonconformance to Regulatory Guide 1.75 will be resolved generically, as discussed in Section 7.8 of this report.)

Originally, the applicants had taken exception to Subsection C.1 of Regulatory Guide 1.75 which states that, "Interrupting devices actuated only by fault current are not considered to be isolation devices within the context of this document." The applicants had stated that interrupting devices actuated by fault current can be isolation devices when justified by test and analysis. We found no basis for this position, however, and, therefore, found it unacceptable. Accordingly, we required the applicants to follow the recommendations of Regulatory Guide 1.75 with regard to interrupting devices. The applicants have committed to conform with our position as specified above.

8.5 Fire Stops and Seals

We requested the applicants to provide information regarding design criteria and procedures for fire stops and seals to be used in the electrical design of their facility. This information was needed to evaluate the fire detection and protection system for the plant's electrical cables. We have received the applicants' response and will report our conclusions regarding the applicants' criteria for fire stops and seals in a supplement to the Safety Evaluation Report.

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9.0 AUXILIARY SYSTEMS

9.1 Summary Description

We have reviewed the design of the auxiliary systems for the Wolf Creek plant, as described in Section 9.0 of the PSAR, including their safety related objectives and the manner in which these objectives are achieved.

Except for the ultimate heat sink and portions of (1) the essential service water system, (2) the service water system, (3) the potable and sanitary water systems and (4) the fire protection system, the auxiliary systems for the Wolf Creek plant are described in the SNUPPS PSAR and are the same as for the other SNUPPS plants. The other portions of the auxiliary systems are described in the Wolf Creek Site Addendum Report.

The auxiliary systems necessary to assure safe plant shutdown include the essential service water system, the component cooling water system, the chemical and volume control system, the ultimate heat sink, the residual heat removal system, the control room air conditioning system, portions of the auxiliary building ventilation system, the emergency diesel generator auxiliary systems, including diesel engine room ventilation systems, and the fire protection system.

The systems necessary to assure safe handling of fuel and adequate cooling of the spent fuel include the new and spent fuel storage systems, the fuel pool cooling system, and the fuel handling system.

We have reviewed those auxiliary systems whose failure would not prevent safe shutdown but could, either directly or indirectly, be a potential source of a radiological release to the environment. These systems include the equipment and floor drainage system, and the radwaste ventilation system.

We have also reviewed other auxiliary systems and those non-seismically designed systems whose failure would neither prevent safe shutdown nor result in potential radioactive releases. These include the demineralized water makeup system, the service water system, and the turbine building ventilation system.

As a result of our review, we find that the proposed auxiliary systems for the Wolf Creek plant are similar in design and function to those in other pressurized water reactor facilities that have been previously reviewed and accepted. Our specific conclusions for each system that is safety related are presented in the following sections.

9.2 Fuel Storage and Handling

9.2.1 New Fuel Storage

The new fuel storage facility will provide for dry storage of new fuel. The new fuel storage racks will be designed to store 37 percent of a core and to preclude the possibility of insertion of new fuel assemblies in other than prescribed locations. The new fuel storage facility, including the vault and the racks, will be designed as seismic Category I and will be located within the seismic Category I fuel building. The fuel array in the storage racks will be such that when flooded with unborated water, the effective multiplication factor, K_{eff}, will not exceed 0.98.

We have reviewed the design criteria and bases of the new fuel storage facility and find that they are in compliance with Criterion 62 of the General Design Criteria. Therefore, we conclude that the design is acceptable.

9.2.2 Spent Fuel Storage

The spent fuel pool will be designed to provide storage for one and two-thirds cores. The spent fuel storage racks will be designed so that when fully loaded the effective multiplication factor, K_{eff}, of the array will not exceed 0.95 when flooded with non-borated water. The racks will also be designed to prevent fuel assemblies from being inserted in other than prescribed locations.

The fuel building service crane, in conjunction with the fuel handling area, will be designed to preclude moving objects over the spent fuel storage racks in order to prevent the dropping of heavy objects onto these racks. The cask-handling crane, in conjunction with the fuel handling area, will be designed to preclude moving the spent fuel shipping cask over the spent fuel pool to prevent a shipping cask from dropping into the spent fuel pool. The spent fuel pool and the fuel storage racks will be designed as seismic Category I and will be protected against missiles.

The water level in the spent fuel pool will be maintained by addition of borated water from the refueling storage tank. In addition, an intertie with the essential service water system provides a redundant seismic Category I emergency makeup source to assure makeup capability in the event of postulated accidents including a safe shutdown earthquake.

We have evaluated the spent fuel storage facility and find that the system design meets the guidelines of Regulatory Guide 1.13, "Fuel Storage Facility Design Basis," and the requirements of Criterion 62 of the General Design Criteria. Therefore, we conclude that the proposed design of the spent fuel storage facility is acceptable.

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9.2.3 Spent Fuel Pool Cooling System

The spent fuel pool cooling and cleanup system will be designed to remove the decay heat generated by stored spent fuel assemblies. A second function of the system will be to maintain visual clarity and purity of the spent fuel pool water, the fuel transfer canal water, and the refueling pool water.

The spent fuel pool cooling system as originally proposed by the applicants was not designed as seismic Category I. The applicants have since committed to change the design to seismic Category I.

The system piping will be arranged so that loss of piping integrity or operator error will not result in draining the spent fuel pool below a minimum depth above the stored fuel to assure sufficient cooling of the stored spent fuel.

We have reviewed the design criteria and bases and conclude that the design of the spent fuel pool cooling system, as modified, is now acceptable.

9.2.4 Fuel Handling System

The fuel handling system will provide for the safe handling of the spent and new fuel assemblies. The fuel handling system will also provide for the safe disassembly, handling, and reassembly of the reactor vessel head during refueling operations.

All fuel handling equipment will be designed to withstand the forces of the safe shutdown earthquake.

The fuel handling system will be designed in compliance with the applicable recommendations of Regulatory Guide 1.13 and Criterion 61 of the General Design Criteria. We conclude, therefore, that the design of the fuel handling system is acceptable.

9.3 Water Systems

9.3.1 Essential Service Water System

The essential service water system will provide cooling water for plant components that require cooling for safe shutdown of the reactor following postulated accidents. These components include component cooling water heat exchangers, containment air coolers, diesel generator coolers, safety injection pump room coolers, residual heat removal pump room coolers, containment spray pump room coolers, centrifugal charging pump room coolers, component cooling water pump room coolers, auxiliary feedwater pump room coolers, control room air conditioning system condensers, Class IE switchgear air conditioning system condensers, penetration room coolers and the station air compressor.

1563 010

The essential service water system will also provide emergency makeup to the spent fuel pool and component cooling water systems and the backup water supply to the auxiliary feedwater system.

The system will be designed to seismic Category I standards and to the single failure criterion. The system will also be designed so that postulated environmental occurrences cannot impair the system's functional capability.

The essential service water system will have two redundant flow paths with independent supply headers which will be fed separately from the ultimate heat sink cooling lake. The system serves two identical trains of engineered safety features equipment which are required for safe shutdown of the plant after a postulated accident. Either essential service water flow path will be capable of supplying the required cooling water to meet the single failure criterion.

Each train of the essential service water system will contain a 100 percent capacity water pump required to remove the heat from the plant. The pumps will be cross-connected to provide the flexibility of feeding either essential service water header with either pump. Two locked-closed isolation valves are provided on this cross-connection.

The essential service water system pumps will be located in a pumphouse designed to protect the pumps and other components against tornadoes, tornado missiles, and the safe shutdown earthquake. Other parts of the essential service water system will be protected against tornado missiles by either being buried underground or being located in seismic Category I structures. All structures and components of the system will be located such that the failure of any non-seismic Category I structure would not constitute a hazard to the system.

We have reviewed the applicants' design criteria and bases and the single-failure analysis for the essential service water system and find that they meet the applicable requirements of Regulatory Guide 1.29, "Seismic Design Classification," and Criteria, 2, 4, 5, 44, 45, and 46 of the General Design Criteria. Therefore, we conclude that the system design is acceptable.

9.3.2 Component Cooling Water System

The component cooling water system will provide cooling water to selected nuclear auxiliary components during normal plant operation and to engineered safety feature systems under postulated accident conditions, including a postulated loss-of-coolant accident. During accident conditions, the component cooling water system will provide the required cooling water for heat removal from the residual heat removal pumps, the containment spray pumps, the safety injection pumps and the centrifugal charging pumps.

The component cooling water system design will include two separate and redundant flow trains, each capable of providing sufficient cooling for safe shutdown.

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Safety-related portions of the system, including sources of makeup, will be designed to remain functional following a safe shutdown earthquake. A single failure in either train will not affect the functional capability of the other train. Furthermore, all components will be isolatable and the system will be monitored for radioactive inleakage.

For added operational flexibility, each of the two redundant flow trains will be inter-connected to the other train on both the pump suction side and the pump discharge side. Each interconnecting line will have two manually-operated isolation valves. This arrangement, together with redundant active components, will enable the component cooling water system to provide coolant to the required equipment even considering the moderate energy line break criteria.

The normal makeup to the component cooling water system will be from the demineralized water system. The emergency makeup will be provided from the essential service water system. The proposed design also includes remote-operated valves to permit alignment of the system for safe shutdown operation during postulated accidents.

As a result of our review of the design criteria and bases, we conclude that the design of the component cooling water system is acceptable.

9.3.3 Ultimate Heat Sink

The ultimate heat sink for the Wolf Creek plant will consist of a cooling lake to be impounded by a main dam and several saddle dams. The safety-related portion of the ultimate heat sink will be the water impounded by a submerged dam within the cooling lake. The essential service water system will normally obtain its cooling water from the cooling lake via the circulating water system and the service water system. During postulated accident conditions, the essential service water system will obtain its cooling water from the pond impounded by the submerged dam.

The submerged dam will be designed as seismic Category I. It will also be designed to withstand the most severe postulated natural phenomena including the loss of the main dam.

We have reviewed the ultimate heat sink design criteria and bases and find that they meet the recommendations of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants." Therefore, we conclude that the system is acceptable.

9.4 Process Auxiliaries

9.4.1 Compressed Air System

The compressed air system will consist of three compressors, three aftercoolers, three air receivers, a filtering and drying unit for instrument air, and alarm and control panels, all of which will be located in the turbine building. The compressed air system will have no safety function since compressed air is not necessary for safe shutdown of the plant. All pneumatically operated devices in the plant that are essential for safe shutdown will be designed to fail to the safe position upon loss of air.

We have reviewed the design criteria and bases for the compressed air system and conclude that the design is acceptable.

9.4.2 Equipment and Floor Drain System

The equipment and floor drain system will collect liquid wastes for processing and disposal. The system has no direct safety-related function.

The system will be designed such that potentially radioactive areas and non-radioactive areas are served by separate systems. Separate subsystems will be provided in redundant rooms containing engineered safety feature equipment. Drains in the auxiliary building will be run so that leakage in one engineered safety feature train will not flow through the drains into the rooms of the other train, and so that each train will drain to a separate sump. The waste is then pumped to the floor drain tank for reprocessing.

Based on our review of the design bases, we conclude that the design of the equipment and floor drain system is acceptable.

9.4.3 Chemical and Volume Control System

The chemical and volume control system will be designed to control and maintain reactor coolant inventory. Purification of the letdown fluid by removal of corrosion and fission products will also be accomplished by the makeup and purification portions of this system. The design of this system is in conformance with that described and analyzed in RESAR-3, Consolidated Version.

The system will be designed as seismic Category I for all essential portions required for safe reactor shutdown. To evaluate system safety, component failures or malfunctions were assumed concurrent with a postulated loss-of-coolant accident and the consequences evaluated. We find the results of the failure analyses to be acceptable.

On the basis that the design of the chemical and volume control system is the same as that for RESAR-3, Consolidated Version, which was previously reviewed and accepted for other applications, we conclude the design criteria and bases are acceptable.

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9.5 Air Conditioning, Heating, Cooling and Ventilation Systems

9.5.1 Control Building

The control building air conditioning, heating, cooling and ventilation systems will consist of the control building supply and exhaust systems, the control room, Class IE electrical equipment and access-control air conditioning systems, and the access-control exhaust system.

The control room air conditioning system will comprise completely redundant, independent, full-capacity air-side and mechanical refrigeration subsystems. Each subsystem will be powered from independent, Class IE power sources and will receive cooling water from a separate essential service water header. The control room filtration and pressurization systems will consist of completely redundant full-capacity fan and charcoal adsorption units. Each subsystem will be powered from independent Class IE power sources.

The Class IE electrical equipment air conditioning system will consist of completely redundant, independent, and full capacity air-side and mechanical refrigeration subsystems. Each subsystem will be powered from independent Class IE power sources and will receive cooling water from a separate essential service water header.

The control room air conditioning system, including the filtration and pressurization systems, and the Class IE electrical equipment air conditioning systems, will be designed as seismic Category I. A single failure will not prevent these systems from performing their intended functions. Those portions of the systems that will have direct connections with the outside will be provided with isolation provisions that will be designed to withstand the differential pressure associated with extreme wind and tornado conditions.

Alarms will be provided in the control room to indicate high radiation in the control building, high temperature in the charcoal adsorber beds, and smoke, high radiation, and high chlorine gas concentration in the control building intake. In addition, local and control room alarms will be provided to indicate detection of smoke at each of the various levels of the control building.

Indication of a loss-of-coolant accident, a fuel-handling accident, chlorine gas (greater than 1.0 part per million by volume), or high radiation in the influent will automatically isolate the control building by closing the outside air dampers in the building ventilation and exhaust systems.

We have reviewed the design criteria and bases and evaluated the single failure analysis provided by the applicants and find that the system design criteria and bases are consistent with the applicable recommendations of Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Atmospheric Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," and meet the

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requirements of Criterion 19 of the General Design Criteria. Therefore, we conclude that the design of the control building air conditioning, heating and ventilation systems is acceptable.

9.5.2 Auxiliary Building

The auxiliary building heating and ventilation system will comprise those subsystems that function to maintain a suitable environment for equipment and personnel during normal plant operating and shutdown conditions and subsequent to a design basis accident.

The system will include local pump room coolers designed to maintain ambient temperatures in various rooms suitable for continued operation of motors necessary for safe shutdown of the plant. These rooms include the centrifugal charging pump rooms, safety injection pump rooms, containment spray pump rooms, residual heat removal pump rooms, component cooling water pump rooms, and the motor-driven auxiliary feedwater pump rooms. In addition, the system will include coolers for the penetration rooms to provide a suitable ambient temperature for the electrical equipment located in these rooms.

These coolers will also be designed to remain functional during and after a postulated safe shutdown earthquake. Furthermore, the pump room coolers and the penetration room coolers will be designed so that a single failure and loss of offsite power, can not impair the system's functional capability.

Based on our review and evaluation of the design criteria and bases and the single failure analysis provided by the applicants, we conclude that the design of the auxiliary building heating and ventilation system is acceptable.

9.5.3 Fuel Building

The fuel building ventilation system will provide a suitable atmosphere for personnel and equipment located in the fuel building. The fuel-handling accident is discussed in Section 15.6 of this report.

The fuel building ventilation exhaust system and the intake air isolation system will be designed as seismic Category I. Sufficient redundancy will be provided to assure functional capability of the system in the event of a single failure.

Based on our review and evaluation of the design criteria and bases and the single failure analysis provided by the applicants, we conclude that the fuel building ventilation system is acceptable.

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9.5.4 Diesel Generator Building

The diesel generator building ventilation system will be designed to provide a suitable atmosphere for the operation of the diesel generators. The system will be automatically activated upon starting of the diesel generator and will automatically isolate when the diesel generator is shutdown. A manual override will provide means to manually activate the system to provide cooling during occupation of the building regardless of the operating mode of the diesel engine. Supply fans will direct outside air into the respective diesel generator room to meet cooling requirements. The exhaust fans will take suction from their respective diesel generator room and discharge directly to the atmosphere. A recirculation mode between the exhaust and supply air will be provided to maintain the ambient temperature in a specified range. This recirculation mode will primarily be used for winter operation to prevent freezing.

The system will have sufficient redundancy and electrical separation to maintain its function in the event of a single failure. The system will be designed as seismic Category I.

Each diesel generator ventilation subsystem will be located in a separate penthouse above its respective diesel generator and each penetration through the penthouse will be provided with protection from external missiles.

Based on our review and evaluation of the design criteria and bases and the single failure analysis provided by the applicants, we conclude that the design of diesel generator building ventilation system is acceptable.

9.5.5 Essential Service Water Pumphouse

The essential service water pumphouse ventilation system will provide a suitable atmosphere for the essential service water pump motors located in the pump rooms. Each pump room will be provided with a separate subsystem powered from the same Class IE power source as its associated pump. Electric unit heaters will also provide heating in each room.

Each system will be designed as seismic Category I and will be located within the associated pump room. Each penetration will be protected from external missiles.

Each system will be automatically actuated upon the start of its associated essential service water pump. The supply fans will take suction from the outside air and discharge it directly into the pumprooms. The exhaust fans will take suction from their respective pump rooms and discharge to the atmosphere. Necessary controls will be provided to maintain the pump room temperature within a predetermined range.

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We have reviewed and evaluated the design bases and criteria, and the single failure analysis provided by the applicants, and conclude that the design of the essential service water pumphouse ventilation and heating systems is acceptable.

9.6 Other Auxiliary Systems

9.6.1 Fire Protection System

The fire protection system will be designed to detect fires, to protect the plant against damage from fire, and to minimize hazards to personnel and reduce property loss due to fires.

The fire protection system will include diversified monitoring, detection, alarm, suppression and extinguishing equipment to protect the area or equipment from damage by fire. The major subsystems will include hydraulically-designed automatic wet sprinklers, hydraulically-designed water spray systems, automatic pre-action sprinklers, Halon 1301 fire suppression systems, standpipes and hose racks, portable extinguishers, fire walls and barriers, and fire and smoke monitoring, detection and alarm systems.

Fire and smoke detection systems will be provided to serve the cable spreading room, control room, computer room, diesel generator rooms, switchgear rooms, battery rooms, motor control center, load centers, and inside containment.

Fire barrier walls will be provided to isolate the lubricating oil storage room, control rooms, cable spreading rooms, auxiliary building, containment, fuel building, essential service water pumphouse, Class IE switchgear, Class IE batteries and switchgear, auxiliary boiler, turbine building, computer rooms, emergency diesel generator rooms, and rooms enclosing engineered safety feature components. Noncombustible construction material will be employed throughout all buildings to minimize fire potential. The use of heat- and flame-resistant construction materials throughout all buildings will reduce the potential of a fire hazard, particularly in areas that contain, or may interact with, safety-related equipment. The basic fire protection for a safety related area or equipment will be achieved through separation or by fire barriers.

Halon 1301 fire suppression systems for total flooding will be provided to protect switchgear rooms, cable spreading rooms, the computer room, electrical penetration rooms, battery rooms, and electrical cable chases from fires. Operation of the Halon 1301 system will be initiated by a rapid rise in ambient temperature or attainment of a fixed high temperature. Halon 1301 will be stored in cylinders in strategic locations and will not release through the piping leading from the cylinder to the area where fire is detected, until the sensor actuates a signal to open the valve at the storage cylinder. Since the pressure in each cylinder will be monitored periodically, there will be little danger of undetected leaks of Halon 1301 into the areas through the piping. Electrical penetrations into the control room will be sealed to prevent leakage of Halon 1301 into the control room.

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Safe shutdown of the plant during and after a fire in a safety-related area will be accomplished by redundant components of the safety related systems.

The fire protection system outside the scope of the SNUPPS standard plant design will include two 100 percent capacity fire pumps (one motor-driven and one diesel-driven), one motor-driven jockey pump, yard mains, hydrants, standpipes, hose stations, and fire detection devices for site buildings.

Water for the fire protection system will be provided from the circulating water screen house intake bay. Wet standpipe hose stations will be placed in applicable areas of the administration and shop buildings. Wet pipe sprinkler systems will be provided in the diesel fire pump room of the circulating water screen house and throughout the warehouse. Sprinkler systems will be initiated automatically.

We reviewed and evaluated the design criteria and bases of the fire protection system and find that they satisfy the requirements of Criterion 3 of the General Design Criteria. Therefore, we conclude that the system is acceptable.

9.6.2 Diesel Generator Fuel Oil Storage and Transfer System

The diesel generator fuel oil storage and transfer system will provide onsite storage and delivery of fuel oil to the diesel generators.

The system will provide onsite storage of fuel oil for at least seven days of operation for the diesel generators following a loss of offsite power. The applicants state that, within this period, additional fuel could be delivered to the plant site by truck or rail.

The system will be designed as seismic Category I and will be protected from missiles, flooding, fire and freezing. The system will consist of two completely redundant and independent systems, each powered by a separate Class IE source.

We have reviewed and evaluated the design criteria and bases, and single failure analysis provided by the applicants, and conclude that the diesel generator fuel oil storage and transfer system is acceptable.

9.6.3 Diesel Generator Cooling Water System

The diesel generator cooling water system will provide the intermediate cooling system between the diesel generator and the essential service water system.

Each diesel engine will have its own cooling water system which will consist of a closed cycle jacket cooling water system, a standby jacket coolant heater, an air intercooler, and a lube oil cooler, pumps and valves. The heat will be rejected to the essential service water system.

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The system will be designed as seismic Category I. We conclude that the proposed diesel generator cooling water system design is acceptable.

9.6.4 Diesel Generator Starting System

Each diesel engine will have its own starting system to be supplied as a complete package with each diesel generator unit. Each diesel generator will be provided with two completely independent starting air systems, each consisting of separate air compressors, air storage tanks and associated valves, piping and controls. The air compressors will be electric-motor driven. Compressed air will be stored in two independent tanks, each with sufficient storage capacity to start the diesel engine five times without compressor assistance.

The systems will be designed as seismic Category I and will be protected from floods and missiles. Sufficient redundancy will be provided and the redundant systems will be arranged so that a malfunction or failure in one system will not impair the ability of the other system to start the diesel engine.

We have reviewed the design criteria and bases of the diesel generator starting system and evaluated the single failure analysis provided by the applicants and conclude that the system is acceptable.

9.6.5 Diesel Generator Lubrication System

The diesel generator lubrication system will provide lubricating oil to the various moving parts of the diesel engine. Each diesel generator will be provided with an independent engine lubrication system consisting of an oil circulation pump, driven by the diesel engine, an auxiliary, alternating-current powered, oil circulation pump, an oil filter and an oil cooler which will reject heat to the engine cooling water system.

All system components will be manufactured and mounted to withstand the effects of a safe shutdown earthquake. We have reviewed the design criteria and bases of the diesel generator lubrication system and conclude that the system is acceptable.

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10.0 STEAM AND POWER CONVERSION SYSTEM

10.1 Summary Description

The steam and power conversion system for the Wolf Creek plant is described in Section 10.0 of the SNUPPS PSAR. The circulating water system portion of this system is described in Section 10.0 of the Wolf Creek Site Addendum Report.

The steam and power conversion system will be of conventional design, similar to those of previously approved plants. The system will be designed to remove heat energy from the reactor coolant by means of four steam generators and to convert it to electrical energy by means of the turbine-driven generator. The condenser transfers the unusable heat in the cycle to the circulating water system. The entire system will be designed for the maximum expected energy from the nuclear steam supply system.

In the event of a turbine trip or a large load reduction, the heat transferred from the reactor coolant to the steam generators will be dissipated via the turbine bypass system to the condenser, or through the power operated relief valves and safety valves to the atmosphere if the condenser is not available.

10.2 Turbine-Generator

The function of the turbine-generator is to receive steam from the steam generators and convert a portion of the thermal energy in the steam into electrical energy.

The turbine control system includes control actions, alarms and trips initiated by deviation of steam parameters and of turbinegenerator performance from preset values. The automatic control functions will be programmed to protect the reactor coolant system and the turbine-generator by appropriate corrective actions.

The turbine overspeed protection system will provide two independent and redundant means of tripping the turbine before design overspeed is approached. The primary trip device will be the mechanical-hydraulic overspeed trip with redundancy being provided by the electrical-hydraulic backup overspeed trip. The electrical-hydraulic backup overspeed trip will be set to operate at a speed approximately 0.5 percent above the mechanical-hydraulic overspeed trip setting.

In addition to overspeed trips, the turbine control system will provide means of tripping the turbine in response to various operating conditions. These conditions include: excessive thrust bearing wear, reactor trip, generator electrical trips, manual trip from the control room, excessive vibration, manual lever located at the turbine, no-load trip, moisture separator drain system high-level, prolonged loss of stator coolant, low hydraulic fluid pressure, loss of both speed signals, loss of the

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direct-current power supply to the electrohydraulic control system, and low bearing oil pressure.

We have reviewed the turbine generator overspeed protection design criteria and bases and conclude that they are acceptable.

10.3 Main Steam Supply System

The function of the main steam supply system is to convey steam from the steam generators to the high-pressure turbine and other auxiliary equipment.

Steam generated in each steam generator will be routed to the turbine by a main steam line. Each main steam line will contain one power operated relief valve, five spring-loaded safety valves and one main steam isolation valve. The main steam supply system, up to, and including, the main steam isolation and bypass valves, the auxiliary feedwater pump turbine driver, and the main steam safety and relief valves, will be designed as seismic Category I. The main steam valve operators and associated circuitry will also be designed as seismic Category I and, together with the main steam isolation valves, will be protected against missiles.

The main steam isolation valves will be capable of isolating the steam generators within five seconds of receiving a signal from the engineered safety features actuation system. In the event of a steam line break, this action will prevent continuous uncontrolled steam release from more than one steam generator. The protection will be afforded against breaks inside or outside the containment. The applicants have agreed to provide an analysis in the Final Safety Analysis Report to demonstrate the valve's capability to close within the specified time under accident flow rates.

We have reviewed the design requirements and bases for the main steam supply system and conclude that they are acceptable.

10.4 Circulating Water System

The circulating water system will furnish the service water system and the main steam condenser with cooling water from the cooling lake.

The applicants analyzed the effects of a complete rupture of an expansion joint in the circulating water system near the condenser within the turbine building. Level switches will be installed in the condenser pit to stop the circulating water pumps and to close the pump discharge valves automatically upon a high water level (four feet from the top of the pit) indication in the condenser pit. In addition, high water level in the sumps located in the condenser pit will be alarmed in the control room and sump pumps will be provided. For their analysis, the applicants conservatively assumed the failure of the circulating water system isolation valves to close fully and the failure of the sump pumps in the condenser pit.

Based on the results of this analysis, the applicants state that the volume of the water which will drain to the turbine building will not exceed the combined volume of the condenser pit and the cavity below the turbine-generator. Therefore, for this postulated accident, flooding above grade level in the turbine building will not occur.

We conclude that the design criteria and bases for the circulating water system are acceptable.

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11.0 RADIOACTIVE WASTE MANAGEMENT

11.1 Summary Description

The radioactive waste management systems for the Wolf Creek plant are described in Section 11.0 of the SNUPPS PSAR. The offsite radiological monitoring program and the estimated doses due to the anticipated releases of gaseous and liquid radioactive effluents are described in Section 11.0 of the Wolf Creek Site Addendum Report.

The radioactive waste management systems will be designed to provide for controlled handling and treatment of liquid, gaseous and solid wastes. The liquid waste system will process decontamination and laboratory wastes, laundry and shower wastes and wastes from equipment and floor drains. The gaseous waste system will provide holdup capacity to decay short lived noble gases, stripped from the primary coolant, and treatment of ventilation exhausts through high efficiency particulate air filters and charcoal absorbers, as necessary to reduce releases of radioactive materials to "as low as practicable" levels in accordance with 10 CFR Parts 20 and 50.34a. The solid waste system will provide for the solidification, packaging and storage of radioactive wastes generated during station operation prior to shipment offsite for burial. Solid packaged wastes will be shipped to a licensed facility for burial.

Our evaluation of the radioactive waste management systems, as presented below and as presented in the Draft Environmental Report for the Wolf Creek plant, dated July 1975, was performed to determine conformance with the design objectives of our report "Concluding Statement of Position of the Regulatory Staff", Docket No. RM-50-2, dated February 20, 1974. We have not completed our review of these systems to establish conformance with the dose design objectives of Appendix I to 10 CFR Part 50 (effective June 4, 1975) nor have we completed the required cost-benefit analysis. To effectively implement the requirements of Appendix I to 10 CFR Part 50, we are presently reassessing the parameters and mathematical models used in calculating releases of radioactive materials in effluents. After we complete our review of the radioactive waste management systems to determine conformance with the design objectives of Appendix I to 10 CFR Part 50, we will report the results in a supplement to the Safety Evaluation Report.

In our evaluation of the radioactive waste management systems we have considered: (1) the capability of the systems to maintain releases below the limits in Appendix B to 10 CFR Part 20 (Column 2 of Table II), during periods of fission product leakage at design levels from the fuel, (2) the capability of the systems to meet the processing

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demands of the station during anticipated operational occurrences, (3) the quality group classification and seismic category applied to the system design, (4) the design features incorporated to preclude uncontrolled releases of radioactive materials due to tank overflows and (5) the potential for gaseous release due to hydrogen explosions in the gaseous radwaste system.

In our evaluation of the solid radwaste treatment system we have also considered: (1) system design objectives in terms of expected types, volumes and activities of waste processed for shipment offsite, (2) waste packaging and conformance to applicable federal packaging regulations, and provisions for controlling potentially radioactive airborne dusts during baling operations, and (3) provisions for onsite storage prior to shipping.

In our evaluation of the process and effluent monitoring system we have considered the system's capability (1) to control the release of radioactive materials to the environment, (2) to monitor all normal and potential pathways for release of radioactive materials to the environment, and (3) to monitor the performance of process equipment and detect radioactive material leakage between systems.

We have determined the quantities of radioactive materials that will be released in the liquid and gaseous effluents and the quantity of material that will be shipped offsite as solid waste for burial during normal opera;tion. The principal radionuclides associated with liquid and gaseous effleunts and solid wastes are listed in the Draft Environmental Statement for the Wolf Crrek plant, dated July 1975, In making these determiniations we considered waste flows and activities and equipment performance consistent with expected normal plant operation, including anticipated operational occurrences, over the 40-year life of the plant. Liquid and gaseous source terms were calculated using the PWR-GALE code described in "Draft Regulatory Guide 1.BB, Calculation of Releases of Radioactive Materials in Liquid and Gaseous Effluents from Pressurized Water Reactors (PWR's)," Docket No. RM-50-2, February 20, 1974. The principal parameters used in these calculations, along with their bases, are given in Appendix B to Draft Regulatory Guide 1.BB.

11.2 Liquid Waste System

The liquid radioactive waste treatment system will consint of process equipment and instrumentation necessary to collect, process, monitor and recycle or dispose of radioactive liquid wastes. The liquid radioactive waste will be processed on a batch basis to permit optimum control of releases. Prior to being released, samples will be analyzed to determine the types and amounts of radioactivity present. Based on the results of the analysis,the waste will be retained for further processing, recycled for eventual use in the plant or released under controlled conditions. The boron recycle system will process a portion of the chemical and volume control system flow (shim bleed) for boron control along with nonaerated waste collected in the reactor coolant drain tank. The principal boron recycle system processing will consist of evaporation

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and demineralization. Aerated radioactive wastes will be segregated based on their origin and processed through the liquid waste processing system. Turbine building floor drain wastes will be discharged without treatment unless sampling indicates processing through the liquid waste processing system is necessary. Detergent (laundry and decontamination) wastes will be processed through a waste treatment system which includes a holdup tank, a reverse osmosis unit and a waste monitoring tank. The principal components making up each of these systems, along with their principal design parameters, are listed in Table 11.1.

The design capacities of the boron recycle system and liquid waste processing system evaporators will be 21,000 and 50,000 gallons per day, respectively. We calculated the average expected waste flows to the boron recycle system and liquid waste processing system to be 2200 and 2000 gallons per day, respectively. The difference between the expected flows and design capacity will provide adequate reserve for processing surge flows. We consider the system capacity and system design to be adequate for meeting the demands of the plant during anticipated operational occurrences. Blowdown from the steam generators normally will be returned directly to the turbine condenser. However, the system will be designed to permit release of blowdown to the circulating water system discharge line. Blowdown will be discharged without treatment unless radionuclide concentrations exceed a predetermined level. In this case, flow will be automatically terminated by one of two radiation monitor controlled valves and blowdown will be processed through a system consisting of two mixed bed demineralizers. The demineralizers will have a design processing capacity of 288,000 gallons per day. We calculate the average expected blowdown rate will be approximately 72,000 gallons per day. We consider the system design capacity to be adequate for meeting the needs of the plant.

The liquid radwaste systems will be located in a seismic Category I structure. The seismic and quality group designations of the equipment, which are consistent with our guidelines, are listed in Table 11.1. The system will also be designed to preclude the uncontrolled release of radioactive materials due to overflows from indoor and outdoor tanks by providing level instrumentation which will alarm in the control room, and by use of curbs and retention walls to collect liquid spillage and retain it for processing. We consider these provisions to be capable of preventing the uncontrolled release of radioactive materials to the environment. We find the applicants' proposed system design to be acceptable.

We have determined that during normal operation the proposed liquid radwaste treatment systems will be capable of reducing the release of radioactive materials in liquid effluents to approximately 0.29 curies per year excluding tritium and dissolved gases, and 350 curies per year for tritium. An isotopic listing of our calculated

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TABLE 11.1

DESIGN PARAMETERS OF PRINCIPAL COMPONENTS
CONSIDERED IN LIQUID RADWASTE EVALUATION

COMPONENTS	NO.	CAPACITY EACH**	QUALITY GROUP*
Boron Recycle System			
Recycle Holdup Tank	2	56.000 gal	C
Evaporator Feed Demineralizer	2	35 gpm	C
Evaporator	1	15 gpm	C
Evaporator Condensate Demineralizer	1	35 gpm	D
Liquid Waste Processing System			
Laundry and Hot Shower Tank	1	10,000 gal	D
Reactor Coolant Drain Tank	1	350 gal	D
Floor Drain Tank	1	10,000 gal	D
Waste Holdup Tank	1	10,000 gal	C
Waste Monitor Tank	2	5,000 gal	D
Chemical Drain Tank	1	600 gal	D
Waste Evaporator Condensate Tank	1	5,000 gal	D
Waste Evaporator	2	35 gpm	C
Mixed Bed Demineralizer	2	35 gpm	D
Reverse Osmosis Unit	1	2 gpm	D
Spent Resin Storage	1	2,600 gal	C
Solid Waste Processing System			
Solidification Holdup Tank	1	1,500 gal	D
Evaporator Bottoms Tank	1	500 gal	D
Spent Resin Storage Tank	1	4,000 gal	D
Steam Generator Blowdown			
Mixed Bed Demineralizer	2	200 gpm	D
Blowdown Surge Tank	1	1,400 gal	D

* Quality Group C components will be of seismic Category I design and Quality Group D components will be of non-seismic design.

** gal - gallons
gpm - gallons per minute

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liquid source term is given in Table 3.6 of the Draft Environmental Statement for the Wolf Creek facility. Based on that evaluation, we have determined that the release of radioactive materials in liquid effluents will not result in whole body or critical organ doses in excess of five millirems per year at or beyond the site boundary, and that the proposed system will be capable of limiting the release of radioactive material in liquid effluents, exclusive of tritium and dissolved gases, to less than five curies per year.

We have considered the consequences of reactor operation with one percent of the operating fission product inventory in the core being released to the primary coolant. We have determined that, under these conditions, the concentrations of radioactive materials in liquid effluents in unrestricted areas will be a small fraction of the limits defined in Appendix B to 10 CFR Part 20 (Column 2 of Table II).

The consequences of component failures which will result in contaminated liquid releases to the environs were evaluated for components containing liquid radioactive materials located outside reactor containment. The scope of the review included the calculation of radionuclide inventories in station components at design basis fission product levels, the mitigating effects of the plant design, and the effect of site geology and hydrology.

The tank that will contain the highest total quantity of activity is the solidification holdup tank. This tank will have a volume of 1500 gallons and is assumed to be 80 percent full with a liquid activity concentration of approximately six microcuries per milliliter (based on one percent operating power fission product inventory released to the primary coolant). The flow of ground water will move radionuclides in the direction of the ultimate heat sink cooling lake (see Section 2.4.4), and will result in a liquid transit time of about 38 years. We estimate a ground water dilution factor of 1,700 and a dilution factor in the cooling lake of 2.4×10^{7}.

Considering dilution only, a rupture of the solidification holdup tank will give a concentration of 1.5×10^{-10} microcuries per milliliter in the cooling lake. This value is a small fraction of the limits in Appendix B to 10 CFR Part 20 (Column 2 of Table II) for unrestricted areas.

Based on the foregoing evaluation we conclude that the postulated failure would not result in radionuclide concentrations in excess of 10 CFR Part 20 limits at the nearest potable water supply. Therefore, it is not necessary for the applicants to incorporate additional provisions in their design to mitigate the effects of component failures involving contaminated liquids.

Our review of the liquid radwaste system included (1) the system's capability to process the types and volumes of wastes expected during normal operations and anticipated operational occurrences in accordance with Criterion 60 of the General Design

Criteria (2) the design provisions incorporated in accordance with Criterion 60 to preclude uncontrolled releases of radioactive material due to leakage or overflows, (3) the quality group classification and seismic design criteria and their conformance with Commission guidelines, and (4) the design provisions incorporated in conformance with Regulatory Guide 8.8 (paragraph C.3), "Information Relevant to Maintaining Occupational Radiation Exposures as Low as Practicable (Nuclear Reactors)". We have reviewed the applicants' system descriptions, process flow diagrams, piping and instrumentation diagrams and design criteria for the components of the liquid waste treatment system and for those auxiliary supporting systems that are essential to the operation of the liquid radwaste treatment system. We have also performed an independent calculation of the releases of radioactive liquid effluents, based on the calculational methods of Draft Regulatory Guide 1.BB.

We have determined that the applicants' designs, design criteria and design bases for the liquid radioactive waste treatment and monitoring system are in conformance with the design objectives of our report, "Concluding Statement of Position of the Regulatory Staff", Docket No. RM-50-2, dated February 20, 1974.

We have not completed our review of the liquid radwaste treatment system's capability of meeting the dose design objectives of Appendix I to 10 CFR Part 50 (effective June 4, 1975) and the required cost-benefit analysis. We will report the results of this review in a supplement to the Safety Evaluation Report.

11.3 Gaseous Waste System

Gaseous radwaste treatment systems will be designed to process gaseous plant wastes based on the origin of the wastes in the plant and the expected activity levels. The gaseous waste treatment system will consist of a gaseous waste processing system, a main condenser effluent processing system and ventilation systems that control the release of radioactive effluents to the environment.

The gaseous waste processing system will collect and process gases stripped from the primary coolant along with miscellaneous tank cover gases in a continuously recirculating nitrogen loop. Operating with two 40 standard cubic feet per minute compressors (one in continuous use and the other as backup) and eight 600 cubic-foot gas decay tanks (each of which is capable of being isolated from all others), the gaseous waste processing system will have adequate capacity to allow operation during periods of equipment downtime. We consider the system capacity and the system design to be adequate for meeting the demands of the plant during normal operations and anticipated operational occurrences. The system design includes redundant hydrogen and oxygen analyzers, downstream of the recombiners, which will initiate an alarm if hydrogen or oxygen concentrations vary beyond the design concentration limits. The system design will limit the hydrogen concentration downstream of the recombiner to three percent by volume and the oxygen concentration to 60 parts per million in the recombiner discharge line by automatically terminating the hydrogen and/or oxygen flow as required.

In this manner the potential for explosive hydrogen/oxygen mixtures will be minimized.

The system will be designed as Quality Group C and seismic Category I and will be located in a seismic Category I structure. We find the system quality group, seismic design criteria, and the design provisions incorporated to reduce the potential of hydrogen explosion, to be acceptable.

Gaseous wastes from the main condenser will be processed through high efficiency particulate air filters for particulate removal and through charcoal adsorbers for iodine removal. Noble gases will not be affected by the treatment provided. The system releases will be proportional to the rate of primary to secondary system leakage and the primary coolant activity. In the event of excessive primary to secondary leakage, the affected steam generator will be isolated before radioactive material concentrations in the main condenser offgas exceeds the limits in 10 CFR Part 20.

Ventilation exhausts from the auxiliary, radwaste and fuel handling buildings will be processed through high efficiency particulate air filters and charcoal adsorbers prior to release. Ventilation exhaust from the containment building will be processed through high efficiency particulate air filters prior to release. In addition, the containment building atmosphere will be recirculated through filters and charcoal adsorbers prior to purging to the ventilation exhaust system. The turbine building ventilation exhausts will be released to the environment without treatment. The plant ventilation systems will be designed to induce air flows from potentially less radioactively contaminated areas to areas having a greater potential for radioactive contamination. The ventilation system will have adequate capacity to limit radioactive material concentrations, in areas within the plant that are accessible during operation, to below the limits in 10 CFR Part 20.

We have determined that the proposed gaseous radwaste treatment systems and plant ventilation systems will be capable of reducing the release of radioactive materials in gaseous effluents to approximately 1400 curies per year of noble gases and 0.04 curies per year of iodine-131. An isotopic listing of our calculated gaseous source term is given in Table 3.7 of the Draft Environmental Statement for the Wolf Creek facility. Based on that evaluation, the release of radioactive materials in gaseous effluents will not result in an annual air dose, at or beyond the site boundary, in excess of 10 millirad for gamma radiation and 20 millirad for beta radiation, the annual thyroid dose to an individual will not exceed 15 millirem considering the location of the nearest cow and the annual quantity of iodine-131 released will not exceed one curie.

We have reviewed the effects of reactor operation with one percent of the operating fission product inventory in the core being released to the primary coolant. We have

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determined that under these conditions, the concentrations of radioactive materials in gaseous effluents will be a small fraction of the limits in 10 CFR Part 20.

Our review of the gaseous radwaste system included (1) the system's capability to reduce releases of radioactive materials in gaseous effluents to "as low as practicable" levels, in accordance with 10 CFR Parts 20 and 50.34a, considering normal operation and anticipated operational occurrences, (2) the quality group and seismic design criteria, and (3) the design provisions incorporated to reduce the potential for hydrogen explosions. We have reviewed the applicants' system descriptions, process flow diagrams, piping and instrumentation diagrams and design criteria for the components of the gaseous waste treatment system and for those auxiliary supporting systems that are essential to the operation of the gaseous radwaste treatment system. We have also performed an independent calculation of the releases of radioactive materials in gaseous effluents, based on the calculational methods in Draft Regulatory Guide 1.BB.

We have determined that the applicants' designs, design criteria, and design basis for the gaseous radwaste system are in conformance with the design objectives of our report, "Concluding Statement of Position of the Regulatory Staff," Docket No. RM-50-2, dated February 20, 1974.

We have not completed our review of the gaseous radwaste system's capability of meeting the dose design objectives of Appendix I to 10 CFR Part 50 (effective June 4, 1975) and the required cost-benefit analysis. We will report the results of this review in a supplement to the Safety Evaluation Report.

11.4 Solid Waste System

The solid radwaste treatment system will be designed to collect and process wastes based on their physical form and the need for solidification prior to packaging. Wet solid wastes, consisting of spent demineralizer resins, evaporator bottoms, filter sludges, and chemical drain tank effluents, will be combined with a solidification agent and catalyst to form a solid matrix and sealed in the shipping containers. Dry solid wastes, consisting of ventilation air filters, contaminated clothing and paper, and miscellaneous items, such as tools and glassware, will be compacted in 55-gallon steel drums. Miscellaneous solid wastes, such as irradiated primary system components will be handled on a case-by-case basis, based on their size and activity. Expected solid waste volumes and activities shipped offsite will be 600 drums per year of wet solid waste containing an average of 10 curies per drum and 450 drums per year of dry solid waste containing less than five curies total.

Drum filling operations will be controlled remotely from consoles located outside the drum fill area. Drumming operations will have interlock features to prevent opening of filling valves when a drum is not properly positioned in the filling station. In addition, the equipment will be designed so that any spills will be collected in a drain pan and prevented from dripping on the floor. Baling of dry wastes will be carried out inside a closed dust shroud.

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The solid radwaste systems will be located in a seismic Category I structure. The seismic and quality group designations of the equipment, which are consistent with our guidelines, are listed in Table 11.1.

Storage facilities for up to 800 drums of solid radioactive wastes will be provided below grade in the radwaste building. Based on our estimate of 1050 drums per year, we find the storage capacity adequate for meeting the demands of the plant. Wastes will be packaged in 55-gallon steel drums in accordance with the requirements of 10 CFR Parts 20 and 71 and 49 CFR Parts 170 through 178, and shipped to a licensed burial site in accordance with Nuclear Regulatory Commission and Department of Transportation regulations.

Our review of the solid radwaste treatment system included the system's capability of processing the types and volumes of wastes expected during normal operation and anticipated operational occurrences in accordance with 10 CFR Parts 20 and 71 and 49 CFR Parts 170 through 178, and the quality group classification and seismic design criteria. We have also reviewed the provisions for onsite storage and the provision for controlling airborne dusts during waste compaction. The applicants' piping and instrumentation schematic diagrams and descriptive information were reviewed.

The basis for acceptance in our review has been conformance of the applicants' designs, design criteria and design bases for the solid radwaste system to the applicable regulations and guides, as referenced above, as well as to staff technical positions and industry standards.

Based on the foregoing evaluation, we conclude that the proposed solid radwaste system is acceptable.

11.5 Process and Effluent Radiological Monitoring System

The process and effluent radiological monitoring system will be designed to provide information concerning radioactivity levels in systems throughout the plant, indicate radioactive leakage between systems, monitor equipment performance, and monitor and control radioactivity levels in plant discharges to the environs. Both liquid and gaseous streams will be monitored. Table 11.2 indicates the proposed locations of continuous monitors. Monitors on effluent release lines will automatically terminate discharges should radiation levels exceed a predetermined value. Systems which are not amenable to continuous monitoring, or for which detailed isotopic analyses are required, will be periodically sampled and analyzed in the plant laboratory.

We have reviewed the locations and types of effluent and process monitoring to be provided. Based on the plant design and on the locations of continuous monitoring and intermittent sampling stations, we have concluded that all normal and potential release pathways will be monitored. We have also determined that the sampling and monitoring provisions will be adequate for detecting radioactive material leakage to

TABLE 11.2

PROPOSED LOCATIONS OF
PROCESS AND EFFLUENT MONITORING

Stream Monitored

Liquid*

Component Cooling Heat Exchanger Output
Steam Generator - Secondary Side
Steam Generator - Blowdown
Service Water Return Line
Boron Recycle System Distillate
Liquid Radwaste Release
Turbine Building Floor Drain

Gas**

Containment Purge Exhaust
Unit Vent
Condenser Air Discharge
Radwaste Building Exhaust

* All liquid streams will be monitored for gross gamma activity.

** All gas streams will be monitored for particulates (gross beta), iodine (gamma) and noble gas (beta).

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normally uncontaminated systems and for monitoring plant processes which affect radioactivity releases. On this basis we conclude that the monitoring and sampling provisions will meet the requirements of Criteria 13, 60 and 64 of the General Design Criteria and the guidelines of Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants", and are, therefore, acceptable.

Our review of the radiological monitoring systems includes the provision proposed for sampling and monitoring all station effluents in accordance with Criterion 64, for providing automatic termination of effluent releases and assuring control over discharges in accordance with Criterion 60 and Regulatory Guide 1.21, for sampling and monitoring plant waste process streams for process control in accordance with Criterion 13, for conducting sampling and analytical programs in accordance with the guidelines in Regulatory Guide 1.21, and for monitoring process and effluent streams during postulated accidents. The review included piping and instrument diagrams and process flow diagrams for the liquid, gaseous, and solid radwaste systems and ventilation systems, and the location of monitoring points relative to effluent release points on the site plot diagram.

The basis for acceptance in our review has been conformance of the applicants' design, design criteria and design bases for the process and effluent monitoring systems to the regulations and guides, as referenced above, as well as staff technical positions and industry standards.

Based on the foregoing evaluation, we conclude that the proposed provisions for monitoring process and effluent streams are acceptable.

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12.0 RADIATION PROTECTION

12.1 Summary Description

The radiation protection program for the Wolf Creek plant is described in Section 12.0 of the SNUPPS PSAR, with the exception of the health physics portion which is described in Section 12.0 of the Wolf Creek Site Addendum Report.

We have evaluated the adequacy of the proposed radiation protection program for the Wolf Creek plant. The objective of this evaluation was to determine whether occupational radiation exposures can be controlled within the guidelines of 10 CFR Parts 20 and 50 and if the as low as practicable objective, defined in Regulatory Guide 8.8, will be maintained.

Our review covered the applicants' radiation protection design features including shielding and layout of facility, the area monitoring program which details radiological and airborne radioactivity monitoring features, the ventilation systems which will be designed to provide a suitable radiological environment, and the health physics program.

The basis for acceptance of the radiation protection program for the Wolf Creek plant has been its conformance to the applicable regulations and guides, as well as to established criteria and industry standards. We conclude that the proposed policy, design and operational considerations relating to occupational exposures are acceptable.

12.2 Shielding

The facility shielding design was reviewed to assure that (1) exposure to operating personnel will meet the requirements of 10 CFR Part 20, and (2) radiation exposures to operating personnel during normal operation (including anticipated operational occurrences, refueling, maintenance, inservice inspections, etc.) will be maintained to as low as practicable.

To maintain radiation exposures to as low as practicable, the applicants' design classified all plant areas into radiation zones, based on expected frequency and duration of occupancy by operating personnel. The design of the radiation shielding will consider the dose rate criterion for each zone based on the radiation sources within the zone. All radioactive sources that would effect the shield design have been considered. Also, processing systems will be located to minimize shielding.

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Shielding analyses will be made using accepted criteria for codes, models and assumptions. Consistent with the design, the applicants have addressed the steps that will be taken to assure that low dose rate zones will not be compromised by inadvertent increases in radiation levels.

Consequently, pipes carrying radioactive liquids, including field run piping, filters, demineralizers, tanks, evaporators, pumps, and sampling areas will be designed to be located in properly shielded compartments. Access will be provided by using shielding blocks or labyrinths. Where tanks within compartments will contain significant quantities of radioactivity, they will be shielded from each other. Therefore, each component or tank within a compartment will be isolated to allow maintenance, inspection, and some non-routine operations with as low as practicable radiation interference from other components or tanks. In addition, shielded valve compartments with reach rods, temporary shielding for maintenance, removable sections of shield walls, and concrete plugs to replace worn-out equipment or spent filter cartridges, respectively, are among those devices that will be used to maintain exposures to as low as practicable.

During the construction phase, visual inspection will be made of the shielding to detect major defects. When the plant is in operation, gamma and neutron radiation surveys will be made to insure that no shielding defects or inadequacies are present that might affect personnel exposures during normal operation and maintenance.

As a result of our concerns during the review, the applicants modified the design to reduce radiation exposures to personnel by removing access to the fuel transfer tube from the piping penetration room since this access might have caused inadvertent exposure to personnel during fuel transfer. Also, the fuel pool skimmer filter was relocated from the fuel building to a concrete vault in the radwaste building.

On the basis of comparison of the applicants' design criteria, shield models and operating philosophy, with that of the staff's, we conclude that adequate consideration has been given to the shielding and layout of facilities and components to (1) maintain exposures to operating personnel within the applicable limits of 10 CFR Part 20 and, (2) reduce unnecessary exposure during normal operation of the facility, consistent with the guidelines of Regulatory Guide 8.8.

We, therefore, find the applicants' shielding program acceptable.

12.3 Area Monitoring

The radiological monitoring program will be designed to continuously measure the radiation levels in (1) controlled access areas (i.e., areas with a design dose rate for workers from less than 2.5 millirem per hour to 100 millirem per hour), (2) areas where radiation sources are stored, and (3) areas where radioactive material enters or leaves the plant. Each instrument of the system will have a sensor, whose dose

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rate output is recorded in the control room, and will have both an audible and visual alarm at the location of the instrument and in the control room. An instrument functional check will be performed monthly on each monitor from the control room, using a remotely operating check source. Secondary calibration techniques will be periodically performed using a National Bureau of Standards calibrated source.

In-plant continuous airborne radioactivity monitoring will be performed by eight fixed gas and particulate monitors located throughout the plant. In making the selection of the detector systems and their location, the applicants considered the nature and type of radioactive release so the appropriate responses could be taken as required. The monitors will be located in personnel operating areas and in ducts serving areas containing processes which, in the event of major leakage, could result in concentration within the plant.

Based on the proposed location of area monitors, their sensitivity and range, and their alarm annunciation and recording devices, we conclude that the scope of the area monitoring program will provide satisfactory radiological protection to in-plant personnel and, therefore, is acceptable.

12.4 Ventilation

The applicants' ventilation systems for restricted areas will be designed to provide suitable radiological environment for personnel and equipment, and to assure compliance with the limits set forth in 10 CFR Part 20. The path of the ventilation air will be from areas of low radioactivity toward areas of higher activity to prevent the spread of airborne radioactive material and, thereby, assure contamination control.

Access and service requirements for filter absorber units will comply with Sections 4(a) and 5(a) of Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants." Ventilation systems will be inspected to assure proper flow paths and exhaust vents.

Various compartments throughout the plant will be provided with high efficiency particulate air filter banks, with charcoal filters added at selected locations to preclude a buildup of airborne contamination.

We conclude that the design of the ventilation systems will be based on design criteria that provide reasonable assurance that the systems have the capability to maintain concentrations of airborne activity in areas normally occupied in accordance with 10 CFR Part 20 and, therefore, is acceptable.

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12.5 Health Physics Program

The applicants' stated health physics program for radiation protection is based on compliance with the requirements of appropriate Commission regulations. Consistent with this commitment, programs and procedures will be implemented conforming to the as low as practicable philosophy of Regulatory Guide 8.8, including personnel dosimetry by film badge or thermoluminescent dosimeters; protective clothing and respiratory protection, including a respiratory fitting and testing program; access control measures, such as checkpoint stations, locked and/or annunciated doors and gates, fences, alarms, rope barriers and signs, used to preclude unauthorized entry into high radiation or contamination areas; radiation work permits which will be issued based on identified radiation areas; portable radiation survey meters to assist in contamination control; and laboratory facilities and equipment for analyzing and evaluating the radiological status of the plant. As low as practicable inhalation doses will be maintained by implementing operating procedures that include good housekeeping practices (e.g., immediate clean-up of spills and repair of leaks) and collection of breathing zone samples to verify safe conditions.

Monitoring instrumentation, laboratory facilities and counting room equipment will be operated by the Health Physics staff. Self-reading dosimeters will be issued to individuals badged with thermoluminescent dosimeters (including visitors), whose work conditions make day-to-day indication of exposure desirable, and will be maintained by the Health Physics staff for recording daily exposures. Dosimeter records will furnish the exposure data for better administrative control of radiation exposure. A bioassay program consisting of urinalysis and wholebody counting will be performed on supervisory personnel and those personnel who work routinely in radiation areas and will provide supporting data on the effectiveness of the air monitoring program. Whole body counting will be used to quantitatively detect fractions of body burdens. Bioassay for tritium will be performed when the need for verification of tritium exposure, based on tritium monitoring, is required.

On the basis of the plant design criteria, health physics related equipment and procedures, the training and retraining for all plant personnel in radiation protection, and the applicants' compliance with Regulatory Guide 8.8, we conclude that the applicants' health physics program will provide plant personnel with adequate protection against the radiation hazards associated with the normal operation of the plant and will limit occupational exposures to as low as practicable in accordance with 10 CFR Part 20.

Therefore, we conclude that the program is acceptable.

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13.0 CONDUCT OF OPERATIONS

13.1 Organization and Qualifications

The Kansas Gas & Electric Company is responsible for the design, construction, testing and operation of the Wolf Creek plant. A technical representative from the Kansas Gas & Electric Company is a member of the SNUPPS Project Organization technical committee and is responsible for providing the technical interface between the Kansas Gas & Electric Company and the SNUPPS Project Organization on the standard portion of the Wolf Creek plant.

The Kansas Gas & Electric Company has established a Nuclear Development Department as the primary group to implement its responsibility for the design and construction of the Wolf Creek plant. The Superintendent of the Nuclear Development Department reports to the Assistant Vice-President, Operations. Quality Assurance aspects are discussed in Section 17.0 of this report.

The proposed station organization for the operation of the Wolf Creek plant will consist of a technical staff of approximately 65 persons under the direction of the Plant Superintendent. Reporting to the Plant Superintendent will be: an Assistant Plant Superintendent; a Plant Operations Supervisor responsible for the day-to-day operation of the plant with a staff of approximately 30 persons; a Technical Support Supervisor responsible for providing technical assistance for plant operations with a staff of approximately 17 persons; and a Maintenance Supervisor responsible for plant maintenance with a staff of approximately 13 persons. This is a conventional type of plant organization for plant operations. The shift crew composition will consist of six persons, two of whom will be licensed senior operators and one of whom will be a licensed operator. Technical support for the plant staff will be provided by the Superintendent of Nuclear Development and his staff.

The Kansas Gas & Electric Company has stated that plant personnel will meet the requirements set forth in ANSI N18.1, "Selection and Training of Personnel for Nuclear Power Plants." This complies with Regulatory Guide 1.8 and is, therefore, acceptable.

We conclude that the Kansas Gas & Electric Company has established an acceptable organization to implement its responsibilities relative to the design and construction of the Wolf Creek plant and that the proposed plant organization, proposed personel qualifications, and proposed plans for offsite technical support for plant operations are acceptable.

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13.2 Training Program

The overall conduct and administration of the plant training program is the responsibility of the Plant Superintendent. The Plant Training Coordinator may be delegated the responsibility of development and implementation of the program.

The Kansas Gas & Electric Company states that a training program will be established to provide plant personnel with sufficient knowledge and operating experience to start up, operate, and maintain the plant in a safe and efficient manner. The training program is to be developed by Kansas Gas & Electric Company with principal assistance from the Westinghouse training staff.

Training for the personnel to be licensed will include: basic nuclear training; research reactor training and operation; observation at an operating PWR; a plant system lecture series; simulator training and practical on-the-job training. Maintenance and technical staff personnel will receive specialized training in their particular fields. Station personnel will also receive training in security and emergency plans, administrative procedures and radiation protection, as appropriate.

We conclude that the proposed training program is acceptable.

13.3 Emergency Planning

The Kansas Gas & Electric Company has described its preliminary plans for coping with emergencies. Emergency and evacuation procedures will be developed to implement the emergency plan which will be submitted with the Final Safety Analysis Report. The onsite emergency effort will be under the direction of the Shift Supervisor and may be augmented as necessary by offsite personnel including various State and local agencies. The Kansas Gas & Electric Company has identified the notification responsibilities within the organization to ensure prompt and effective communication in the event of an emergency.

For assistance in dealing with emergencies, the Kansas Gas & Electric Company will establish arrangements with the Coffey County Sheriff, Coffey County Civil Defense Director, Coffey County Volunteer Fire Department, State Highway Patrol, State Department of Health and Environment, State Civil Defense, and the Federal Bureau of Investigation. The Adjutant General's Department, Civil Defense Division, has been identified as having the overall responsibility for emergency planning in the State of Kansas. The Kansas Gas & Electric Company has made initial contact with the Civil Defense, Radiation Safety Program Director for the purposes of developing coordinated emergency response plans for environs of the Wolf Creek plant.

In-plant monitors will provide input for determining the classification of radiological emergencies. In the event of a serious accident, the Kansas Gas & Electric Company intends to initiate a preplanned program of increased environmental sampling

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and a radiation survey of environs. In addition, a preplanned evacuation procedure will be established.

We have performed analyses and have confirmed the practicability of evacuation, as an emergency protective measure, in the environs of the proposed plant and have determined that appropriate criteria have been identified for the design of an acceptable plan. The Kansas Gas & Electric Company has indicated that the residents in the environs of the plant will be informed of the plan to be used in the event of an emergency. Onsite direction of the emergency effort will be assured from the control room which will be designed for continuous occupancy during the course of an accident and will be equipped with various decisional aids, protective equipment, and portable survey instruments. The Kansas Gas & Electric Company has also described the offsite assistance expected from the State and local agencies previously identified.

The plant emergency facilities will include first aid and decontamination facilities for the treatment of injured personnel. An emergency vehicle will be available onsite for the transportation of seriously injured personnel to offsite medical facilities. The Kansas Gas & Electric Company has made preliminary contacts with the University of Kansas Medical Center, the Stormont-Vail Hospital, and the Newman Memorial County Hospital for providing hospital facilities and services to individuals affected by radiological emergencies.

The Kansas Gas & Electric Company will conduct a training program covering emergency procedures for all plant employees and long-term contract workers. In addition, the Kansas Gas & Electric Company has indicated its willingness to participate in a coordinated effort with the State Civil Defense which has the responsibility for training personnel of the various agencies involved with emergency response plans.

The various plant features to assure evacuation capability, in addition to design features to permit safe station shutdown, will include radiation emergency alarms, site evacuation alarms, redundant communications systems, and an adequate site road network connected to good offsite roads. The Kansas Gas & Electric Company's capability for facility reentry will be supported by the establishment of an offsite assembly area equipped with protective clothing, radiation monitoring instruments, and communications equipment.

We conclude that the emergency planning program as presented by the Kansas Gas & Electric Company meets the requirements of Part II of Appendix E to 10 CFR Part 50, and is consistent with facility design features, analyses of postulated accidents, and characteristics of the proposed site location. Furthermore, it provides reasonable assurance that appropriate protective measures can be taken within and beyond the site boundary in the event of a serious accident.

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13.4 Plant Procedures and Records

The Kansas Gas & Electric Company states that plant procedures are to be performed in accordance with written and approved operating and emergency procedures. Areas that will be covered include normal startup, operation and shutdown, maintenance and abnormalities in operation. American National Standards Institute Standard ANSI N18.7, 1972 "Administrative Controls for Nuclear Power Plants," will be used as a guide in preparation of these procedures.

The Kansas Gas & Electric Company has also committed to keep plant records in conformance with ANSI N18.7 and Criterion XVII of Appendix B to 10 CFR Part 50. In addition, the preliminary plans for review and audit of plant operations generally meet the provisions of ANSI N18.7.

We conclude that the Kansas Gas & Electric Company's proposed program for preparation, review, approval and use of written procedures, and the commitment to document operating and maintenance activities are acceptable.

13.5 Industrial Security

The Kansas Gas & Electric Company has provided a general description of plans for protecting the plant against potential acts of industrial sabotage. Provisions for the screening of employees at the plant, and for design phase review of plant layout and protection of vital equipment have been described and conform to Regulatory Guide 1.17, "Protection of Nuclear Power Plants Against Industrial Sabotage."

We conclude that the Kansas Gas & Electric Company's preliminary plans and arrangements for protection of the plant against acts of industrial sabotage are acceptable. The Kansas Gas & Electric Company will provide additional details of these plans for our review at the operating license stage of review.

14.0 INITIAL TESTS AND OPERATION

The scope and objectives of the initial test program for the Wolf Creek plant are described in Section 14.0 of the SNUPPS PSAR, which incorporates Section 14.1 of RESAR-3, Consolidated Version. The administrative procedures for this program are discussed in Section 14.0 of the Wolf Creek Site Addendum Report.

The initial test program for the Wolf Creek plant will be conducted by the Kansas Gas & Electric Company who will receive technical support from the nuclear steam supply system vendor (Westinghouse Electric Corporation), the architect-engineers (Bechtel Power Corporation and Sargent and Lundy Engineers), and the construction contractor. The applicants have committed to developing and executing the test program in accordance with Regulatory Guide 1.68, "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors."

A comprehensive testing program will be established to assure that equipment and systems perform in accordance with design criteria prior to fuel loading. As the installation of individual components and systems will be completed, they are to be tested and evaluated according to predetermined and approved written testing techniques, procedures, or check-off lists. Analyses of test results will be made to verify that systems and components are performing satisfactorily and if not, to provide a basis for recommended corrective action. The program includes tests, adjustments, calibrations, and system operations necessary to assure that initial fuel loading, initial criticality and subsequent power operation can be safely undertaken.

In general, preoperational testing will be completed prior to fuel loading. As the construction of individual systems are completed, preoperational tests are performed to verify, as nearly as possible, the performance of the system under actual operating conditions. Fuel loading begins when all prerequisite system tests and operations are satisfactorily completed. The purpose of this phase of activities will be to prepare the system for nuclear operation and to establish that all design requirements necessary for operation are achieved. A testing sequence for preoperational tests and for startup tests will be established.

On the basis of our review, we conclude that an acceptable test and startup program can and will be implemented by the applicants. The applicants will provide additional details of this program for our review at the operating license stage of review.

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15.0 ACCIDENT ANALYSES

15.1 General

We have evaluated the responses of the facility to various potential accidents and their consequences. We and the applicants have considered a full spectrum of plant conditions, that for evaluation purposes, are divided into four categories in accordance with their anticipated frequency of occurrence and risk to the public. The four categories of events evaluated in the PSAR are:

(1) Condition I - Normal Operation and Operational Transients

(2) Condition II - Faults of Moderate Frequency

(3) Condition III - Infrequent Faults

(4) Condition IV - Limiting Faults

All transients and accidents have been evaluated at the core design power level of 3411 megawatts thermal, with the exception of the loss-of-coolant and the fuel handling accidents. The latter two accidents were evaluated at 3636 megawatts thermal, about two percent above the ultimate thermal design capability of 3579 megawatts.

Condition I events are those which are expected to occur during the course of normal power operation, refueling or maintenance. Condition I occurrences are accommodated by sufficient design margins between any plant parameter and the value of that parameter which would require actuation of the reactor protection system. Condition I events are controlled by reactor systems that will automatically maintain prescribed conditions in the plant, even under the most conservative set of reactivity parameters with respect to both system stability and transient performance.

Condition II faults, at worst, result in a reactor trip with the plant being capable of return to operation. Condition II events do not propagate to a more serious Condition III or IV event and are not expected to result in fuel rod failure or reactor coolant system overpressurization. The criterion adopted to assure that the reactor coolant system pressure boundary integrity will be maintained is that the system pressure will remain below the code pressure limits set forth in Section III of the ASME Boiler and Pressure Vessel Code (i.e., 110 percent of reactor coolant system pressure or 2750 pounds per square inch absolute). The criterion adopted to assure that no fuel damage will occur is that the minimum departure from nucleate boiling ratio must satisfy the the 95/95 criterion. (The 95/95 criterion provides a

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95 percent confidence that 95 percent of the hot fuel rods in the core will not experience departure from nucleate boiling.

Condition III events are very infrequent faults that may in some cases result in the failure of a small fraction of the fuel rods; however, sufficient fuel damage could result precluding immediate resumption of operation. A Condition III event will not (1) generate a Condition IV fault, (2) result in a loss of function of the reactor coolant system, thereby assuring that the core geometry remains amenable to cooling, or (3) result in a release of radioactivity in sufficient quantities to interrupt or restrict public use of those areas beyond the exclusion radius.

Condition IV events are limiting design bases that are not expected to occur, but are postulated because their consequences include the potential for the release of significant quantities of radioactive material. System design bases in consideration of Condition IV faults preclude a fission product release to the environment which would result in exceeding the limits established in 10 CFR Part 100. A Condition IV event will not result in the loss of the required functions of systems necessary to mitigate the consequences of the accident, such as the emergency core cooling system and the containment.

Table 15.1 of this report is a listing of typical events in each category.

15.2 Abnormal Transients

The applicants have submitted analyses of abnormal transients and have shown that the integrity of the reactor coolant system pressure boundary will be maintained and that the minimum departure from nucleate boiling ratio satisifies the 95/95 criterion. The maximum pressure transient has been identified as the loss of external electrical load and/or turbine trip from maximum power conditions (102 percent power) and would result in a peak pressure of 2550 pounds per square inch absolute.

A minimum departure from nucleate boiling ratio of 1.35 was identified for the uncontrolled rod withdrawal with the reactor operating at full power. This satisfies the 95/95 criterion. In the analyses of Condition II events, the departure from nucleate boiling calculations were performed using a 0.86 critical heat flux multiplier, which provided a 14 percent design margin. Recent departure from nucleate boiling tests reported in Westinghouse Topical Report WCAP-8296, "Effect of 17 x 17 Fuel Geometry on DNB" have confirmed the conservatism of this multiplier.

To assure that the departure from nucleate boiling ratio remains above 1.30 for any combination of system parameters, the following power and temperature reactor trips will be provided to prevent the design thermal and flow conditions from being exceeded: (1) power range high neutron flux, (2) high pressure, (3) low pressure, (4) overpower, and (5) overtemperature. In addition, departure from nucleate boiling protection against flow transients will be provided by the low reactor coolant flow

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TABLE 15.1

CATEGORIES OF TYPICAL TRANSIENTS AND FAULTS

Condition I

- Reactor startup
- Reactor shutdown
- Refueling operations

Condition II

- Control rod assembly withdrawal while the reactor is subcritical
- Partial loss of forced reactor coolant flow
- Startup of an inactive reactor coolant loop
- Turbine trip
- Loss of normal feedwater
- Loss of offsite power

Condition III

- Improper loading of a fuel assembly
- Complete loss of forced reactor coolant flow
- Minor secondary system pipe break
- Control rod assembly withdrawal at full power
- Waste gas decay tank rupture

Condition IV

- Control rod ejection
- Fuel handling accident
- Steam generator tube rupture
- Major secondary system pipe rupture
- Reactor coolant system rupture (LOCA)
- Single reactor coolant pump locked rotor

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and pump undervoltage reactor trips. It should be noted that the complete loss of forced reactor coolant flow, while listed as a Condition III event, is also in compliance with the 95/95 criterion, as required by the staff for Condition II transients.

Our evaluation of abnormal transients has indicated that the transients presented do not lead to unacceptable consequences.

15.3 Accidents

The applicants have evaluated a broad spectrum of accidents that might result from postulated failures or improper operation of equipment. These highly unlikely accidents have been analyzed in detail and are representative of the spectrum of types and physical locations of postulated events involving the various engineered safety feature systems. In addition, as required by WASH-1270 and as discussed in Section 7.2 of this report, the applicants submitted an evaluation of anticipated transients without scram, including the design changes necessary to mitigate the consequences of anticipated transients for which credit for reactor trip is not permitted.

The locked rotor accident was analyzed by postulating an instantaneous seizure of one reactor coolant pump rotor. The reactor flow would decrease rapidly and a reactor trip would occur as a result of a low flow signal. A thermal analysis of the hot rod in the core was performed and revealed that approximately three percent of the rods would experience departure from nucleate boiling. The applicants concluded that there would be no gross fuel cladding failure since the maximum cladding temperature was calculated to be 1837 degrees Fahrenheit. Corresponding to this cladding temperature, the peak reactor coolant system pressure during this accident will not exceed the ASME Code pressure limits for the reactor pressure boundary, and therefore no accidental release of radioactivity will occur. On the basis of our review, we have concluded that the consequences of this postulated accident are acceptable.

Rod cluster control assembly misalignment accidents, including a dropped full length rod cluster control assembly, a dropped full length rod cluster control assembly bank, and a misaligned full or part length rod cluster control assembly have been analyzed by the applicants. The analyses were performed using the TURTLE computer code to determine X-Y peaking factors. The THINC IV computer code was then used to calculate the departure from nucleate boiling ratio. For the transient response to a dropped rod cluster control assembly or rod cluster control assembly bank, the LOFTRAN computer code is used.

In the event of a dropped rod cluster control assembly, the automatic controller may return the reactor to full power. Analysis indicates that a departure from nucleate boiling ratio of less than 1.30 will not occur during this event. For the case of dropped rod cluster control assembly groups, the reactor is tripped by the power

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range negative neutron flux trip and the reactor is shut down without core damage. For cases where a rod cluster control assembly group is inserted to its insertion limit with a single rod cluster control assembly in the group fully withdrawn, analysis indicates that the departure from nucleate boiling ratio is greater than 1.30.

Misaligned rods are detectable by: (1) asymmetric power distributions sensed by excore nuclear instrumentation or core exit thermocouples, (2) rod deviation alarms, and (3) rod position indicators. A deviation of a rod from its bank by 14.4 inches or twice the resolution of the rod position indicator, will not cause power distributions to exceed design limits. Additional surveillance will be required to assure rod alignment if one or more rod position channels are out of service.

We conclude that the applicants' provisions for mitigating a rod cluster control assembly misalignment accident assure that the reactor can be safely shut down and that the departure from nucleate boiling ratio will not decrease below 1.30.

The consequences of the inadvertent loading of a fuel assembly into an improper position were analyzed by the applicants. Comparisons of calculations of the power distributions for the normal fuel loading pattern and five cases of fuel assembly and burnable poison misloadings were presented. These represent the spectrum of probable inadvertent improper loadings. With the exception of the case in which, near the center of the core, a burnable poison rod is located in the correct Region 2 position but in a Region 1 assembly mistakenly loaded into the Region 2 position, the resultant distortion of the power distribution would be detectable by the instrumentation provided. In the case described, the distortion of power density (Fq) would be approximately the uncertainty in the measurement of the peak local power density and hence would cause no safety problems.

Incore instrumentation using movable fission chamber detectors is provided to detect loading mistakes. A power distribution measurement using this system will be required by the technical specifications to determine if misloadings exist. Thermocouples in approximately one-third of the fuel assemblies would also provide an indication of a loading mistake. In most cases, an improperly loaded fuel assembly would cause a quadrant power tilt that would be detected also by the excore nuclear instrumentation. In addition to the instrumentation system to detect misloadings, strict administrative controls are provided to prevent such an event.

We conclude that an improperly loaded fuel assembly or burnable poison cluster that would cause a significant safety problem could be detected with the instrumentation provided.

The rod cluster control assembly ejection accident was analyzed by the applicants. The mechanical failure of a control rod mechanism pressure housing would result in

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the ejection of a rod cluster control assembly. The consequences of this would be a rapid reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage.

Although mechanical provisions have been made to make this accident extremely unlikely, the applicants have analyzed the consequences of such an event. Methods used in the analysis are reported in Westinghouse Topical Report WCAP-7588, Revision 1, "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics," which we have reviewed and accepted, as documented in the Commission's letter, dated August 28, 1973. This report demonstrates that the model used in the accident analysis is conservative relative to three-dimensional kinetics.

The ejected rod worths and reactivity coefficients used in the analysis have been reviewed and are acceptable.

The applicants' criteria for gross damage of fuel are a clad temperature of 2700 degrees Fahrenheit and an energy deposition of 200 calories per gram. Four cases of rod ejection were analyzed: beginning-of-life at 102 percent and zero power, and end-of-life at 102 percent and zero power. The worst case was the end-of-life 102 percent power case, which resulted in a clad temperature of 2565 degrees Fahrenheit and a 181 calorie per gram energy deposition. Therefore, gross fuel damage will not occur in a rod cluster control assembly ejection accident. We conclude that the assumptions and methods of analysis used by the applicants are in accordance with Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," and are, therefore, acceptable.

Pursuant to the final acceptance criteria for emergency core cooling systems published in the Federal Register on January 4, 1974 (Appendix K to 10 CFR Part 50), the applicants are required to submit a loss-of-coolant accident analysis satisfying the requirements of the new criteria. The applicants have committed to submit the analysis in conformance with the new criteria. The applicants have already submitted a partial analysis in Revision 9 to the SNUPPS PSAR. The applicants are scheduled to submit additional information in September 1975, to complete the response. We will review the information submitted and to be submitted and report our evaluation as stated in Section 6.3.3 of this report.

15.4 Radiological Consequences of Accidents

The postulated design basis accidents analyzed by the applicants for offsite radiological consequences are the same as those analyzed for previously reviewed and approved pressurized water reactor facilities. These include a loss-of-coolant accident, a steam line break accident, a steam generator tube rupture, a fuel handling accident, a rupture of a radioactive gas storage tank in the gaseous radioactive waste treatment system, and a control rod ejection accident.

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We have reviewed these accidents and have further evaluated the radiological consequences of the loss-of-coolant accident as discussed in Section 15.5 of this report, and of the fuel handling accident, as discussed in Section 15.6 of this report.

On the basis of our experience with the evaluation of the steam line break and the steam generator tube rupture accidents for pressurized water reactor plants of similar design, we have concluded that the radiological consequences of these accidents can be controlled by limiting the permissible reactor coolant system and secondary coolant system radioactivity concentrations. At the operating license stage of review, we will include limits in the technical specifications on the reactor coolant system and secondary coolant system activity concentrations such that the potential two-hour doses at the exclusion radius, as calculated by the staff for these accidents, will be small fractions of the guideline doses of 10 CFR Part 100. Similarly, we will include limits in the technical specifications on gas decay tank activity such that any single failure (such as a relief valve lifting and sticking open) would not result in doses that are more than a small fraction of the 10 CFR Part 100 guideline values.

The radiological consequences of the control rod ejection accident will be evaluated at the operating license stage of our review. A technical specification will limit the allowable operational leakage of reactor coolant into the steam generator secondary side to assure that the radiological consequences of this accident will be well within the dose guidelines of 10 CFR Part 100.

15.5 Loss-of-Coolant Accident Dose Model

The facility will utilize a pressurized water reactor with a low leakage reactor containment building. The reactor building spray systems will be equipped with a sodium hydroxide additive injection system. The purpose of the additive is to increase the iodine removal capability of the spray following the postulated design basis loss-of-coolant accident. Section 6.2.2 of this report discusses the operation of the containment spray system.

We have evaluated the radiological consequences of a postulated loss-of-coolant accident. The assumptions used in the calculations are given in Table 15.2. The resultant loss-of-coolant accident dose values are given in Table 15.5 and include credit for iodine removal by the containment sprays.

The applicants will provide hydrogen recombiners for controlling any formation of hydrogen after a design basis loss-of-coolant accident. The applicants will also provide a backup purging mode. We have evaluated the additional dose an individual at the low population zone boundary might receive due to purging the containment after the design basis accident. Our assumptions for this analysis are listed in Table 15.4 and the calculated doses are listed in Table 15.5. We find that the calculated low population zone doses from purging, when added to the loss-of-coolant accident doses, are well within the guidelines of 10 CFR Part 100.

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As part of the evaluation of the loss-of-coolant accident, we have included the consequences of leakage of recirculating emergency core cooling water, containing radioactive fission products in amounts consistent with the source term assumptions of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss of Coolant Accident." After the loss-of-coolant accident, this water is circulated outside of the containment to the auxiliary building to be cooled. If a source of leakage should develop, such as from a pump seal, we believe a portion of the iodine would become gaseous and would exit to the outside atmosphere. The offsite doses resulting from such a sequence of events depends upon the temperature and magnitude of the assumed leakage and the site meteorology. If the leakage occurred when the water temperature was below 212 degrees Fahrenheit, a leak rate of about 30 gallons per minute, over a period of 30 minutes, would result in doses (without filters) which could exceed the guideline values of 10 CFR Part 100 (for a relative concentration of 1.9×10^{-4} seconds per cubic meter). If the leakage occurred when the fluid is near its peak temperature, then part of the leaking water would flash to steam, leading to additional iodine release. In this case, about ten gallons per minute leakage for 30 minutes (for the same relative concentration) would result in doses, without filters, which could exceed 10 CFR Part 100 guidelines from this source alone.

If the emergency core cooling system equipment area is served by filters effective in removing iodine, the offsite doses from possible pump leakage in this area would be within the guidelines of 10 CFR Part 100, even for substantial amounts of leakage. As a result of the analysis discussed above, we required that the emergency core cooling system equipment and piping areas outside containment be served by filters which are effective in removing iodine and which conform to the requirements of engineered safety features systems. Our position is further discussed in Section 6.2.3 of this report.

15.6 Fuel Handling Accident

For the analysis of the fuel handling accident, we have assumed that a fuel assembly was dropped in the fuel pool during refueling operations and that all of the fuel rods in the assembly were damaged, thereby releasing the volatile fission gases from the fuel rod gap into the pool. The radioactive material that escaped from the fuel pool was assumed to be released to the environment over a two-hour time period with the iodine activity reduced by filtration through the fuel building exhaust system. The assumptions and parameters used in the analysis are shown in Table 15.4 and the dose results are given in Table 15.5. The dose model and dose conversion factors employed in the analysis were the same as those given in Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors." We find that the resultant doses are well within the guidelines of 10 CFR Part 100.

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TABLE 15.2

ASSUMPTIONS USED FOR CALCULATION OF

LOSS-OF-COOLANT ACCIDENT DOSES

Power Level (megawatts thermal):	3636
Operating Time (years):	3
Fraction of Core Inventory Available for Leakage (percent):	
Iodines	25
Noble Gases	100
Initial Iodine Composition in Containment (percent):	
Elemental	91
Organic	4
Pariculate	5
Containment Leak Rate (percent per day):	
0-24 hours	0.1
> 24 hours	0.05
Containment Volume (cubic feet):	
Total Volume	2.5×10^6
Unsprayed Fraction (percent)	20
Containment Mixing Rate between Sprayed and Unsprayed Volume (cubic feet per minute):	20,000
Containment Spray System:	
Maximum Elemental Iodine Decontamination Factor	100
Removal Coefficients (per hour):	
Elemental Iodine	10
Particulate Iodine	0.45
Organic Iodine	0
Relative Concentration Values (seconds per cubic meter):	
0-2 hours	1.9×10^{-4}
0-8 hours	2.7×10^{-5}
8-24 hours	1.8×10^{-5}
1-4 days	7.4×10^{-6}
4-30 days	2.0×10^{-6}

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TABLE 15.3

ASSUMPTIONS USED IN THE ANALYSIS OF

FUEL HANDLING ACCIDENT DOSES

Power Level (megawatts thermal):	3636
Number of Fuel Rods Damaged:	264
Total Number of Fuel Rods in Core:	50,952
Radial Peaking Factor of Damaged Rods:	1.65
Shutdown Time (hours):	100
Inventory Released from Damaged Rods (Iodines and Noble Gases) (percent):	10
Pool Decontamination Factors	
Iodines	100
Noble Gases	1
Iodine Fractions Released From Pool:	
Elemental (percent)	75
Organic (percent)	25
Filter Efficiencies for Iodine Removal:	
Elemental (percent)	95
Organic (percent)	95
0-2 Hours Relative Concentration Value (seconds per cubic meter) at Exclusion Radius	1.9×10^{-4}

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TABLE 15.4

ASSUMPTIONS USED IN THE ANALYSIS OF

HYDROGEN PURGE DOSES

Power Level (megawatts thermal)	3636
Containment Volume (cubic feet)	2.5×10^6
Holdup Time in Containment Prior to Purge Initiation (days)	9
Purge Duration (days)	30
Purge Rate (standard cubic feet per minute)	52
Dose Reduction Factors	
Containment Sprays	20
Purge Filters	10
4-30 Day Relative Concentration Value (seconds per cubic meter) at Low Population Zone Boundary	2.0×10^{-6}

TABLE 15.5

RADIOLOGICAL CONSEQUENCES OF ACCIDENTS (REM)

ACCIDENT	EXCLUSION AREA (1200 Meters)		LOW POPULATION ZONE (4023 meters)	
	Thyroid	Whole Body	Thyroid	Whole Body
Loss-of-Coolant	61	3	36	1
Fuel Handling	2.8	0.9	---	---
Hydrogen Purge Dose	---	---	9	1.2

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16.0 TECHNICAL SPECIFICATIONS

The Technical Specifications in an operating license define certain features, characteristics, and conditions governing operation of a facility that cannot be changed without prior approval of the Commission. Final Technical Specifications will be developed and evaluated at the operating license stage. However, in accordance with Section 50.32 of 10 CFR Part 50, an application for a construction permit is required to include preliminary Technical Specifications. The regulations require an identification and justification for the selection of those variables, conditions, or other items which are determined as a result of the preliminary safety analysis and evaluation to be probable subjects of Technical Specifications for the facility, with special attention given for those items which may significantly influence the final design.

We have reviewed the proposed Technical Specifications presented in Section 16.0 of the PSAR with the objective of identifying those items that would require special attention at the construction permit stage, to preclude the necessity for any significant change in design to support the final Technical Specifications. The proposed Technical Specifications are similar to those being developed or in use for plants of a similar design to the Wolf Creek plant. We have not identified any items which require special attention at this stage of our review.

On this basis we have concluded that the proposed Technical Specifications are acceptable.

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17.0 QUALITY ASSURANCE

17.1 General

The description of the Quality Assurance (QA) Program for the Wolf Creek plant is contained in Section 17.0 of the PSAR. Section 17.0 of the SNUPPS PSAR includes the Q Programs for the SNUPPS Project Organization, the Bechtel Power Corporation (architect-engineer for the standard portion of the plant) and the Westinghouse Electric Corporation (nuclear steam supply system supplier). Section 17.0 of the Wolf Creek Site Addendum Report includes the QA Programs for the Kansas Gas and Electric Company (operating utility for the plant), the Daniel International Corporation (formerly Daniel Construction Company) (construction contractor for the plant), and Sargent & Lundy (architect-engineer for the site).

Our evaluation of the description of the QA Program is based on our review of the information presented in the PSAR and detailed discussions with the applicants and the SNUPPS Project Organization to determine if they and the principal contractors comply with the requirements of Appendix B to 10 CFR Part 50.

17.2 SNUPPS Project Organization

The five SNUPPS utilities have established a project organization, with participation by each utility, to manage the design and procurement of the standard portion of the four SNUPPS plants by use of common QA requirements and programs.

The SNUPPS Project Organization is administered by a Management Committee established by the SNUPPS utilities and is managed by an Executive Director who acts for the SNUPPS utilities. The project direction and organization are shown in Figure 17.1.

The SNUPPS QA Committee shown in Figure 17.1 consists of one QA representative from each SNUPPS utility. The functions of the QA Committee are to develop the QA manual of procedures, to review and approve Bechtel and Westinghouse QA Programs and to verify their adequacy for the project, to provide formal audits of the SNUPPS Project Organization, and to evaluate the effectiveness of the QA Program implementation for the Management Committee. Each member of the QA Committee is under administrative control by his respective utility.

The Executive Director is responsible to the Management Committee for the review and direction of the Bechtel activities. He also coordinates the interface requirements between each SNUPPS utility, Bechtel, and Westinghouse. He is responsible for the implementation of the QA Program through the QA Manager.

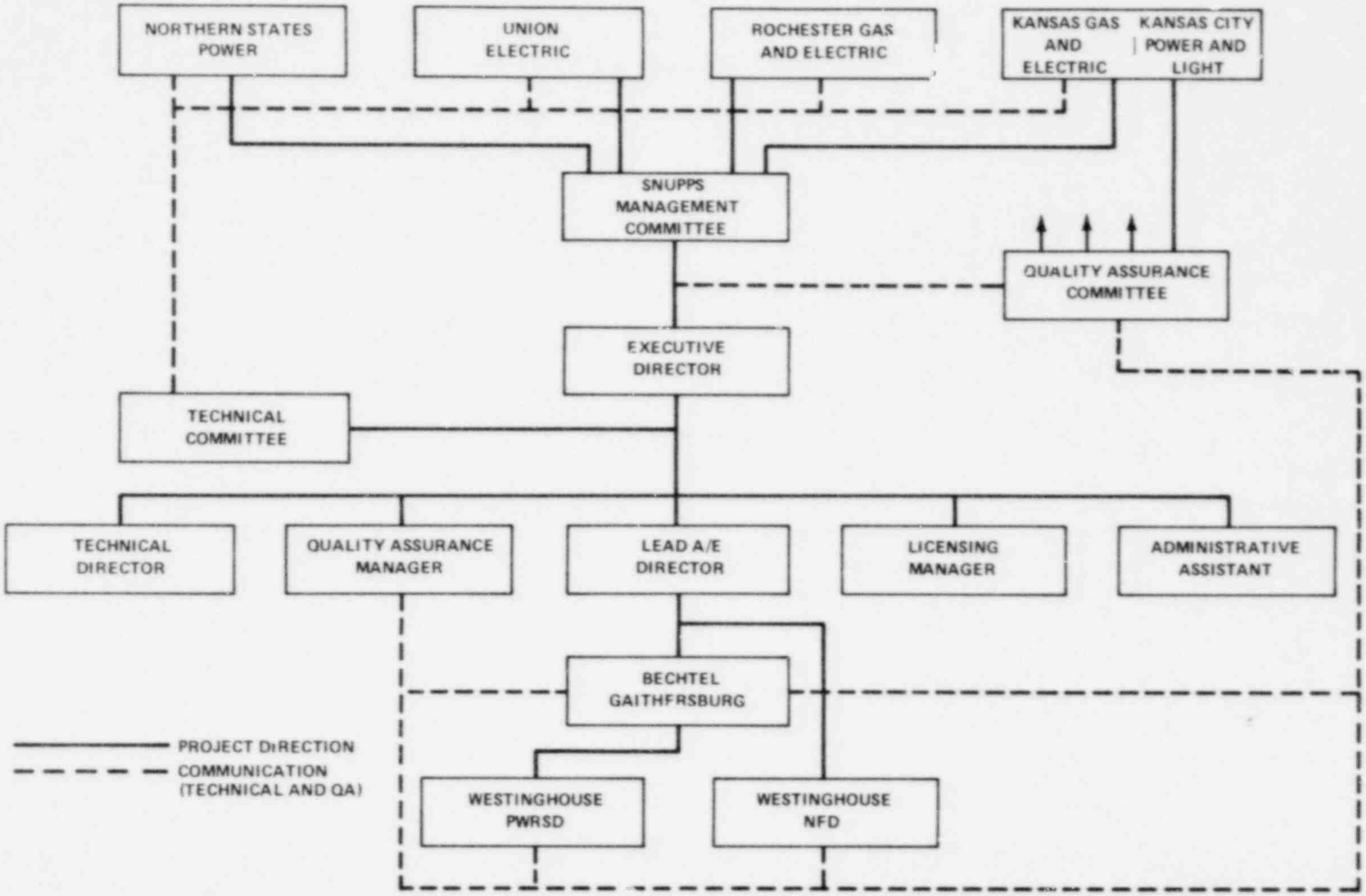

FIGURE 17.1 SNUPPS PROJECT ORGANIZATION

The QA Manager implements the SNUPPS Project Organization QA Program through his staff. He provides surveillance and audits of the SNUPPS Project Organization staff and of Bechtel activities and participates in Westinghouse audits conducted by Bechtel. We find that the organizational level of the QA Manager provides him with adequate independence and that he reports to a sufficiently high management level to accomplish his objectives.

Each member of the QA Committee can initiate stop work action, regarding activities managed by the SNUPPS Project Organization, through the Executive Director. The QA Manager has stop work authority over the SNUPPS Project Organization staff and activities and can initiate stop work action for Bechtel activities through the Executive Director.

We find that the description for implementing the SNUPPS Project Organization QA Programs, with utility management involvement and authority to enforce QA requirements, including stop work authority, is acceptable.

Our evaluation of the QA organizations for the SNUPPS Project Organization (QA Committee and QA Manager) is that they are independent of the organizations whose work they assure; they have clearly defined authorities and responsibilities; they have adequately defined personnel qualifications; they are organized so that they can identify QA problems in the SNUPPS Project Organization as well as in the principal contractors organizations; they can initiate, recommend or provide solutions; and they can verify implementation of solutions. We therefore conclude that the QA organization for the SNUPPS Project Organization complies with Appendix B of 10 CFR Part 50 and is acceptable.

The original QA Program description in the PSAR for the SNUPPS Project Organization did not provide enough detail to adequately describe the QA Program for design, procurement, and construction of the standard portion of the SNUPPS plants. In response to our request for a more detailed comprehensive description of the QA Program, Section 17.0 of the PSAR has been revised.

A matrix which relates the procedures of the various manuals to the applicable QA criteria of Appendix B to 10 CFR Part 50 is given in the PSAR. Based on our review of this matrix, we conclude that each criterion has been specifically included in written procedures within the SNUPPS Project Organization QA Program. The structures, systems, and components comprising the safety items subject to this program have been identified in the PSAR.

The QA Program for the SNUPPS Project Organization has been developed to comply with the applicable Regulatory Guides and industry standards including the Commission staff comments provided in Section D of "Guidance on Quality Assurance Requirements During the Construction Phase of Nuclear Power Plants," May 10, 1974 (WASH 1309) and in "Guidance on Quality Assurance Requirements During Design and Procurement Phase of Nuclear Power Plants," May 24, 1974 (WASH 1283-Revision 1).

The SNUPPS Project Organization will assure that its principal contractors and subcontractors have adequate QA Programs, that inspections will be performed to document inspection instructions by qualified personnel, and that results will be recorded. The SNUPPS Project Organization will assure by audits that personnel performing inspections are free from undue cost and schedule pressures of the project.

The SNUPPS Project Organization has established program requirements on itself and on its contractors which assure there will be a documented system of records attesting to quality.

A system of planned and documented audits, described in the PSAR, will be used by the SNUPPS Project Organization to verify compliance with the requirements of the QA Program and to assess its effectiveness. Audit results will be reviewed and corrective action taken by responsible management. Follow-up action is taken to assure corrective action. We find the SNUPPS Project Organization audit commitments acceptable.

Based on our review of Section 17.0 of the PSAR, we find for the SNUPPS Project Organization QA Program that there are adequate and well defined procedures, a commitment to the Commission's QA guidance, assurance of an independent inspection program, a documented system of records attesting to quality, an audit system to inform management of the effectiveness of the QA Program, and satisfactory management assessment of the QA Program.

We conclude that the SNUPPS Project Organization QA Program for the standard portion of the SNUPPS plants includes an acceptable QA organization with adequate policies, procedures, and instructions to implement a program that satisfies the requirements of Appendix B to 10 CFR Part 50.

17.3 Bechtel Power Corporation

Bechtel has been designated the architect-engineer by the SNUPPS utilities, to be responsible for the design, engineering, and procurement of equipment and materials for the standard portion of the SNUPPS plants, excluding the nuclear steam supply systems. Bechtel is also responsible for vendor surveillance, for witnessing selected tests during manufacturing, and for conducting audits of Westinghouse supplied equipment.

Figure 17.2 shows the Bechtel organization for the standard portion of the SNUPPS plants as it relates to engineering, procurement, construction, and QA. The Manager of SNUPPS Project reports through the Manager Division Project Operations to the Vice President and Division Manager of the Gaithersburg Power Division. The QA Manager reports directly to the Vice President and Division Manager as do the Division Managers of Project Operations, Construction, Engineering, and Procurement. We find the independence of the QA organization acceptable.

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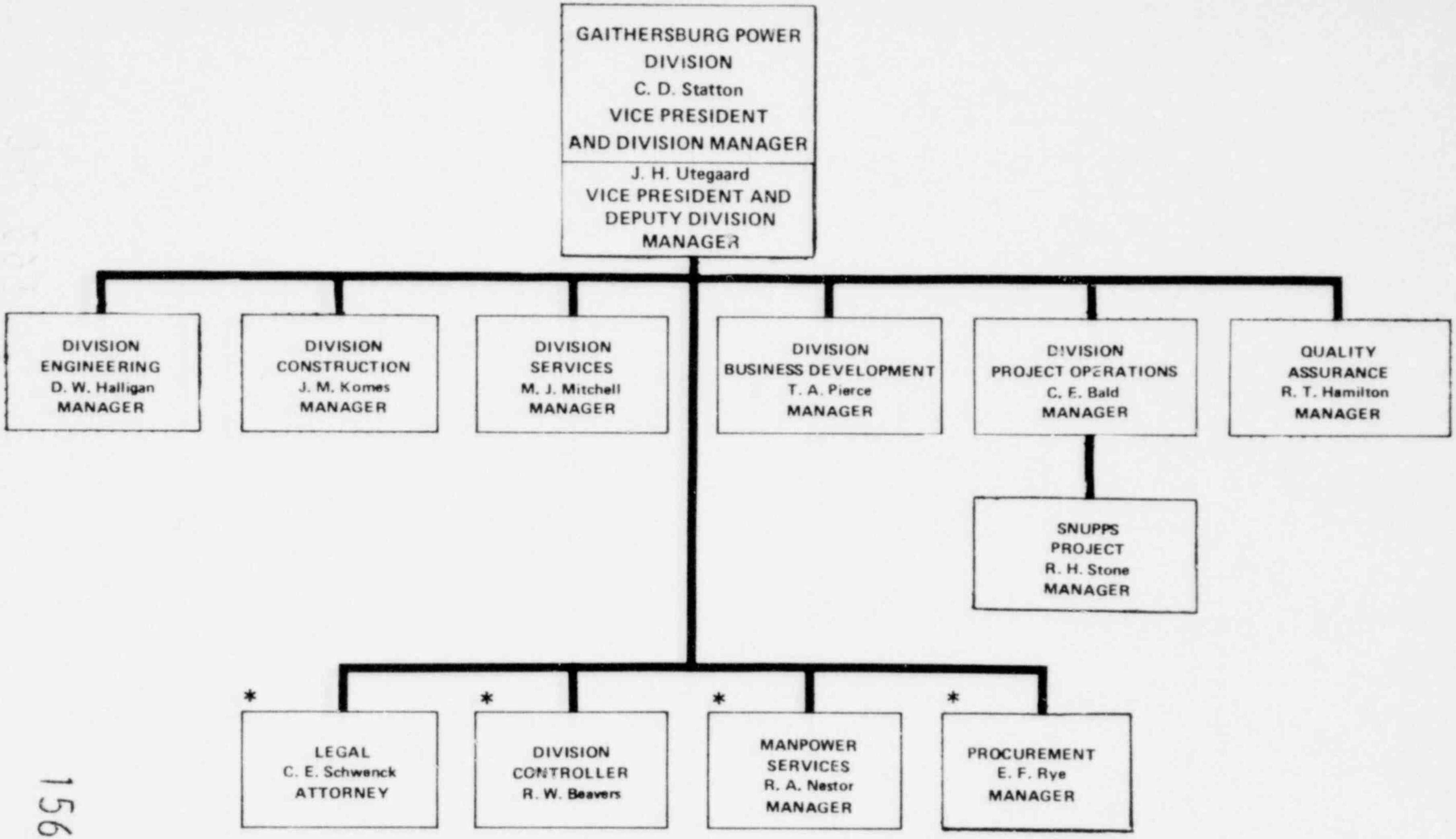

Figure 17.2 THE BECHTEL ORGANIZATION GAITHERSBURG POWER DIVISION

*RECEIVES FUNCTIONAL GUIDANCE FROM SAN FRANCISCO HOME OFFICE

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The Executive Vice President and General Manager of the Bechtel Thermal Power Organization, of which the Gaithersburg Power Division is a part, has issued a management statement of policy which requires mandatory implementation of the QA Program.

The Gaithersburg Division Manager is responsible for the Bechtel QA Program as it applies to the SNUPPS plants. The QA Manager is responsible for execution of the QA Program, for approval of QA procedures and instructions for the QA organization, and for review and concurrence, prior to issue, of the QA provisions contained in quality control procedures and instructions prepared by other departments in the division.

Quality verification activities, such as surveillance and audits, are conducted by the QA organization. Quality verification activities, such as procurement inspection, are conducted by the inspection organization in Procurement.

In reply to our concern about independence of the inspection personnel in Procurement, Bechtel has modified the duties and responsibilities of the QA organization to include review and approval of inspection procedures and instructions. We find that this check, by an organization outside of Procurement, provides additional independence for the inspectors from undue pressures of cost and schedules.

The QA Program for Bechtel has been developed to comply with the applicable Regulatory Guides and industry standards including the Commission staff comments provided in Section D of "Guidance on Quality Assurance Requirements During the Construction Phase of Nuclear Power PLants," May 10, 1974 (WASH-1309) and to "Guidance on Quality Assurance Requirements During Design and Procurement Phase of Nuclear Power Plants," May 24, 1974 (WASH 1283-Revision 1).

The QA policies, procedures, and instructions for the Bechtel QA Program are documented in the QA manual, the procurement inspection department manual, the engineering procedures manual, the Quality Control manual - ASME nuclear components, and the construction procedures manual. Bechtel has provided a cross index of Bechtel's QA procedures and the related criteria of Appendix B to 10 CFR Part 50. Based on our review of this information, we conclude that implementation of each criterion of Appendix B to 10 CFR Part 50 has been included within Bechtel's documented QA policies, procedures, and instructions for the SNUPPS Project. Bechtel has identified in the PSAR the safety related structures, systems, and components that are subject to the Bechtel QA program for the SNUPPS plants.

Bechtel has described a training and indoctrination program for its personnel. This program covers indoctrination and training in standards, policies, and procedures covering specific areas of work; qualification of inspection, examination and testing personnel; indoctrination in procurement inspection requirements; training and qualification of audit personnel; and qualification of personnel to code requirements for pressure boundary and structure welding and nondestructive test. We find the program acceptable.

Design documents are prepared by Project Engineering personnel and are verified or checked in accordance with engineering procedures. These checks are performed by personnel other than those who performed the original design but who have adequate technical capabilities for checking the work. We find the Bechtel description for design control adequate.

A comprehensive program of audits is described in the PSAR which covers the various activities of the Bechtel QA Program. Activities covered by planned audits include project engineering, field construction, nuclear steam supply system supplier's engineering and manufacturing activities, project engineering design, procurement activities, construction activities, and quality control activities at the jobsite.

Management reviews of the status and adequacy of the QA Program are accomplished by management outside of the QA organization through review of audit reports and periodic status reports of the Division QA Manager.

In our review, we have evaluated the Bechtel QA Program for compliance with the Commission's regulations and applicable Regulatory Guides and industry standards. Based on this review, we conclude that the Bechtel QA Program contains the necessary provisions, requirements and controls for compliance with Appendix B to 10 CFR Part 50 and applicable guides and standards and, therefore, is acceptable for the design, procurement, and construction of the standard portion of the SNUPPS plants.

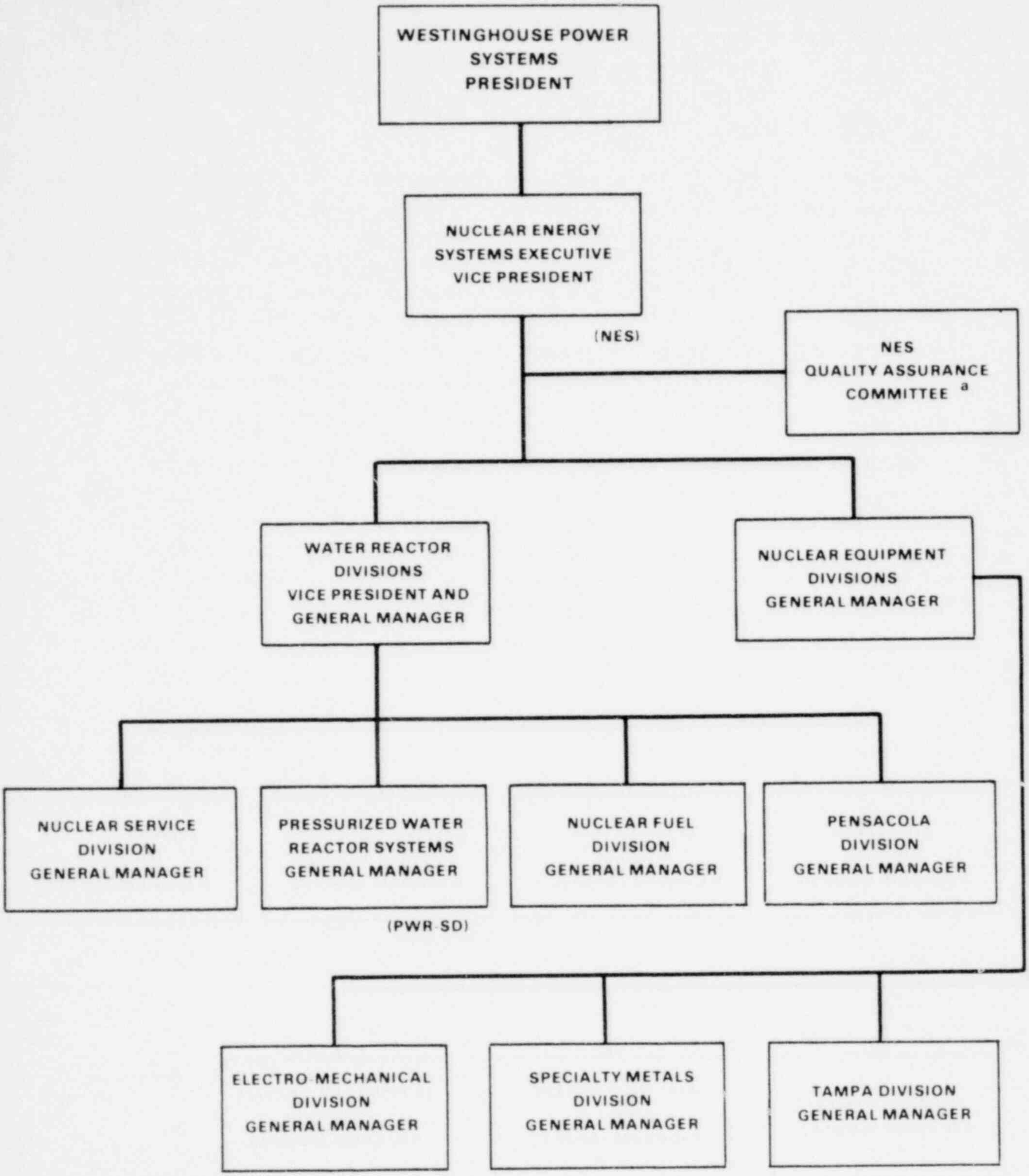
17.4 Westinghouse Electric Corporation

Westinghouse has been contracted by the SNUPPS utilities to provide the nuclear steam supply system for the SNUPPS plants. The SNUPPS PSAR includes, by reference to RESAR-3, Consolidated Version, a description of the Westinghouse QA Program for the SNUPPS plants. Section 17.0 of RESAR-3, Consolidated Version describes the QA Program for Westinghouse's Nuclear Energy Systems Divisions (NES).

Figure 17.3 shows the organization of NES. The Nuclear Energy Systems Divisions operate under an Executive Vice President who reports to the President, Westinghouse Power Systems. The Pressurized Water Reactor Systems Division of NES is the lead division for design and procurement.

Each NES division contains an organization specifically responsible for Quality Assurance and for Quality Control which reports at a management level to assure independence consistent with Criterion 1 of Appendix B to 10 CFR Part 50.

Quality Assurance management in each NES division is free of prime responsibility for schedule or cost, has the authority to stop work pending resolution of QA matters, and has the freedom to (1) identify QA problems, (2) initiate, recommend, or provide solutions through designated channels, (3) verify implementation of solutions, and (4) control further processing, delivery, or installation of nonconforming items. We find that Westinghouse has adequately defined the responsibilities of the organizations performing QA activities and that they are acceptable.



a) The NES Quality Assurance Committee is Composed of the Quality Assurance and Reliability Managers from Each of the NES Divisions. The Committee's Chairman is the PWR-SD Product Assurance Manager.

Figure 17.3 ORGANIZATION OF WESTINGHOUSE NUCLEAR ENERGY SYSTEMS DIVISIONS (NES)

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The Executive Vice President of NES has established a QA Committee which includes the Quality Assurance Reliability Managers of each NES division. The Product Assurance Manager for the Pressurized Water Reactor Systems Division is Chairman of the QA Committee. The QA Committee audits each NES division annually to assess the scope, implementation, and effectiveness of the division's QA Program.

The QA Program applies to all safety related structures, systems, and components within the Westinghouse scope of work. Westinghouse has committed to follow the QA guidance provided by the Commission in "Guidance on Quality Assurance Requirements During Design and Procurement Phase of Nuclear Power Plants," May 24, 1974 (WASH 1283-Revision 1) and "Guidance on Quality Assurance Requirements During Construction Phase of Nuclear Power Plants" May 10, 1974 (WASH 1309). We find this commitment acceptable.

The QA program for each NES division amplifies the common Nuclear Energy Systems Divisions QA policy, as necessary for local application, to account for the different scopes of work in each division. Each NES division documents and implements a program which assures that safety related items meet the applicable requirements of Appendix B to 10 CFR Part 50. Each division manager authorizes, reviews, and approves the QA Program for his division.

A matrix which relates the procedures of the various manuals to the applicable QA criteria of Appendix B to 10 CFR Part 50 is given in RESAR-3, Consolidated Version. Based on our review of this matrix, we conclude that each criterion has been specifically included in written procedures within the Westinghouse QA Program.

The QA Program includes provisions for the control of design information. Design imputs are reviewed, and analyses are accomplished in accordance with applicable codes, standards, and regulatory requirements. Knowledgeable groups within Westinghouse, including QA and reliability personnel, independently review drawings and equipment specifications prior to issuance.

To provide control of purchased items and services, Westinghouse evaluates the QA Program of each prospective supplier of safety related items. Quality Assurance engineers review purchase requisitions, purchase orders, and subsequent change notices. The Nuclear Energy Systems Divisions review and retain supplier documentation which demonstrates acceptable quality. Audits and feedback of discrepancy data are used by the Nuclear Energy Systems Divisions QA engineers to measure supplier performance.

Westinghouse executes a comprehensive audit program which provides NES management with information on the effectiveness of the QA Program. Westinghouse audits activities affecting QA at Westinghouse and at supplier facilities. Audit areas include all QA related procedures and operations. Trained personnel, not having direct responsibilities in the area being audited, conduct the QA audits in accordance with defined procedures and checklists.

In our review, we have evaluated the Westinghouse QA Program for compliance with the Commission's regulations and applicable Regulatory Guides and industry standards. Based on this review, we conclude that the Westinghouse QA Program contains the necessary QA provisions, requirements, and controls which comply with Appendix B to 10 CFR Part 50 and applicable guides and standards and, therefore is acceptable for the nuclear steam supply system for the SNUPPS plants.

17.5 Kansas Gas & Electric Company

The Kansas Gas & Electric Company, as a participating member of the SNUPPS utilities, is organized to control the activities of SNUPPS Project Organization and its principal contractors through membership in the Quality Assurance Committee described in Section 17.2 of the Safety Evaluation Report. Control of the activities at the site will be handled directly by the Kansas Gas & Electric Company.

The Senior Vice President, who reports to the President, is assigned the responsibility for the Operating Department and provides the interaction with the SNUPPS Project Organization. He has assigned the Vice-President for Operations the responsibility for engineering and Quality Assurance (Figure 17.4). The Director of Quality Assurance, who is responsible for managing the QA program, reports to the Vice-President for Operations.

Figure 17.4 shows the Director of Quality Assurance on the same organizational level as other directors in the Nuclear Development Department whose QA activities he verifies. The Director of Quality Assurance has well defined responsibilities and authorities for implementing the QA Program with documented procedures and instructions. Based on our review of the Kansas Gas & Electric Company's organization and associated responsibilities, we find that QA has adequate independence and reports at a sufficiently high management level to accomplish its objectives.

The Kansas Gas & Electric Company implements its QA Program by means of internal audits of the Nuclear Development Department site operations which are conducted by the QA organization. The Director of Quality Assurance and site QA engineers are authorized by the Vice President for Operations to stop work when considered necessary. We find that the Kansas Gas & Electric Company's provisions for implementing its QA Program; with corporate level management involvement, authority from the Vice-President for Operations to enforce QA requirements, and QA stop work authority; are acceptable.

Based on our evaluation of the Kansas Gas & Electric Company's QA organization, we find that it is sufficiently independent of the organizations whose work it verifies; it has clearly defined authorities and responsibilities; it has adequately defined qualification and training requirements for its staff; it is so organized that it can identify QA problems in the other organizations performing QA related work; it can initiate, recommend or provide solutions; and it can verify implementation of

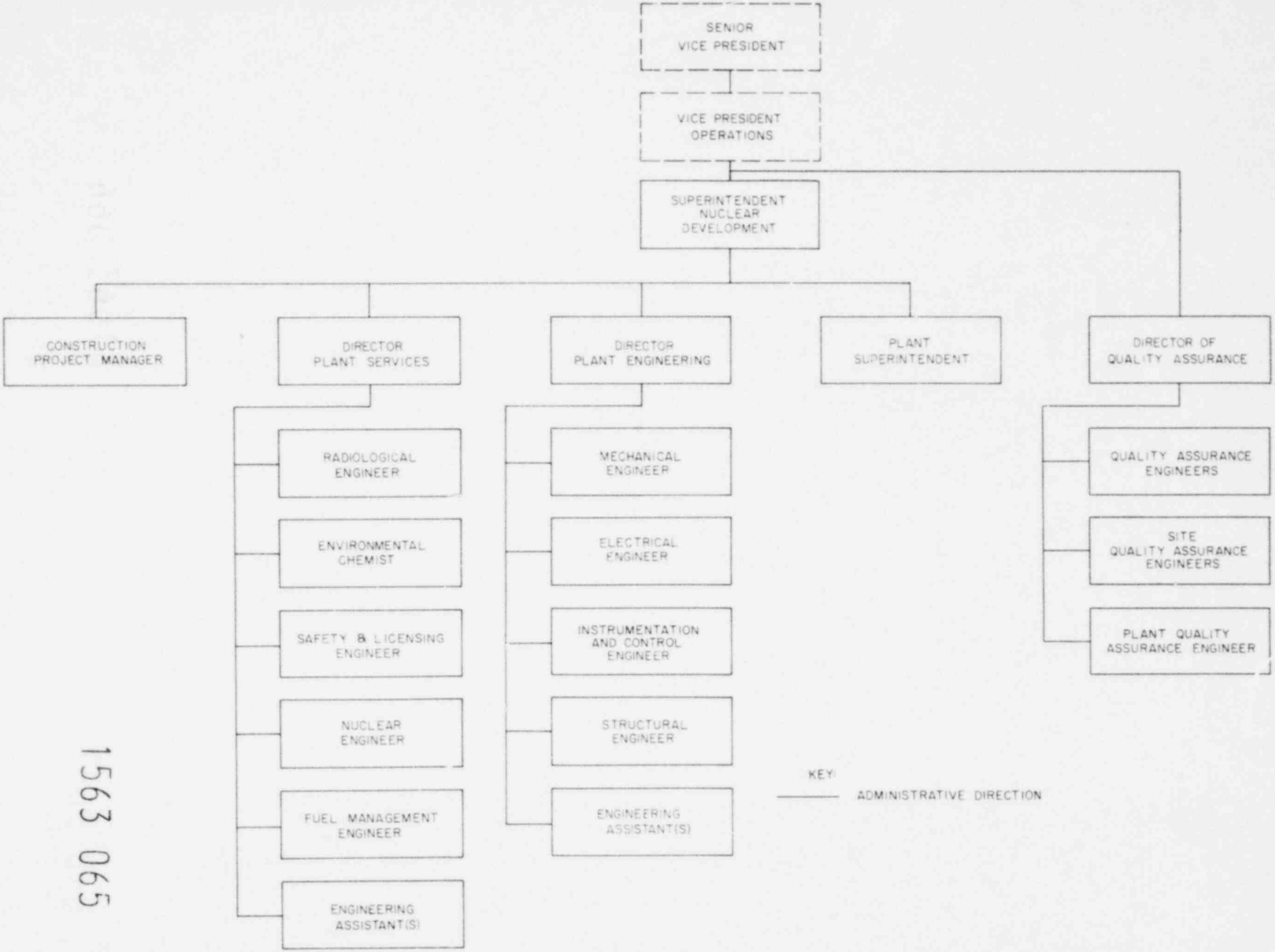

Figure 17.4 KANSAS GAS & ELECTRIC COMPANY ORGANIZATION

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solutions. We, therefore, conclude that the Kansas Gas & Electric Company's organization responsible for QA functions complies with the requirements of Appendix B to 10 CFR Part 50, and is acceptable.

The original QA Program description for the Kansas Gas & Electric Company in the Wolf Creek Site Addendum Report did not provide enough detail to adequately describe the QA Program for design, procurement, and construction of the Wolf Creek plant. In response to our request, Section 17.1 of the Wolf Creek Site Addendum Report was amended to provide a more detailed and comprehensive description of the Wolf Creek QA Program.

The Wolf Creek Site Addendum Report provides a list of the policies and procedures used to administer the QA Program. These policies and procedures form the basis for the Kansas Gas & Electric Company's QA Program for the Wolf Creek Plant. The QA Program specifies the QA requirements to which the Wolf Creek project will comply. The Wolf Creek Site Addendum Report also includes a listing of the QA Program requirements and procedures for the Kansas Gas & Electric Company plus a matrix of these requirements and procedures cross referenced to each criterion of Appendix B to 10 CFR Part 50. Based on our review of this information, we conclude that each criterion of Appendix B to 10 CFR Part 50 has been acceptably included in the QA Program procedures for the Wolf Creek plant.

The Kansas Gas & Electric Company has developed a detailed indoctrination and training program to assure that QA personnel and utility personnel are qualified to a level commensurate with their responsibility.

The QA Program for Wolf Creek is structured in accordance with the Regulatory Guides and industrial standards that are addressed by the Commission in "Guidance on QA Requirements During Design and Procurement Phase of Nuclear Power Plants," May 24, 1974 (WASH 1283-Revision 1) and "Guidance on QA Requirements During the Construction Phase of Nuclear Power Plants," May 10, 1974 (WASH 1309). We find with this commitment, and as a result of our review of the Kansas Gas & Electric Company's description of their QA policies and procedures, that the Kansas Gas & Electric Company has provided an acceptable QA Program. The structures, systems, and components comprising the safety items subject to this program have been identified in the Wolf Creek Site Addendum Report.

The Kansas Gas & Electric Company's verification of an adequate QA Program will be done by surveillance of its principal contractors and subcontractors to assure that inspections will be performed to documented inspection instructions by qualified personnel, and that results will be recorded.

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A system of planned and documented audits, described in the Wolf Creek Site Addendum Report, will be used by the Kansas Gas & Electric Company to verify compliance with all aspects of the QA Program and to assess its effectiveness. The Director of Quality Assurance is responsible for initiating and implementing audits of the Kansas Gas & Electric Company's personnel at the general office and at the site. Auditing of principal contractors (architect-engineer and the nuclear steam supply system supplier), is done by SNUPPS Project Organization described in Section 17.2 of this report. The Kansas Gas & Electric Company's audit results will be reviewed and corrective action taken by responsible management. Follow-up action is taken to assure corrective action. We find the Kansas Gas & Electric Company's audit commitments acceptable.

In our review, we have evaluated the Kansas Gas & Electric Company's QA Program for compliance with the Commission's regulations and applicable Regulatory Guides and industry standards. Based on this review, we conclude that the Kansas Gas & Electric Company's QA Program contains the necessary QA organization, provisions, requirements, and controls for compliance with Appendix B to 10 CFR Part 50 and applicable guides and standards and, therefore, is acceptable for the design, procurement, and construction of the Wolf Creek plant.

17.6 Daniel International Corporation (Formerly Daniel Construction Company)

Daniel, as the construction contractor, will control and implement the Quality Control program required during the construction of the Wolf Creek plant. Activities include bulk procurement, receiving of materials and items from Bechtel, and onsite fabricating, erecting, manufacturing, inspecting, cleaning, and testing. The Daniel organization for Wolf Creek is shown in Figure 17.5.

The Corporate Construction Division Group Vice President is responsible for the overall QA Program. The Corporate QA Manager is responsible for establishing and implementing the QA Program.

Quality Assurance personnel from the corporate QA office prepare the QA Program for Wolf Creek. The QA Manager reviews and approves the QA Program and all subsequent changes.

The Power Division is assigned responsibility for construction. A Project Manager, reporting to the Power Division, is assigned responsibility for construction of the Wolf Creek plant. The Quality Control Manager, reporting to the Project Manager, is responsible to provide the inspection services, documentation, and control of construction activities and has stop work authority to control unsatisfactory performance.

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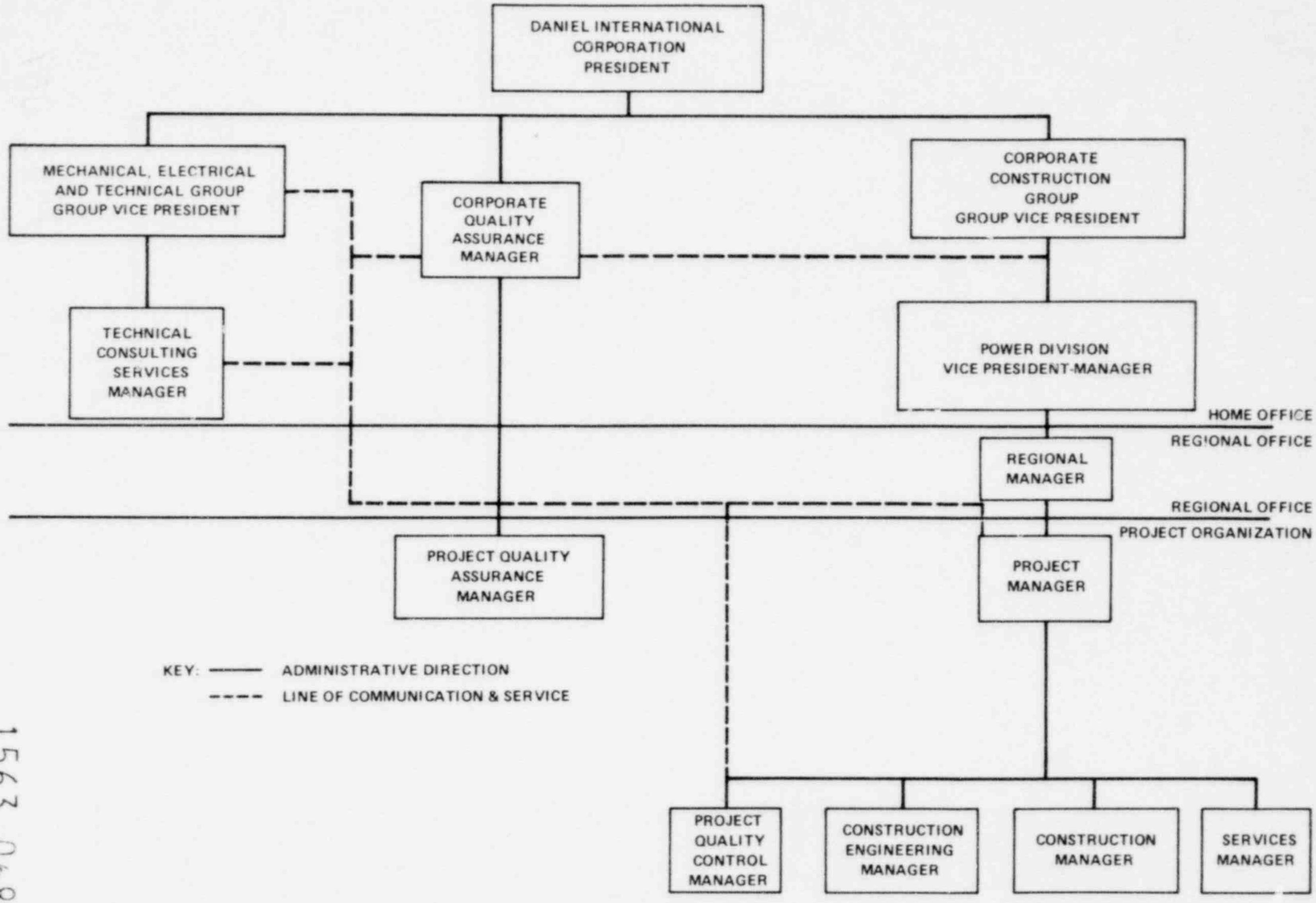

FIGURE 17.5 DANIEL INTERNATIONAL CORPORATION ORGANIZATION

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The Wolf Creek Site Addendum Report provides a list of the policies and procedures used to administer the Daniel QA Program plus a matrix of these requirements and procedures cross referenced to the criteria of Appendix B to 10 CFR Part 50. Daniel has committed to comply with the Regulatory Guides and standards addressed by the Commission in "Guidance on QA Requirements During Design and Procurement Phase of Nuclear Power Plants," May 24, 1974 (WASH 1283-Revision 1) and "Guidance on QA Requirements During the Construction Phase of Nuclear Power Plants," May 10, 1974 (WASH 1309). We find with the above commitment, and with the summary description of the procedures, that Daniel has provided an acceptable QA Program for construction of the Wolf Creek plant.

A system of planned and documented audits is conducted by the corporate QA organization the resident Project QA Manager. The planned audit activities include the construction work operations of subcontractors, suppliers, and associated records.

Management reviews of the status and adequacy of the QA Program are accomplished through review of audit reports for QA trends. Based on our review of the Daniel organization responsible for QA functions and the QA Program provisions, we conclude that Daniel has a QA Program which complies with Appendix B to 10 CFR Part 50, and applicable guides and standards and, therefore, is acceptable for the construction of the Wolf Creek plant.

17.7 Implementation of Quality Assurance Program

The Office of Inspection and Enforcement has inspected the QA Program implementation for the Wolf Creek plant. Based on these inspections, the Office of Inspection and Enforcement concludes that the Wolf Creek QA Program is consistent with the status of the project.

17.8 Conclusions

In our review, we have evaluated the QA Programs of the Kansas Gas & Electric Company, the SNUPPS Project Organization, Daniel, Bechtel, and Westinghouse for the design, and construction of the Wolf Creek plant to determine compliance with the Nuclear Regulatory Commission's regulations and applicable Regulatory Guides and industry standards. Based on this review, we conclude that the QA Programs comply with Appendix B to 10 CFR Part 50 and applicable guides and standards and, therefore, are acceptable for the design, procurement and construction of the Wolf Creek plant.

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18.0 REVIEW BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The application for the Wolf Creek plant is being reviewed by the Advisory Committee on Reactor Safeguards. We intend to issue a supplement to this Safety Evaluation Report after the Committee's report to the Commission relative to their review is available. The supplement will append a copy of the Committee's report and will address comments made by the Committee, and will also describe steps taken by the Commission's staff to resolve any issue raised as a result of the Committee's review.

19.0 COMMON DEFENSE AND SECURITY

The application reflects that the activities to be conducted would be within the jurisdiction of the United States and that all the directors and principal officers of the applicants are citizens of the United States.

The applicants are 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 applicants have agreed to safeguard any such data that might become involved in accordance with the requirements of 10 CFR Part 50. The applicants 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.

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20.0 FINANCIAL QUALIFICATIONS

The Commission's regulations which relate to financial data and information required to establish financial qualifications for an applicant for a facility construction permit are Paragraph 50.33(f) of 10 CFR Part 50 and Appendix C to 10 CFR Part 50. We are continuing our review of the financial qualifications of the applicants and will report the results of our evaluations in a supplement to this report.

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21.0 CONCLUSIONS

Based on our analysis of the proposed design of the Wolf Creek facility and upon favorable resolution of the outstanding matters discussed in this report, we will be able to conclude that, in accordance with the provisions of Section 50.35(a) of 10 CFR Part 50:

(1) The applicants have described the proposed design of the facility including, but not limited to, the principal architectural and engineering criteria for the design, and have identified the major features or components incorporated therein for the protection of the health and safety of the public;

(2) Such further technical or design information as may be required to complete the safety analysis and which can reasonably be left for later consideration will be supplied in the final safety analysis report;

(3) Safety features or components which require research and development have been described by the applicants and the applicants have identified, and there will be conducted, research and development programs reasonably designed to resolve safety questions associated with such features or components;

(4) On the bases of the foregoing, there is reasonable assurance that (i) such safety questions will be satisfactorily resolved at or before the latest date stated in the application for completion of construction of the proposed facility, and (ii) taking into consideration the site criteria contained in 10 CFR Part 100, the proposed facility can be constructed and operated at the proposed location without undue risk to the health and safety of the public;

(5) The applicants are technically qualified to design and construct the proposed facility;

(6) The applicants have reasonably estimated the costs and are financially qualified to design and construct the proposed facility; and

(7) The issuance of permits for construction of the facility will not be inimical to the common defense and security or to the health and safety of the public.

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APPENDIX A

CHRONOLOGY OF RADIOLOGICAL REVIEW OF THE

WOLF CREEK GENERATING STATION, UNIT NO. 1

April 24, 1973	Letter to applicant regarding advance notification of submission of application
May 31, 1973	Letter from applicant in response to letter of April 24, 1973
August 1, 1973	Meeting with SNUPPS to discuss site-related information to be included in the SNUPPS Initial Report
August 20, 1973	Meeting with SNUPPS to discuss adequacy of onsite meteorological programs for SNUPPS sites
August 29, 1973	Meeting with SNUPPS to discuss quality assurance program and pre-application requirements
September 6, 1973	Meeting with SNUPPS to discuss geology and seismology of proposed plant sites
October 9, 1973	Letter to applicant concerning staff report on anticipated transients without scram
October 15, 1973	Letter from SNUPPS providing information relative to quality assurance
October 29, 1973	Letter from SNUPPS transmitting SNUPPS Initial Report
November 6, 1973	Meeting with SNUPPS to discuss quality assurance program
December 20, 1973	Letter from SNUPPS regarding "Anticipated Transients Without Scram"
February 11, 1974	Letter to SNUPPS transmitting comments on SNUPPS Initial Report
March 21, 1974	Letter from SNUPPS regarding schedule for submittal of information
April 1, 1974	Letter from applicant transmitting application for preliminary review
April 26, 1974	Letter from applicant transmitting site addendum for preliminary review
April 30, 1974	Letter from SNUPPS transmitting PSAR for preliminary review
May 7, 1974	Letter to applicant acknowledging receipt of application
May 17, 1974	Application docketed
June 18, 1974	Letter to applicant regarding results of preliminary review
June 19, 1974	Meeting with SNUPPS to discuss results of preliminary review
July 18, 1974	Letter from SNUPPS regarding letters of June 18, 1974
July 19, 1974	Submittal of Amendment No. 1 (Revision 1 to Environmental Report)
July 25, 1974	Submittal of Amendment No. 2 (Revision 0 to Site Addendum), in response to letter of June 18, 1974
August 15, 1974	Letter to applicant transmitting copy of RESAR-3 correspondence

August 15, 1974	Letter to applicant concerning acceptability of Site Addendum
August 16, 1974	Letter from SNUPPS transmitting Revision 1 to PSAR, consisting of responses to letter of June 18, 1974
August 27, 1974	Letter to applicant transmitting review schedule
September 3, 1974	Submittal of Amendment No. 3 (Revision 1 to Site Addendum) in response to letter of June 18, 1974 - incorporation of Revision 1 of SNUPPS PSAR as Amendment No. 4
September 11, 1974	Letter from applicant in response to letter of August 27, 1974
September 11-12, 1974	Site visit
September 13, 1974	TWX to applicant regarding Wolf Creek schedule and fuel loading
September 13, 1974	Submittal of Amendment No. 5 (Revision 2 to Environmental Report)
September 19, 1974	TWX to applicant regarding Wolf Creek schedule and fuel loading
September 19, 1974	Meeting with SNUPPS to discuss overall construction schedules
September 19, 1974	Letter from applicant in response to TWX of September 13, 1974
September 20, 1974	Letter from applicant in response to TWX of September 19, 1974
September 25, 1974	Letter from applicant advising of delay in completion of Wolf Creek
September 25, 1974	Letter from SNUPPS transmitting Revision 2 to PSAR, consisting of additional responses to letter of June 18, 1974, and other changes
September 26, 1974	Letter from applicant concerning ATWS analyses
September 30, 1974	Letter to applicant requesting additional information and response to staff positions
October 1, 1974	Letter from applicant transmitting correction to letter of September 26, 1974
October 2, 1974	Letter from applicant incorporating Revision 2 of SNUPPS PSAR as Amendment No. 6
October 7, 1974	Letter to applicant requesting additional information and response to additional staff positions
October 8, 1974	Meeting with SNUPPS to discuss status of Bechtel topical reports relating to the review of the SNUPPS application
October 17, 1974	Letter from applicant transmitting information regarding site suitability
November 1, 1974	Letter from applicant transmitting "Through and Overflow Rockfill Dams-New Design Techniques" and "A Mathematical Theory for the Calculation of the Stability of Slopes in Open Cast Mines"
November 7, 1974	Meeting with SNUPPS to discuss meteorological questions
November 8, 1974	Letter from SNUPPS transmitting Revision 3 to PSAR, consisting of responses to request dated September 30, 1974, and other changes
November 8, 1974	Submittal of Amendment No. 7 (Revision 2 to Site Addendum) in response to letter of October 7, 1974 - incorporating Revision 3 to SNUPPS PSAR as Amendment No. 8

November 13, 1974	Meeting with applicant to discuss proposed design of ultimate heat sink dam
November 14, 1974	Letter to applicant requesting additional information
November 29, 1974	Letter from applicant concerning security plan
December 13, 1974	Letter from applicant transmitting Site Addendum information
December 20, 1974	Submittal of Amendment No. 9 (Revision 3 to Site Addendum), in response to request dated November 14, 1974
December 31, 1974	Letter from SNUPPS transmitting Revision 4 to PSAR, consisting of responses to requests and other changes
January 3, 1975	Letter from applicant incorporating Revision 4 of SNUPPS PSAR as Amendment No. 10
January 3, 1975	Letter from applicant transmitting report, "On the Spreading of a Heavy Gas Released Near the Ground"
January 8, 1975	Letter from applicant transmitting information to be incorporated in Site Addendum
January 15, 1975	Letter to applicant stating that proprietary information submitted November 29, 1974 will be withheld from public disclosure
January 21, 1975	Meeting with applicant to discuss issues regarding application
January 23, 1975	Letter to SNUPPS requesting additional information and response to staff positions
January 23, 1975	Letter to applicant requesting additional information
January 30, 1975	Meeting with SNUPPS to discuss containment analysis
February 4, 1975	Submittal of Amendment No. 11 (Revision 4 to Site Addendum)
February 13, 1975	Letter from SNUPPS transmitting information relative to heat exchanger and feedwater systems
February 13, 1975	Letter to SNUPPS requesting additional information
February 14, 1975	Submittal of Amendment No. 12 (Revision 3 to Environmental Report)
February 17, 1975	Letter from applicant incorporating information submitted on February 13, 1975, by SNUPPS into the Wolf Creek application
February 20, 1975	Letter to applicant requesting additional information
February 25, 1975	Meeting with SNUPPS to discuss seismic analysis methods for SNUPPS plants
March 7, 1975	Letter from SNUPPS transmitting Revision 5 to PSAR, consisting of responses to requests dated January 23 and February 13, 1975
March 7, 1975	Letter from applicant incorporating Revision 5 of SNUPPS PSAR as Amendment No. 14
March 11, 1975	Submittal of Amendment No. 13 (Revision 5 to Site Addendum), in response to request of January 23, 1975
March 11, 1975	Letter to SNUPPS requesting additional information
March 12, 1975	Letter from SNUPPS transmitting errata to Revision 5 of PSAR
March 21, 1975	Letter from applicant incorporating errata to Revision 5 of SNUPPS PSAR

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March 24, 1975	Letter from SNUPPS furnishing schedule for responding to requests for additional information
March 26, 1975	Letter from applicant transmitting information concerning security plan
March 27, 1975	Submittal of Amendment No. 15 (Revision 6 to Site Addendum), in response to request of February 20, 1975
March 28, 1975	Letter from SNUPPS transmitting Revision 6 to PSAR, consisting of responses to requests of January 23 and February 13, 1975
March 31, 1975	Letter from applicant incorporating Revision 6 of SNUPPS PSAR as Amendment No. 16
April 11, 1975	Letter from applicant transmitting information relative to LWA
April 11, 1975	Letter to SNUPPS concerning containment temperature analysis for main stream line breaks
April 15, 1975	Meeting with SNUPPS to discuss containment temperature analysis
April 21, 1975	Letter to SNUPPS requesting additional information regarding Bechtel quality assurance program
April 30, 1975	Letter from SNUPPS transmitting Revision 7 to PSAR, consisting of information concerning equipment design and other changes
April 30, 1975	Letter from applicant incorporating Revision 7 of SNUPPS PSAR as Amendment No. 17
May 2, 1975	Letter from SNUPPS transmitting information relative to the Bechtel quality assurance program
May 5, 1975	Letter from applicant incorporating SNUPPS letter of May 2, 1975
May 5, 1975	Letter from SNUPPS transmitting responses to requests for information regarding SNUPPS PSAR
May 5, 1975	Letter from SNUPPS transmitting additional information relative to the Bechtel quality assurance program
May 8, 1975	Letter from applicant incorporating SNUPPS letter of May 5, 1975
May 8, 1975	Letter from applicant incorporating SNUPPS letter of May 5, 1975 concerning quality assurance
May 9, 1975	Letter from SNUPPS transmitting information concerning main steam line breaks
May 9, 1975	Submittal of Amendment No. 18 (Revision 7 to Site Addendum), in response to requests of January 23 and February 20, 1975
May 12, 1975	Letter from applicant incorporating SNUPPS letter of May 9, 1975
May 13, 1975	Meeting with SNUPPS to discuss containment subcompartment analysis
May 19, 1975	Letter from SNUPPS transmitting information to clarify previous revisions to SNUPPS PSAR
May 21, 1975	Letter from applicant transmitting information concerning structural fill and backfill
May 22, 1975	Letter from applicant incorporating SNUPPS letter of May 19, 1975
May 22, 1975	Letter to applicant regarding staff position on loose parts monitoring systems
June 6, 1975	Letter from applicant in response to letter of May 22, 1975

June 16, 1975	Letter from SNUPPS transmitting Revision 8 to PSAR, which incorporates information previously submitted and contains information regarding quality assurance and other changes
June 24, 1975	Submittal of Amendment No. 19 (Revision 8 to Site Addendum), consisting of information relative to earlier commitments
June 25, 1975	Letter from applicant incorporating Revision 8 of SNUPPS PSAR as Amendment No. 20
June 26, 1975	Meeting with SNUPPS Utilities to discuss outstanding issues in review of SNUPPS PSAR
July 3, 1975	Letter to applicant concerning ECCS-FAC analysis
July 11, 1975	Letter from SNUPPS confirming commitments made at June 26, 1975 meeting and transmitting other information for PSAR
July 11, 1975	Letter from applicant incorporating SNUPPS letter of July 11, 1975
July 23, 1975	Letter from SNUPPS transmitting information concerning design against tornado-generated missiles, vibration testing of reactor internals, and other design information
July 24, 1975	Letter from applicant incorporating SNUPPS letter of July 23, 1975
July 24, 1975	Letter to applicant concerning gross failed-fuel monitoring system
July 25, 1975	Meeting with SNUPPS Utilities to discuss outstanding issues in review of SNUPPS PSAR
July 29, 1975	Letter from SNUPPS confirming commitments made at July 25, 1975 meeting, and transmitting information relative to manually controlled electrically operated valves, engineered safety feature requirements, and gross failed fuel monitoring system
August 1, 1975	Letter from applicant incorporating SNUPPS letter of July 29, 1975
August 4, 1975	Letter from SNUPPS transmitting Revision 9 to PSAR which incorporates information submitted July 11 and July 23, 1975, and contains information concerning ECCS-FAC analysis and other design information
August 4, 1975	Letter from applicant incorporating Revision 9 of SNUPPS PSAR as Amendment 21
August 4, 1975	Letter to applicant regarding WCAP-7705, "Engineered Safeguards Final Device Actuator Testing"
August 7, 1975	Letter from SNUPPS transmitting information concerning design analysis
August 8, 1975	Letter from applicant incorporating SNUPPS letter of August 7, 1975
August 13, 1975	Letter from applicant providing information concerning seismology
August 15, 1975	Letter from SNUPPS advising of plans for response to staff letter of August 4, 1975
August 18, 1975	Letter from applicant transmitting a report on preoperational meteorological monitoring
August 19, 1975	ACRS Subcommittee meeting with staff and SNUPPS Utilities
August 26, 1975	Letter from SNUPPS transmitting information in response to staff letter of August 4, 1975 regarding design criteria for fire stops and seals
August 26, 1975	Letter from applicant incorporating SNUPPS letter of August 26, 1975
September 3, 1975	Meeting with SNUPPS Utilities to discuss proposed design change for SNUPPS radwaste building

APPENDIX B

ELECTRICAL AND MECHANICAL

EQUIPMENT SEISMIC QUALIFICATION

PROGRAM

A. Seismic Test for Equipment Operability

(1) A test program is required to confirm the functional operability of all seismic Category I electrical and mechanical equipment and instrumentation during and after an earthquake of magnitude up to and including the safe shutdown earthquake. Analysis without testing may be acceptable only if structural integrity alone can assure the design intended function. When complete seismic testing is impracticable, a combination of test and analysis may be acceptable.

(2) The characteristics of the required input motion should be specified by one of the following:

(a) response spectrum
(b) power spectral density function
(c) time history

(3) Equipment should be tested in the operational condition. Operability should be verified during and after the testing.

(4) The actual input motion should be characterized in the same manner as the required input motion, and the conservatism in amplitude and frequency spectrum used should cover the range from 1 through 33 hertz. Any exceptions taken require justification.

(5) Seismic excitations generally have a broad frequency content. Random vibration input motion should be used. However, a single frequency input, such as sine beats, may be applicable provided one of the following conditions are met:

(a) The characteristics of the required input motion indicate that the motion is dominated by one frequency (i.e., by structural filtering effects).

(b) The anticipated response of the equipment is adequately represented by one mode.

(c) The input has sufficient intensity and duration to excite all modes to the required magnitude, such that the testing response spectra will envelop the corresponding response spectra of the individual modes.

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(6) The input motion should be applied to one vertical and one principal (or two orthogonal) horizontal axes simultaneously unless it can be demonstrated that the equipment response along the vertical direction is not sensitive to the vibratory motion along the horizontal direction, and vice versa. The time phasing of the inputs in the vertical and horizontal directions must be such that a purely rectilinear resultant input is avoided. The acceptable alternative is to have vertical and horizontal inputs in-phase, and then repeated with inputs 180 degrees out-of-phase. In addition, the test must be repeated with the equipment rotated 90 degrees horizontally.

(7) The fixture design should meet the following requirements:

(a) Simulate the actual service mounting.

(b) Cause no dynamic coupling to the test item.

(8) The in-situ application of vibratory devices to superimpose the seismic vibratory loadings on the complex active device for operability testing is acceptable when application is justifiable.

(9) The test program may be based upon selectively testing a representative number of mechanical components according to type, load level, size, etc. on a prototype basis.

B. Seismic Design Adequacy of Supports

(1) Analyses or tests should be performed for all supports of electrical and mechanical equipment and instrumentation to assure their structural capability to withstand seismic excitation.

(2) The analytical results must include the following:

(a) The required input motions to the mounted equipment should be obtained and characterized in the manner as stated in Section A(2).

(b) The combined stresses of the support structures should be within the limits of ASME Section III, Subsection NF - "Component Support Structures" (draft version) or other comparable stress limits.

(3) Supports should be tested with equipment installed. If the equipment is inoperable during the support test, the response at the equipment locations should be monitored and characterized in the manner as stated in Section A(2). In such a case, equipment should be tested separately and the actual input to the equipment should be more conservative in amplitude and frequency content than the monitored response.

(4) The requirements of Sections A(2), A(4), A(5), A(6), A(7) are applicable when tests are conducted on the equipment supports.

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APPENDIX C

TECHNICAL POSITION

APPLICATION OF THE SINGLE FAILURE CRITERION TO MANUALLY-CONTROLLED ELECTRICALLY-OPERATED VALVES

A. Background

Where a single failure in an electrical system can result in loss of capability to perform a safety function, the effect on plant safety must be evaluated. This is necessary regardless of whether the loss of safety function is caused by a component failing to perform a requisite mechanical motion, or by a component performing an undesirable mechanical motion.

This position establishes the acceptability of disconnecting power to electrical components of a fluid system as one means of designing against a single failure that might cause an undesirable component action. These provisions are based on the assumption that the component is then equivalent to a similar component that is not designed for electrical operation, e.g., a valve that can be opened or closed only by direct manual operation of the valve. They are also based on the assumption that no single failure can both restore power to the electrical system and cause mechanical motion of the components served by the electrical system. The validity of these assumptions should be verified when applying this position.

B. Branch Technical Position

(1) Failures in both the "fail to function" sense and the "undesirable function" sense of components in electrical systems of valves and other fluid system components should be considered in designing against a single failure, even though the valve or other fluid system component may not be called upon to function in a given safety operational sequence.

(2) Where it is determined that failure of an electrical system component can cause undesired mechanical motion of a valve or other fluid system component and this motion results in loss of the system safety function, it is acceptable, in lieu of design changes that also may be acceptable, to disconnect power to the electric systems of the valve or other fluid system component. The plant technical specifications should include a list of all electrically-operated valves, and the required positions of these valves, to which the requirement for removal of electric power is applied in order to satisfy the single failure criterion.

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(3) Electrically-operated valves that are classified as "active" valves, i.e., are required to open or close in various safety system operational sequences, but are manually-controlled, should be operated from the main control room. Such valves may not be included among those valves from which power is removed in order to meet the single failure criterion unless: (a) electrical power can be restored to the valves from the main control room, (b) valve operation is not necessary for at least ten minutes following occurrence of the event requiring such operation, and (c) it is demonstrated that there is reasonable assurance that all necessary operator actions will be performed within the time shown to be adequate by the analysis. The plant technical specifications should include a list of the required positions of manually-controlled, electrically-operated valves and should identify those valves to which the requirement for removal of electric power is applied in order to satisfy the single failure criterion.

(4) When the single failure criterion is satisfied by removal of electrical power from valves described in (2) and (3), above, these valves should have redundant position indication in the main control room and the position indication system should, itself, meet the single failure criterion.

(5) The phrase "electrically-operated valves" includes both valves operated directly by an electrical device (e.g., a motor-operated valve or a solenoid-operated valve) and those valves operated indirectly by an electrical device (e.g., an air-operated valve whose air supply is controlled by an electrical solenoid valve).

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APPENDIX D
SUPPLEMENT TO REGULATORY GUIDE 1.47

Until Regulatory Guide 1.47 is revised, the following should be used as a supplement in its implementation:

The design criteria for the indication system should reflect the importance of both providing accurate information for the operator and reducing the possibility for the indicating equipment to adversely affect the monitored safety systems. In developing the design criteria, the following should be considered:

(1) The bypass indicators should be arranged to enable the operator to determine the status of each safety system and determine whether continued reactor operation is permissible.

(2) When a protective function of a shared system can be bypassed, indication of that bypass condition should be provided in the control room of each affected unit.

(3) Means by which the operator can cancel erroneous bypass indications, if provided, should be justified by demonstrating that the postulated cases of erroneous indications cannot be eliminated by another practical design.

(4) Unless the indication system is designed in conformance with criteria established for safety systems, it should not be used to perform functions that are essential to the health and safety of the public. Neither should administrative procedures require immedate operator action based solely on bypass indications.

(5) The indication system should be designed and installed in a manner which precludes the possibility of adverse effects on the plant's safety systems. Failure or bypass of a protective function should not be a credible consequence of failures occurring in the indication equipment, and the bypass indication should not reduce the required independence between redundant safety systems.

(6) The indication system should include a capability of assuring its operable status during normal plant operation to the extent that the indicating and/or annunciating function can be verified.

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APPENDIX E

BIBLIOGRAPHY FOR THE WOLF CREEK PLANT

SAFETY EVALUATION REPORT

NOTE: Documents referenced in or used to prepare this Safety Evaluation Report, excluding those listed in the PSAR, may be obtained at the source stated in the Bibliography or, where no specific source is given, at most major public libraries. Correspondence between the Commission and the applicants (PSAR, Environmental Report, and application) and Commission Rules and Regulations and Regulatory Guides may be inspected at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D. C. Correspondence between the Commission and the applicants may also be inspected at the Office of the County Clerk, Coffey County Courthouse, Burlington, Kansas. Specific documents relied upon by the Commission's staff and referenced in this Safety Evaluation Report are listed as follows:

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