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Mr. D. W. Mazur, Managing Director
Washington Public Power Supply System
3000 George Washington Way
Post Office Box 968
Richland, Washington 99352

Dear Mr. Mazur:

Subject: Issuance of Draft Safety Evaluation Report

The NRC staff has prepared the enclosed Draft Safety Evaluation Report (DSER) for Washington Nuclear Project No. 3 (WNP-3). In general, the DSER reflects the status of the staff review of WNP-3 FSAR and associated submittals, through the period of time until construction delay was announced in 1983, and is being issued at this time for the purpose of documenting and preserving the staff's review to date. We feel that this DSER will provide a benchmark from which to commence any future licensing review in the event of restart of the construction of WNP-3.

If you have any questions, please contact B. K. Singh at (301) 492-8423.

Sincerely,

Original signed by:
Thomas M. Novak

Thomas M. Novak, Assistant Director
for Licensing
Division of Licensing
Office of Nuclear Reactor Regulation

Enclosure:
As stated

cc: See next page

*Previous concurred on by:

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10/12/85	10/29/85	10/30/85	11/19/85

Mr. D. W. Mazur
Washington Public Power Supply System

WPPSS Nuclear Project No. 3
(WNP-3)

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DRAFT SAFETY EVALUATION REPORT

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

NUCLEAR PROJECT NO. 3

DOCKET NO. 50-508

November 1985

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1 INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 Introduction

This report is a Draft Safety Evaluation Report (DSER) on the application for an operating license (OL) for Washington Public Power Supply System (WPPSS) Nuclear Project No. 3 (WNP-3). In March 1974, WPPSS filed an application with the U.S. Nuclear Regulatory Commission (NRC) for permits to construct and operate the proposed WNP-3 plant. The plant is located in southeastern Grays Harbor County, Washington, approximately 1 mile south of the Chehalis River near its confluence with the Satsop River. The largest cities within 50 miles of the site are Olympia and Aberdeen. Olympia, the state capital, is 26 miles east of the site; Aberdeen is 16 miles west of the site.

Following reviews by the staff and the Advisory Committee on Reactor Safeguards (ACRS), public hearings were held before an Atomic Safety and Licensing Board. A construction permit (CP) for WNP-3 (CPPR-154) was issued on April 11, 1978. In response to the OL application for WNP-3, the NRC staff performed an acceptance review and, on August 20, 1982, issued a letter accepting the application. Information received on the WNP-3 OL application was docketed on August 22, 1982.

On July 8, 1983, the WPPSS Executive Board adopted a resolution calling for an immediate construction delay of WNP-3 until an ensured source of funding for continued construction could be obtained. The applicant informed the staff that, as of September 30, 1983, construction of WNP-3 was about 75% complete. By letter dated November 18, 1983, WPPSS informed the staff that the projected fuel load date ranges from June 1987 to December 1989. A detailed implementation plan for construction delay at WNP-3 was submitted to the staff on September 15, 1983.

Before issuing an OL for a nuclear power plant, the NRC staff is required to conduct a review of the effects of the plant on public health and safety. The staff safety review of WNP-3 has been based on NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Reactors, LWR

Edition" (SRP). In general, an audit review of each of the areas listed in the Areas of Review section of the SRP was performed according to the guidelines provided in the Review Procedures portion of the SRP. Exceptions to this practice are noted in the applicable sections of this report.

This DSER summarizes the results of the staff's radiological safety review of WNP-3 and delineates the scope of the technical details considered in evaluating the radiological safety aspects of its proposed operation. The design of the station was reviewed against Federal regulations, CP criteria, and the SRP, except where noted otherwise. The SRP covers a variety of site conditions and plant designs. Each SRP section is written to provide the complete procedure and all acceptance criteria for all the areas of review pertinent to the section. However, for any given application, the staff may select and emphasize particular aspects of each SRP section as appropriate for the application. In some cases, the major portion of the review of a plant feature may be done on a generic basis, with the designer of that feature, rather than in the context of reviews of particular applications from utilities. In other cases, a plant feature may be sufficiently similar to that of a previous plant so that a de novo review of the feature is not needed.

During the course of its review, the staff held a number of meetings with representatives of the applicant to discuss the design, construction, and proposed operation of the plant. The staff requested additional information, which the applicant provided in amendments to the Final Safety Analysis Report (FSAR). This information is available to the public for review at the NRC Public Document Room at 1717 H Street, N.W., Washington D.C. 20555 and at the Local Public Document Room at the W. H. Memorial Library, 125 Main Street, South, Montesano, Washington 28563.

Following the incident at the Three Mile Island Unit 2 Nuclear Plant (TMI-2), the Commission paused in its licensing activities to assess the impact of the incident. During this pause, the recommendations of several groups established to investigate the lessons learned from TMI-2 became available. All available recommendations were correlated and assimilated in a "TMI Action Plan," published as NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident." Additional guidance relating to implementation of the Action Plan is in NUREG-0737, "Clarification of TMI Action Plan Requirements," and in Supplement 1

to NUREG-0737. Licensing requirements based on the lessons learned from the TMI-2 incident have been established to provide additional safety margins. These have been incorporated into the design and operation of WNP-3.

As part of its review of the application against the NRC regulations, the staff will ask the applicant to certify that WNP-3 meets the applicable requirements of Parts 20, 50, 51, and 100 of Title 10 of the Code of Federal Regulations (CFR). Following the applicant's response to this request, the staff will address its findings in this area in the Safety Evaluation Report (SER).

In accordance with the provisions of the National Environmental Policy Act (NEPA) of 1969, a Final Environmental Statement (FES) that sets forth the environmental considerations related to the proposed construction and operation of WNP-3 was published in May 1985 (NUREG-1033).

The review and evaluation of WNP-3 for an OL is only one of many stages at which the staff reviews the design, construction, and operating features of the facility. The facility design was extensively reviewed before the applicant was granted a CP for the facility. Construction of the facility has been monitored in accordance with a detailed monitoring and inspection program at the OL stage. The NRC staff has reviewed the final design of the facility to determine that the Commission's regulations have been met. If an OL is granted, WNP-3 must be operated in accordance with the terms of the OL and the Commission's regulations, and the facility will be subject to the staff's continuing inspection program.

In addition to the NRC staff review, the ACRS will review the application and will meet with both the applicant and the staff to discuss the final design and proposed operation of the plant. The Committee's report to the Chairman of the NRC will be included in a supplement to the SER.

The NRC Project Manager assigned to the OL application for WNP-3 is Mr. Braj K. Singh. Mr. Singh may be contacted by calling (301) 492-8423 or by writing:

Mr. Braj K. Singh
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

1.2 Standard Design

The FSAR submitted with the WNP-3 application describes the design of the balance-of-plant (BOP) structures, systems, and components and incorporates, by reference, the Combustion Engineering report, "Combustion Engineering Standard Safety Analysis Report" (CESSAR FSAR). CESSAR describes the design of the System 80 nuclear steam supply system (NSSS).

The initial Commission policy statement on standardization of nuclear power plants was issued on April 28, 1972. It provided the impetus for both industry and the Commission to initiate active planning in their respective areas in order to realize the benefits of standardization while maintaining protection of the health and safety of the public and of the environment. In a subsequent statement issued on March 5, 1°73, the Commission announced its intent to implement a standardization policy for nuclear power plants. WASH-1341, "Programmatic Information for the Licensing of Standardized Nuclear Power Plants," was issued on August 20, 1974. Amendment 1 to WASH-1341, dealing with "options" and "overlaps," was issued on January 16, 1975. The regulations governing the submittal and review of standard designs under the "reference system" option are found in Appendix O to 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities," and 10 CFR 2.110.

CESSAR was submitted by Combustion Engineering in the form of an application for a Final Design Approval (FDA) from the Commission and in response to Option 1 of the Commissions' standardization policy. Option 1 allows for the review of a "reference system" that involves an entire facility design or major fraction of a design outside the context of a license applicaticn. On December 21, 1979, the application for CESSAR was docketed.

The staff evaluation of CESSAR is presented in NUREG-0852, "Safety Evaluation Report Related to the Final Design Approval of the Combustion Engineering Standard Nuclear Steam Supply System (CESSAR)," hereinafter referred to as the CESSAR SER. The CESSAR SER has been incorporated by reference into this SER.

1.2.1 References to the CESSAR SER

The WNP-3 NSSS design description and evaluation are incorporated into the WNP-3 SER by reference to the appropriate CESSAR SER chapter, section, table, or figure. For example, the phrase "the staff evaluation is presented in the CESSAR SER, Section 9.3.5" is used to incorporate the identified CESSAR SER section into this SER. Any other chapter, section, table, or figure referenced in this SER refers to a chapter, section, table or figure of this SER. Each reference to a CESSAR SER section also incorporates all figures, tables, and appendices referred to in the incorporated CESSAR SER section. All topical reports referred to in the CESSAR SER section are also incorporated in the WNP-3 SER.

1.3 General Plant Description

The System 80 nuclear steam supply system, as described in the CESSAR SER, will incorporate a pressurized-water reactor and a two-loop reactor coolant system with a net electrical output of approximately 1240 MW. Each loop of the reactor coolant system will consist of an outlet pipe (hot leg), one steam generator, two inlet pipes (cold legs) and two reactor coolant pumps, one in each cold leg. An electrically heated pressurizer will be connected to one loop and will establish and maintain the reactor coolant pressure.

The nuclear steam supply system will be housed in a containment structure. The containment will consist of a steel containment vessel surrounded by a reinforced concrete shield building. The containment vessel is a cylindrical vessel with a hemispherical dome and ellipsoidal bottom. The shield building is a medium-leakage concrete structure that surrounds the steel containment vessel.

The steam and power conversion system will be designed to remove heat from the reactor coolant in the two steam generators and convert it to electrical energy. The excess heat removed by the condenser will be discharged to cooling towers through the circulating water system.

The plant will be capable of being supplied electrical power from two independent offsite power sources and will also be provided with an independent and redundant onsite emergency power supply system which is capable of supplying electric power to the engineered safety features.

Before an OL is issued, the staff will review the final design 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.

1.4 Comparison with Similar Facility Designs

A comparison of the CESSAR design with a similar facility design is presented in the CESSAR SER, Section 1.3.

1.5 Identification of Agents and Contractors

WPPSS will act as project manager and operating agent for the participants and will have full authority and responsibility to engineer, design, construct, operate, and maintain WNP-3.

The applicant has retained EBASCO to perform architectural, engineering, and construction services. Combustion Engineering has been contracted to design, manufacture, and deliver the nuclear steam supply system and nuclear fuel for the initial core. In addition, Combustion Engineering will furnish technical assistance for erection, initial fuel loading, testing, and initial startup of the nuclear steam supply system.

The turbine generators will be manufactured by Westinghouse. The applicant also will use consultants, as required, in specialized areas.

2 SITE CHARACTERISTICS

2.1 Geography and Demography

The July 1981 edition of "Standard Review Plan for the review of Safety Analysis Reports for Nuclear Power Plants," (SRP, NUREG-0800) includes Chapter 2, Site Characteristics. The Washington Nuclear Project 3 (WNP-3) was reviewed in accordance with Section 2.1.1, 2.1.2, and 2.1.3. The results of this review are contained in Section 2.1 of this Safety Evaluation Report.

2.1.1 Site Location and Description

The site for WNP-3, which was formerly proposed as a two unit site, consists of 2450 acres located in southeastern Grays Harbor-County in the western part of the state of Washington. Figure 2.1 shows the site location including the low population zone (LPZ), and the surrounding area within 4 miles. The principal plant features in conjunction with the exclusion area and site boundary lines (including land ownership) are shown in Figure 2.2. The area within 50 miles of the site is shown in Figure 2.3. Figure 2.4 shows the site location relative to various transportation routes and transmission lines. These include the Chehalis and Satsop rivers, the Union Pacific and Burlington Northern railroad lines, an eight inch natural gas pipeline, the BPA transmission lines, and U.S. Highway 12, as well as other local roads. As shown in Figures 2.1, 2.2, and 2.4, WNP-3 is located about 1.25 miles south of the Chehalis River near its junction with the Satsop River. The site is situated on a ridge above the Chehalis River in an area that is generally rural in nature and sparsely populated but extensively used for timber production, with some scattered farmlands in the fertile river bottom lands. The city of Olympia, the capitol of Washington, is about 26 miles east and the cities of Aberdeen-Hoquiam, Washington are approximately 20 miles west of the site. Tacoma, Washington is the largest populated area in the vicinity and is located 50 miles northeast. The coordinates of WNP-3 are 46° 57' 34" north latitude and 123° 28' 15" west longitude. The Universal Transverse Mercator (UTM) coordinates are 5,200,546 meters north and 464,176 meters east, in zone 10.

2.1.2 Exclusion Area Authority and Control

The exclusion area for the WNP-3 site is defined as a circular area with a 4300 foot (1310) radius measured from the center of the reactor containment building. The applicant owns 1120 acres of the site in which the exclusion area is located. The rest of the site area is owned by private corporations or individuals, but under the control of the Washington Public Power Supply System (WPPSS) by agreement with the owners. These agreements provide easements to the land. All of the mineral rights within the designated exclusion area will be either owned or leased by the applicant. Activities that occur within the exclusion area that are unrelated to plant operations are those associated with tree farming, and transmission line maintenance. Approximately 30 persons and associated temporary structures and facilities may be located within the exclusion area from time to time in order to conduct timber farming activities. These activities are subject to prior planning and approval by the applicant. In addition, by agreement with WPPSS, personnel from the Bonneville Power Administration (BPA) will be conducting maintenance in the BPA transmission corridor falling within the exclusion area boundary. There are no residents living within the exclusion area and there are no major highways, railways, or waterways traversing the area. However, there is some vehicular traffic on the access roads leading to the plant and on portions of the BPS corridor. Easements for these portions of the exclusion area are being negotiated with Grays Harbor County and the BPA. The applicant does not anticipate any problems in obtaining the necessary authorization for controlling this area. Keyes Road will be abandoned and the applicant expects to provide the county with an appropriate right-of-way to run another road through the site. The applicant is making arrangements to control and, if necessary, evacuate the exclusion area in the event of an emergency. Section 13.3 of this report will provide more details on these arrangements when they are finalized.

We conclude that when the negotiations to acquire the appropriate easements, and subsequently obtain full control are completed, the applicant will have the authority and will be able to control all activity unrelated to plant operations within the exclusion area as required by 10 CFR Part 100. We also conclude that the activities unrelated to plant operation within the exclusion area will not interfere with normal plant operation.

2.1.3 Population Distribution

The resident population in the vicinity of the WNP-3 site is shown as a function of distance in the table below. The year 2030 is the nearest census year to the end of plant life.

The closest resident lives about one mile (1609 meters) from the site. The nearest communities in the vicinity of the site with a population of more than 1000 persons are the cities of Elma, located about four miles northeast with a population of 2720 in 1980, and Montesano, which is about six miles west-northwest with a 1980 population of 3247. The closest large communities nearby are the cities of Aberdeen-Hoquiam, located approximately 20 miles west of the site. This combined area had a population of 35,170 in 1980. The population within five miles of the site in 1980 was 5867, and within ten miles it was 15,165. As indicated in Table 2.1, the population within five miles of the site is expected to increase by about 3600 persons during the life of the plant. The applicant reported that there were 514,954 people living within 50 miles of the site in 1980, and this number is expected to increase to 646,145 by 1990. By the year 2030 the population within 50 miles of WNP-3 is expected to reach 1,012,502. Olympia, the capitol of Washington, is about 26 miles east of the site and had a population of 27,447 in 1980. The largest city near the site is Tacoma, Washington, located about 50 miles northeast, with a 1980 population of 158,501. The applicant predicts a population growth rate of about 1.36% per year for the area within 50 miles of the site during the life of the plant as compared to a growth rate of about 0.87% per year predicted by the U.S. Bureau of Economic Analysis (BEA) for this area. The staff made an independent assessment of the population within 50 miles of the plant for the year 2030, based on Bureau of the Census data, and estimated a growth rate of about 0.80% per year. Comparing the staff's estimate and the BEA predictions indicates that the applicant's projections are conservative.

The applicant has designated a low population zone (LPZ) for the site which is a circular area with a three mile (4827 meter) radius measured from the midpoint of the centerline between Unit 3 and the previously planned Unit 5. Except for the Chehalis and Satsop Rivers, the LPZ consists mostly of wooded areas and

some agricultural land, much the same as the rest of the area in the general vicinity around the WNP-3 site.

There is very little transient population within the LPZ. There is some limited recreational activity on the rivers, including hunting and fishing, and there are several tree farming companies scattered throughout the area. According to the applicant, about 906 residents were living within the LPZ in 1980, and the population within the LPZ is expected to increase to about 1500 during the life of the plant. Except for the travelers using U.S. Highway 12, which is three miles north of the plant, going from the Puget Sound area to the Pacific Coast region, there is no significant amount of transient population within the ten miles surrounding the site. There are essentially no migrant workers within the ten mile radius around the plant. Logging operations vary from time to time throughout this area employing about ten persons per operation. On the average, 120 persons per year are employed in logging activities at twelve different locations. Also within the ten mile radius, there are four companies, generally associated with lumber products, that employ about 130 persons. During peak periods this total may rise to about 185. The applicant estimates that as many as 128 fishermen may be seen in the area on a good day, and during the hunting season a total of 2700 may utilize the woods and river tributaries around WNP-3. There are six schools within ten miles of the site. One is located in Satsop, two in Elma, and three in Montesano. Approximately 3550 students and staff occupy these facilities during the school year. Four nursing homes, the Elma youth home, and the county jail, all within ten miles of the plant, averaged about 356 persons in 1981. Although most of these people (fishermen, hunters, students, loggers and other employees) are transitory, some of them reside within the ten mile radius and are included in the population statistics given above.

Section 13.3 of this report provides a discussion of the Emergency Preparedness plans for protecting the public in this area. The nearest densely populated center, of about 25,000 or more persons, as defined by 10 CFR Part 100, is the combined area of Aberdeen-Hoquiam. The applicant has indicated that the center's nearest boundary is approximately 16 miles west of the site. This distance is at least one and one-third times the distance to the LPZ outer radius, as required by 10 CFR Part 100.

2.1.4 Conclusion

Pending the applicant's completion of all ownership and easement transactions, we conclude that the exclusion area, low population zone and population center distance meets the criteria of 10 CFR Part 100 and will be acceptable. This finding is based on the 10 CFR Part 100 definitions of the exclusion area, the low population zone and the population center distance, our analysis of the onsite meteorological data from which the relative concentration factors (X/Q) were calculated (see Section 2.3 of this report), and the calculated potential radiological dose consequences of design basis accidents (see Section 15.0 of this report).

2.2 Nearby Industrial, Transportation, and Military Facilities

2.2.1 Transportation Routes

There are no major highways, railroads, or waterways traversing the WNP-3 exclusion area. A transmission corridor owned by the Bonneville Power Administration (BPA) penetrates the exclusion area but is limited to BPA maintenance vehicles only. Public access to the exclusion area may be attained by means of Keyes Road and Workman Creek Road. These roads are used primarily for local traffic, including tree farming activities. The only major highway nearby is U.S. 12, a divided four-lane highway running in an east-west direction approximately three miles north of the plant. This highway is classified as a "scenic and recreational highway" used principally by motorists traveling between U.S. 1 which runs in a north-south direction along the Pacific Ocean, and I-5 which also runs north-south in the Puget Sound area. Because of the separation distance involved and the topography of the area, accidents occurring on U.S. 12 that may present a hazardous materials problem do not pose a threat to the safe operation of the plant.

A single railroad track maintained and used by the Union Pacific Railroad, and a rail line owned by the Burlington Northern Railroad are located about 1.25 miles and three miles north of the plant, respectively. Both rail lines run parallel to the Chehalis River and U.S. 12 in a valley at an elevation about

350 feet below and separated from WNP-3 by dense forest which is prevalent in the area. Rail traffic along the closest rail line averages about two trains daily, involving mostly shipments of lumber and related products and infrequently shipments of caustic soda, chlorine and propane. The applicant has identified several types of potential accidents involving hazardous material shipments on the closest railroad line (7000 feet) to the plant. The accidents include: (a) the detonation of a box car loaded with 132,000 lbs of TNT, (b) the rupture, spill, and ignition of a tank car filled with 85 tons of propane, and (c) the rupture and spill of a tank car loaded with 90 tons of liquid chlorine. The above events were assumed to occur on the railroad at a point nearest to the plant.

With respect to TNT, the applicant's analysis indicates that the overpressure from detonating 132,000 lbs of TNT at 7000 feet from the plant will not exceed the criteria in Regulatory Guide 1.91. Our review supports this finding and we concur with the applicant that TNT explosions on the railroad do not pose a significant hazard to the plant.

The applicant's analysis of an 85 ton propane tank car spill indicates that flammable concentrations would not reach the plant area. Since the railroad is about 350 feet below plant grade elevation, the heavier than air propane cloud would not tend to flow toward the plant. Detonation of 85 tons of propane at the railroad would produce a peak reflected overpressure of less than 1.0 psi at the plant. Hence we find that the potential propane tank car accident does not pose a significant hazard to the plant.

Potential chlorine spills on the railroad have led the applicant to provide chlorine detectors for the control room. A more detailed discussion of chlorine protection is given in Section 6.4 of this report.

In view of the above, the staff concludes that the potential risks associated with the postulated accidents on the railroad line near the WNP-3 site are sufficiently low and do not exceed the criteria specified in Regulatory Guides 1.91 and 1.78.

In addition to the above, the applicant also has identified a number of hazardous gas sources on site. Two of these have some potential for control room operator incapacitation. Specifically, there is a 15000 gallon tank of aqueous ammonia, and four 1 ton tanks of liquid sulfur dioxide. The staff is currently reviewing the control room habitability system with respect to these gases, and is currently considering this to be an open item.

The Chehalis River, located about 1.25 miles from the plant, and just north of the Union Pacific track, is used principally by small pleasure and fishing boats. Because the river is not commercially navigable in this area (river mile 21), making barge traffic impractical, hazards to the plant from accidents on the river are non-existent. River traffic does not pose a hazard to the intake structure, because make-up water is supplied by a Ranney well collection system (which obtains water from about 80 feet below the surface of the river), and there is no potential for highway or railroad chemical spills flowing into the river and entering the make-up water system.

2.2.2 Nearby Facilities

There is no extensive industrial activity around the WNP-3 site. The Thiokol Corporation, located about five miles east-northeast of the plant, stores materials such as methanol, metallic sodium, and nitrogen, which are used in the production of a bleaching agent for the pulp and paper industry. The Western Washington Paper Co. supplies gases for industrial and medical applications. The gases, stored five miles northeast of the site, include acetylene, hydrogen, propane, argon, helium and nitrogen. There are also several gasoline and diesel fuel distributors that have storage facilities about five miles from the site. Tree farming is the main industry in this area and there is no significant amount of hazardous material involved in this type of operation. Because of the quantities and distances involved, the materials stored at these locations do not pose a hazard to the plant. There are no large industrial expansion programs planned for this area in the foreseeable future.

There are no military bases, bombing ranges, munitions plants, missile installations, or major airports within the general vicinity of the WNP-3 site. There are two grass airstrips and a small airport within five miles of WNP-3.

One of the airstrips, which hasn't been used in years, is located 1.2 miles north-northwest. The other is about 4.2 miles east-northeast of the site and is used very infrequently. The airstrips are intended for private use, usually crop dusting. Elma airport, the closest active airport in the area, is located three miles northeast of the site. It has a 2100 foot paved runway and can accommodate light single and twin-engine fixed wing aircraft and helicopters that weigh under 12,500 pounds. The airport was formerly unattended, so no records were kept on the usage it had. It was estimated that there were between 4000 and 12000 take-off and landing operations in 1980. A minor expansion program has been proposed for the Elma airport. The largest airport near the plant is Bowerman Field and it is located about 22 miles west. Although there were no scheduled commercial flights prior to 1980, commuter service is proposed and it is projected that there will be about 125,000 operations at this airport by the year 2000.

Two low-level Federal Airways (V-204 and V-27) are located in the airspace around WMP-3. There are about five commercial flights per day on route V-204 which pass over the site at a minimum altitude of 4500 feet MSL. Approximately seven commercial flights per day use route V-27 which passes within eleven miles of the plant at a minimum altitude of 3200 feet MSL. Ft. Lewis Army Base and McChord Air Force Base are located 42 and 48 miles east-northeast of the site, respectively. There are one or two military helicopter flights per month that use the V-204 route.

Aircraft from McChord AFB (about 280 flights per year) use a predetermined route which passes within four miles of the plant. No ordinance is carried on the military aircraft flying in this area. The applicant is analyzing the effects on plant structures assuming an aircraft crashes into WNP-3. The staff will review the analysis when it is submitted and evaluate the potential risks to the safe operation of the plant.

No LPG or LNG lines are located within five miles of the plant, but there is one natural gas pipeline nearby. It is an eight inch line that is buried at a minimum depth of 30 inches and has a maximum operating pressure of 305 psig. This line runs in an east-west direction about five miles north of the site.

Two lines, four inches in diameter, are tapped off of the main eight inch gas line and serve the towns of Elma and Montesano. There are no plans to transport other products in these pipelines or to up-grade the system. Because of the distance involved these gas lines do not present a hazard to the WNP-3 facilities.

2.2.3 Conclusion Regarding the Evaluation of Potential Accidents

On the basis of the information provided by the applicant, and our review based upon criteria in 10 CFR Part 50, Appendix A, GDC 4, and in Standard Review Plan Section 2.2.3, we have determined, subject to a satisfactory conclusion pending our evaluation of the applicant's aircraft analysis, that the WNP-3 site will be adequately protected and can be operated with an acceptable degree of safety considering the activities at nearby transportation, industrial, and military facilities.

2.3 Meteorology

Evaluation of regional and local climatological information, including extremes of 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. Information concerning atmospheric diffusion characteristics of a 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. Sections 2.3.1 through 2.3.5 have been prepared in accordance with the review procedures described in the Standard Review Plan (NUREG-0800), utilizing information presented in Section 2.3, of the FSAR, responses to requests for additional information, and generally available reference materials as described in the appropriate sections of the Standard Review Plan.

2.3.1 Regional Climatology

The plant is located in southwest Washington on a ridge near the eastern shore of the Pacific Ocean and in a maritime type of climate.

Maritime air masses dominate the region year round. The mean annual temperature in the area is about 10°C (50°F) ranging from about 2.9°C (37.2°F) in January to about 17.6°C (63.6°F) in July. Annual precipitation in the area is about 1270 mm (50 inches).

The movement of weather systems from the Pacific Ocean over the site maintains considerable cloudiness and nearly constant rainfall due to lifting of this moist air over the mountains in this area of Washington. Severe weather phenomena which affect the site area include about 5 thunderstorms per year which can be expected on about 5 days each year. Thunderstorms occur primarily in spring and summer. Considering the small frequency of thunderstorms, the applicant has estimated the number of lightning strikes to the square kilometer containing the plant to be 1.1 per year. Hail does not usually occur with the thunderstorms and is not a significant phenomena.

Tornadoes are not common in the region. For a two degree latitude-longitude "square", 33127 square kilometers (12791 miles^2) containing the site, 5 tornadoes were reported for the period 1954-1981. Using an observed tornado path area of .088 square Km (.034 sq. miles), the computed probability of occurrence for a tornado at the plant site is about 4.9×10^{-7} per year. The applicant has followed the recommendations of Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants," for this region of the country. The applicant's design basis tornado for category I structures has a 107m/s (240 mph), rotational velocity with a translational velocity of 27m/s (60 mph), a total pressure drop of 2.25 psi and an average rate of pressure drop of 1.2 psi in 1 second. The tornado winds and total pressure drop are consistent with Regulatory Guide 1.76 for the site area.

High wind speed occurrences in the area are usually associated with severe thunderstorms and intense extratropical cyclones. The highest "fastest mile" wind speed reported at Olympia, WA was 27m/s (60 mph) in November 1958. The applicant has selected an operating basis wind speed (defined as the "fastest mile" wind speed at a height of 9.1m (30 feet) with a return period of 100 years) to be 46.9 m/s (105 mph) for consideration in plant design.

Since the ultimate heat sink for the plant is a dry cooling tower, meteorological conditions related to extreme temperature and wind speed are relevant to determination of the adequacy of the tower to perform its function for a 30 day period. A maximum hourly temperature of 38.6°C (101.5°F) was used for design of the plant ultimate heat sink, dry cooling towers. This temperature was statistically determined to be the 40 year maximum hourly average temperature by correlating short term site data with the same period at Elma, WA. Based upon this statistical analysis the maximum hourly average temperature in 40 years at Elma, WA of 40.6°C (105°F) was converted to a maximum hourly average temperature onsite for the 40 year period of 38.6°C (101.5°C). This estimate of the long-term maximum temperature is reasonable and is therefore a conservative basis for the design of the dry cooling towers thereby satisfying the intent of Regulatory Guide 1.27 for 30 day cooling capability. Heavy snowfall is not common in the region; roof loads may accumulate due to a wintertime precipitation comprised primarily of a mixture of snow and rain. Maximum monthly snowfall observed at Olympia was 526 mm (20.5 inches) in January 1972 which was also the maximum snowfall in a 24-hour period. Ice storms, which can plug drains and scuppers as well as disrupt offsite power, are relatively infrequent. The estimate of the snowpack based on ANSI 58.1-1982, extrapolated from the 50-year return period in the standard to a 100-year return period, produces a weight of less than 15 psf. This snowpack weight, when added to the weight produced by the 48-hour probable maximum winter precipitation produces a load of less than the design snowload of 80 psf, which was utilized in the design combined snow and ice load of Category I structures. During the 5 year period 1960-1964, about 41 atmospheric stagnation cases defined as persisting for two days or longer totaling at least 164 days were reported in the area.

As discussed above, the staff has reviewed available information relative to the regional meteorological conditions of importance to the safe design and siting of this plant in accordance with the criteria contained in Section 2.3.1 of the Standard Review Plan. Based on this review, the staff concludes that the applicant has identified appropriate regional meteorological conditions for consideration in the design and siting of this plant. The applicant has met the requirements of 10 CFR Part 100.10 and 10 CFR Part 50, Appendix A, General Design Criterion 2. The design basis tornado characteristics selected by the

applicant conform to the position set forth in Regulatory Guide 1.76, and, therefore, meet the requirement of 10 CFR Part 50, Appendix A, General Design Criterion 4 to determine an acceptable design basis tornado for missile generation.

2.3.2 Local Meteorology

Climatological data from Olympia, WA, and nearby climatological cooperative stations and available onsite data have been used to assess local meteorological characteristics of the plant site.

Extreme temperatures of -22.2°C (-8°F) and 40.6°C (105°F) have been reported in the area. The applicant has considered a summer outdoor design temperature of 28.3°C (83°F) and a winter minimum temperature of 4.4°C (40°F) in the design of all heating, ventilation and air conditioning (HVAC) systems to maintain a 23.9°C (75°F) control room area temperature during normal or accident conditions. Regional analyses in NUREG/CR-1390, "Probability Estimates of Temperature Extremes for the Contiguous United States" show that an ambient temperature of 35°C (95°F) will be exceeded for at least one hour every two years, on the average, and that an ambient temperature of about 42.2°C (108°F) will be exceeded at least one hour every 100 years, on the average. Also, an ambient temperature of less than -8.9°C (16°F) is expected to occur for at least one hour every two years, on the average, and an ambient temperature of less than -22°C (-8°F) is expected to occur for at least one hour every 100 years, on the average. Further justification of the adequacy of the ambient extreme temperatures considered by the applicant for the design of HVAC systems protecting safety-related auxiliary systems and components is required. This will be an open issue only if exceedence of extreme design temperatures for the HVAC system results in failure or malfunction of Category I auxiliary systems and components, which is being evaluated by the Auxiliary Systems Branch.

Precipitation is observed throughout the year, ranging from over 203 mm (8 inches) in December to less than 25 mm (1 inch) in July. Maximum and minimum monthly amounts of precipitation observed at Olympia have been 504 mm (19.8 inches) in January 1953 and 0 mm (0 inches) in August 1946, respectively. The maximum amount of precipitation in a 24 hour period at Olympia was 125 mm (4.93 inches) in February 1951.

Average annual precipitation at Olympia is about 1290 mm (51 inches) and onsite precipitation measurements for the 2 year period 1970-1981 presented by the applicant indicate annual precipitation of about 1600 mm (63 inches). These differences can be attributed to the different periods of record and primarily terrain differences between the two locations.

Wind data taken from the 10 meter level of the onsite meteorological tower for a 2 year period (October 1979 - September 1981), as summarized by the applicant, indicate prevailing winds from the southwest (20%) with a secondary peak frequency from the northeast (8.0%). The mean annual wind speed observed at the 10 meter level of the onsite meteorological tower for the period 1970-1981 was about 1.6 m/sec (4 mph), with calm conditions (defined as wind speeds less than the starting threshold of the anemometer) occurring almost 9.4% of the time.

Atmospheric stability assessments, based on vertical temperature difference measurements for the 2 year period (1979-1981), have been summarized by the applicant for the 10 meter (30 feet) - 60 meter (197 feet) layer. Unstable conditions (indicating rapid diffusion rates) do not occur very frequently. Neutral and stable conditions predominate and occur 99% of the time, reflecting the cloudy and rainy conditions existing in the region.

As discussed above, the staff has reviewed available information relative to local meteorological conditions of importance to the safe design and siting of this plant in accordance with the criteria contained in Section 2.3.2 of the Standard Review Plan. The staff concludes that, with the exception of the design basis temperatures for HVAC systems, the applicant has identified and considered appropriate local meteorological conditions in the design and siting of this plant, and, therefore, meets the requirements of 10 CFR Part 100.10 and 10 CFR Part 50, Appendix A, General Design Criterion 2.

2.3.3 Onsite Meteorological Measurements Program

The onsite pre-operational meteorological measurements program was initiated at the WNP-3 site in 1973 and ended in 1975. They re-commenced in 1979 and ended in 1981. Measurements were made on a tower extending 60 meters (197 feet) above grade. The tower is located about 1207 (3/4 mile) northwest of the

plant structures. The following meteorological measurements were made on the tower: wind speed and direction at the 10 meters, and 60 meters levels; vertical temperature gradient between the 60 meters and 20 meter levels. Ambient temperature was measured at the 10 meter level and dewpoint at the 60 meters level. Precipitation was measured at ground level near the tower.

A digital data acquisition system, backed up by analog strip charts, was used to record meteorological data. Daily checks and quarterly calibration were done on the equipment. The joint data recovery for wind speed, and wind direction at the 10 meter level, and atmospheric stability (defined by the vertical temperature difference between the 60 meter and 10 meter levels for the 2 year period October 1979 - September 1981 presented in the FSAR was in excess of 95%.

The meteorological measurements system complies with the accuracy specifications in Regulatory Guide 1.23, "Onsite Meteorologial Programs." The representativeness of the 2 year period of onsite data to long term conditions was determined by comparisons of data from the site to measurements at Olympia and nearby locations. These comparisons indicate that reasonable estimates of atmospheric dispersion for accidental and routine releases of radioactive effluents can be made from the onsite data record.

The meteorological measurements and data collection program has been terminated after the decision to postpone or possibly cancel completion of Units 3 and 5 was made by WPPSS.

If the project is resumed, the meteorological program on the 60 meter tower will be re-activated and will serve as the operational meteorological monitoring system as well as for emergency preparedness requirements. The meteorological measurements will be available in the control room as well as in the emergency operation facility (EOF) and technical support center (TSC).

The meteorological program described above meets the criteria for meteorological measurements during plant operation and as part of the emergency response capability. Any meteorology measurement upgrades must be completed in accordance with the schedule of NUREG-0737, III.A.2, "Clarification of TMI Action Plan Requirements," and its supplement, and a post implementation staff review will

be conducted. The incorporation of current meteorological information into a real-time atmospheric dispersion model for dose assessments will also be considered as part of the upgraded capability.

The staff has reviewed the onsite meteorological measurements system in accordance with the criteria contained in Section 2.3.3 of the Standard Review Plan. The meteorological measurements program has provided data to represent onsite meteorological conditions as required in 10 CFR Part 100.10. The staff concludes that the historical site data provide a reasonable basis for making assessments of atmospheric dispersion conditions for estimating consequences of design basis accident and routine releases from the plant.

2.3.4 Short-Term (Accident) Diffusion Estimates

To audit the applicant's assessments, the staff has performed an independent assessment of short-term (less than 30 days) accidental releases from buildings and vents using the direction-dependent atmospheric dispersion model described in Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," with consideration of increased lateral dispersion during stable conditions accompanied by low wind speeds. Two years (October 1979 - September 1981) of onsite data available to the staff on magnetic tape, which had 95% data recovery, were used for this evaluation. Wind speed and wind direction data were based on measurements at the 10 meter level and atmospheric stability was defined by the vertical temperature gradient measured between the 60 meter and 10 meter levels. A ground-level release with a building wake factor, cA, of 2133 meter^2 was assumed. The relative concentration (X/Q) for the 0-2 hour time period was determined to be 4.1×10^{-4} sec/m^3 at an exclusion area boundary distance of 1311 meter (.81 miles) in the south sector. The X/Q values for appropriate time periods at the outer boundary of the low population zone 4827 meter (3 miles) are:

Time Period	X/Q (sec/m^3)
0-8 hours	6.0×10^{-5}
8-24 hours	4.0×10^{-5}
1-4 days	1.6×10^{-5}
4-30 days	4.3×10^{-6}

The applicant has calculated a higher X/Q value for the 0-2 hour time period at the exclusion area and low population zone boundary than that calculated by the staff. These differences may be attributed to slightly different use of the models and the data by the staff and the applicant.

Based on the above staff and applicant evaluations performed in accordance with the criteria contained in Section 2.3.4 of the Standard Review Plan, the staff concludes that the applicant has conservatively considered atmospheric dispersion conditions at the exclusion area and low population zone boundaries for assessments of the consequences of radioactive releases for design basis accidents in accordance with the requirements of 10 CFR Part 100.11. The atmospheric dispersion estimates provided above which were independently calculated by the staff have been used by the staff in an independent assessment of the consequences of radioactive releases for design basis accidents.

2.3.5 Long-Term (Routine) Diffusion Estimates

To audit the applicant's estimates, the staff performed an independent calculation of annual average relative concentration (X/Q) and relative deposition (D/Q) values.

Annual average relative concentration (X/Q) and relative deposition (D/Q) values at specific receptor points and in arrays to 80 Km (50 miles) for use in population dose assessment were based on the straight-line gaussian atmospheric dispersion model, described in Regulatory Guide 1.111, modified to reflect spatial and temporal variations in airflow as described in NUREG/CR 2919. Continuous and periodic releases through the plant vents were considered as ground level releases. The results of the evaluation were published in the DES.

The staff compared its long-term diffusion estimates with those provided by the applicant and determined the analyses were in general agreement. The staff concludes that the applicant has considered representative atmospheric dispersion estimates for demonstrating compliance with the numerical guides for doses contained in 10 CFR Part 50, Appendix I.

2.4 Hydrologic Engineering

The staff has reviewed the hydrologic engineering aspects of the applicant's design, design criteria, and design bases for safety-related facilities at the Washington Public Power Supply System Nuclear Project No. 3 (WNP-3). The acceptance criteria used as a basis for staff evaluations are set forth in SRP 2.4.1 through 2.4.14 (NUREG-0800). These acceptance criteria include the applicable GDC reactor site criteria (10 CFR 100), and standards for protection against radiation (10 CFR 20, Appendix B, Table II). Guidelines for implementation of the requirements of the acceptance criteria are provided in RGs, ANSI standards, and Branch Technical Positions (BTPs) identified in SRP 2.4.1 through 2.4.14. Conformance to the acceptance criteria provides the bases for concluding that the site and facilities meet the requirements of 10 CFR 20, 50, and 100 with respect to hydrologic engineering.

2.4.1 Hydrologic Description

WNP-3 is located in Satsop, Washington, approximately 1.4 miles south of the Chehalis River near the confluence of the Satsop River. The site is about 26 miles west of Olympia and about 16 miles east of Aberdeen, Washington. As shown on Figure 2.4.1, WNP-3 is situated on a ridge between Workman Creek and the Chehalis River.

The Chehalis River which heads in the Willapa hills in southwest Washington, flows generally eastward to the city of Chehalis where it changes its course abruptly to the north. About 10 miles north of Chehalis, near Grand Mound, the river flows northwesterly to Elma, then west to Grays Harbor at Aberdeen. The river and its tributaries have a drainage area of about 2,115 mi^2. The drainage area at the site, including the Satsop River, is about 1,765 mi^2. The average annual flow at this location is about 6,820 cfs. The Chehalis River basin is shown on Figure 2.4.2.

The major tributaries of the Chehalis River in the vicinity of the site are the Satsop and Wynoochee Rivers. The Satsop River has a drainage area of about 300 mi^2 and an average annual flow of about 2030 cfs. The Wynoochee River has a drainage area of about 100 mi^2 and an average annual flow of about 1200 cfs.

Both of these tributaries rise on the south side of the Olympic Mountains and flow southward to their confluences with the Chehalis River. A number of small tributaries to the south of the Chehalis River head in the hills surrounding the site. These include: Elizabeth Creek, Hyatt Creek, Fuller Creek, Purgatory Creek and Workman Creek. All of these streams are relatively short, intermittant streams, originating at elevations between 300 to 400 feet above mean sea level (MSL).

As shown on Figure 2.4.2 there are two dams and associated reservoirs on the tributaries of the Chehalis River. The Wynoochee dam and lake, which is a Corps of Engineers project, provides water supply for industry and agriculture and storage for flood control. The lake also offers recreation opportunities for the public. The Skookumchuck dam and reservoir project is operated by the Pacific Power and Light Company (PP&Ls). The reservoir provides make-up water for PP&L's Centralia Steam Electric Station.

The water resources of the Chehalis River valley include both surface and ground supplies. Within 5 miles of the plant, surface water permits have been granted by the Washington State Department of Ecology to about 78 users. Most surface water is used for irrigation, with the remainder for domestic use, livestock watering, fish propagation, fire protection and industrial use. Except for a single domestic water user located within a mile downstream of the plant, there are no other known users of Chehalis River water for domestic purposes between the plant and Grays Harbor.

Groundwater in the Chehalis River valley is obtained from shallow wells which tap the alluvial aquifer and is used mostly for drinking and irrigation. There are 45 known wells within 2 miles of the plant. Five major municipal water systems within 20 miles of the site are served partially or totally by groundwater.

The applicant has provided hydrologic descriptions of the plant site and vicinity. The staff has reviewed the applicant's information in accordance with procedures in SRP 2.4.1. The staff concludes that the requirements of GDC 2 and 10 CFR Part 100, with respect to general hydrologic descriptions, have been met.

2.4.2 Floods

2.4.2.1 Flood Design Considerations

Five potential sources of site flooding were considered by the applicant:

(1) intense local precipitation on the plant yard;
(2) floods on the Chehalis River;
(3) dam failures;
(4) surges and seiches;
(5) tsunamis.

The staff has reviewed the material presented by the applicant in accordance with procedures in SRP 2.4.2 and concludes that in addition to these five flooding sources, other sources of potential flooding of the plant site are the small creeks near the site. Since the applicant did not address potential flooding from these small creeks, the staff made an independent evaluation as described in Section 2.4.3.

2.4.2.2 Effects of Local Intense Precipitation

At WNP-3, a site drainage system consisting of catch basins, drain pipes, and ditches has been provided to carry surface runoff south to Workman Creek and north to the Chehalis River via Fuller and Purgatory Creeks. The drainage system is designed for a 100 year recurrence storm with pipes flowing half full. The intensity of this storm is 2.9 inches per hour. This is less than the probable maximum precipitation (PMP) so during a PMP event, some water could pond on the site.

PMP is the estimated depth of precipitation (rainfall) for which there is virtually no risk of exceeding. At WNP-3, the PMP values used by the applicant for durations of 1 through 6, 12, 24, and 48 hours are as follows:

Duration (hours)	Incremental PMP (inches)	Total PMP (inches)
1	4.32	4.32
2	1.92	6.24
3	2.24	8.48
6	6.24	14.72
12	5.44	20.26
24	5.12	25.28
48	3.00	28.28

These PMP values were determined from Technical Paper 40 (U.S. Weather Bureau, 1963). Although Technical Paper 40 does not present PMP estimates, it does give a method for determining PMP values from 100 year rainfall values which are given in the paper. The applicant states that the 100 year rainfall values used to obtain the PMP values above, are more conservative than those given in NOAA Atlas No. 2, Volume IX-Washington (NOAA, 1973). In addition, these PMP values are also more conservative than those given in Hydrometeorological Report 43 (U.S. Weather Bureau, 1966).

The staff has reviewed the three PMP references used by the applicant and agrees that Technical Paper 40 results in more conservative PMP estimates than the other two references. The staff concludes that the PMP values used by the applicant are appropriate for the evaluation of site and roof drainage.

The minimum elevation of openings to safety-related structures is 390.5 feet MSL. The elevation of the paved road that surrounds the plant is 390.0 feet and the grassed areas within this road are at elevation 389.5 feet MSL. Because the grassed areas are 1 feet lower than openings to safety-related structures, water could pond 1 feet deep in these areas before any structure or equipment would be affected.

Using the PMP rainfall values tabulated above, the applicant estimated the maximum depth of ponding in the grassed areas would be 2.0 inches assuming that the site drainage system is functioning as designed. The applicant states in the FSAR, that in the unlikely event that any drain is completely blocked, rainwater could pond to elevation 390 feet MSL, which is the elevation of the

crown of the surrounding access road. Subsequent runoff would overflow the road to lower-lying areas. Since exterior door openings are at elevation 390.5 feet MSL, the applicant concluded that water from local intense precipitation would not enter safety-related buildings.

The staff has reviewed the material presented by the applicant and concludes that the applicant has not provided sufficient information to support its conclusion that local floods will not enter safety-related buildings. From topographic maps provided by the applicant, it appears that because the areas to the south, east and west slope upward from the plant; water could pond to a higher elevation than the top of the access road, possibly entering safety-related structures.

The staff will require that the applicant identify the areas where water will pond before overflowing the access road and the areas where water will drain to once it overflows the road. In identifying these areas, obstructions to flow such as temporary and permanent buildings, trailers, sheds, etc., should also be shown. It should also be assumed that all of the site drainage system is blocked. Once flow areas have been identified, the applicant should provide assurances that the flow areas are of sufficient capacity to prevent water from ponding to excessive levels. The applicant should also discuss whether the fence which surrounds the site will adversely obstruct flows.

In discussing the effects of local intense precipitation on roofs of safety-related buildings, the applicant stated that the roof drainage system is designed for a 100-year recurrence storm at 50 percent capacity. With the exception of a roof section adjacent to the steam tunnel, safety-related buildings have parapets that are 12 inches above the roof high points and 18 inches above the low points. Thus the average depth of ponding would be about 15 inches. The applicant states that roofs are designed to support the load induced by up to 15.4 inches of water.

The roof section adjacent to the steam tunnel at elevation 417.5 feet MSL is surrounded by walls to elevation 443.5 feet MSL or higher. The applicant stated that this roof is capable of supporting the load induced by a 48 hour PMP plus the 100-year snow pack. This is equal to 28.28 inches of rainfall plus a snow-depth water equivalent of 3.78 inches or 32.06 inches.

The applicant does not state whether the surrounding walls are equipped with scuppers or other means of limiting water depths. Therefore, it is possible that water could pond to a much greater depth than 32.06 inches because the walls are at least 26 feet high (443.5 feet-417.5 feet). The applicant should thus consider rainfall for durations greater than 48 hours. The effect of a 26 foot depth of water should be addressed unless it can be demonstrated that water cannot physically pond to this depth. Alternately, the applicant should consider putting scuppers (or other devices) in walls to limit water depths to those that can be supported safely by the roof section adjacent to the steam tunnel.

The staff has reviewed the material presented by the applicant in the FSAR, using the procedures described in SRP Section 2.4.2. Based on this review, the staff concludes that the applicant has not provided sufficient information to support its conclusions that ponded water will not enter safety-related buildings or that the roof section adjacent to the steam tunnel is capable of supporting potential rainfall loads. Thus the staff cannot conclude at this time, that the plant meets the requirements of GDC 2 with respect to flooding by intense local precipitation.

The staff, however, does conclude that during a PMP event, water levels on roofs of safety-related structures will remain at or below the levels determined by the applicant except for the roof adjacent to the steam tunnel.

2.4.3 Probable Maximum Flood on Streams and Rivers

The Probable Maximum Flood (PMF) is defined as the hypothetical precipitation-induced flood that is considered to be the most severe reasonably possible.

Severe rainfall storms in western Washington occur mostly in the winter months when there is snow on the ground. Consequently, the applicant estimated the PMF for the Chehalis River based on PMP and snowmelt over the drainage basin.

As a first step in estimating the PMF, the applicant subdivided the Chehalis River drainage basin into three subbasins and developed individual unit

hydrographs for each subbasin using Corps of Engineers procedures. A flood hydrograph was then developed for each subbasin using the Corps of Engineers flood hydrograph computer program, HEC-1. To determine a PMF for each subbasin, each hydrograph was then increased by the average annual river flow to account for base flow conditions. The three individual PMF's were then routed, where appropriate, and combined at the site to form a single PMF. To account for potential antecedent flood conditions as recommended in RG 1.59, the applicant assumed that a storm equal to 50 percent of the PMP would occur three days prior to the PMP storm. The antecedent flood resulting from these conditions was combined with the PMF at the site. The resultant hydrograph had a peak discharge of 353,000 cfs.

PMF water levels in the Chehalis River adjacent to the site were determined by means of the Corps of Engineers HEC-2, "Water Surface Profiles" computer program. The PMF discharge (353,000 cfs) stillwater level was estimated to be 53.1 feet MSL. The applicant determined that coincident wind-wave activity would result in a maximum wave runup, including wind setup, of 23.1 feet. Adding this to the PMF stillwater level resulted in a maximum flood level of 76.2 feet MSL. Since the plant is at elevation 390 feet MSL, the applicant concluded that no safety-related structures would be affected by a PMF on the Chehalis River.

The applicant did not address the potential for flooding from the small creeks near the site; however, based upon the following observations, the staff concluded that floods on these small creeks will not affect the safety of the plant.

Workman Creek which runs south of the plant has a stream bed elevation which is more than 200 feet lower than the plant grade elevation. Because the drainage area of this creek is small, less than 10 miles^2, the staff concludes that a PMF would not rise 200 feet in the creek. In addition to Workman Creek, there are two other creeks in the vicinity, Fuller Creek and Purgatory Creek. Both of these have drainage areas of less than one mile^2 and each creek flows away from the plant. The staff thus concludes that floods on these creeks will not affect the safe operation of WNP-3.

At the CP stage, the staff reviewed the applicant's analyses and the effects of coincident wind-wave activities. The staff concurred then with the applicant's analyses and concluded that there is no potential danger to safety-related structures due to the PMF with coincident waves. The staff has reviewed the FSAR material presented by the applicant in accordance with procedures described in SRP 2.4.2 and 2.4.3. Based on this review, the staff concludes that the plant meets the guidelines of RG 1.59, "Design Basis Floods for Nuclear Power Plants", and the requirements of GDC 2 with respect to flooding from the Chehalis River and the small creeks adjacent to the site.

2.4.4 Potential Dam Failures

As shown in Figure 2.4.2, the only dam located upstream of the plant is Skookumchuck Dam. The applicant estimated the effect of a failure of this dam coincident with a Standard Project Flood (SPF) equal to one half of the PMF. It was assumed that the reservoir would be full at the time the dam failed. Using Corps of Engineers procedures, the applicant estimated a dam failure hydrograph with a peak discharge of 260,000 cfs. This hydrograph was routed downstream and combined with SPF's from the Chehalis River sub-basins. The resultant peak flow at the site was determined to be 182,000 cfs. Using the same procedures as were used to determine PMF levels, the applicant determined the flood level at the site due to failure of Skookumchuck dam would be 39.6 feet MSL. Since this elevation is less than the PMF level, the applicant concluded that failure of Skookumchuck Dam will not affect the safety of the plant.

The staff reviewed the applicant's dam failure analysis at the CP stage and concluded that there would be no flooding of the plant due to dam failures. The staff has reviewed Section 2.4.4 of the FSAR, using the procedures described in SRP 2.4.4. The staff concurs that conservative procedures have been used and that potential dam failures pose no threat to the plant. Thus the staff concludes that the plant meets the requirements of GDC 2 and 10 CFR 100, Appendix A, with respect to flooding by dam failures.

2.4.5 Probable Maximum Surge and Seiche Flooding

There are no historical seiche records at Grays Harbor, the largest body of water near the site, and the probability of seiche occurrence at Grays Harbor is extremely remote due to the shallow tidal flats. The plant site located at River Mile 21 and elevation 390 feet MSL is not susceptible to surge or seiche flooding.

2.4.6 Probable Maximum Tsunami Flooding

WNP-3 is located about 30 miles inland from the Pacific Ocean with all safety-related equipment at an elevation of 390 feet MSL or higher. The maximum historical tsunami wave height originating within 1000 miles of the site was 32.8 feet and occurred in Cook Inlet, Alaska, in 1901. The most damaging tsunami of local origin, for the Washington coast area, was generated by the Alaskan Earthquake of 1964 (epicenter more than 1000 miles from the site), and caused minor damage at the Ocean Shores Development which is located on the spit that protects the Grays Harbor entrance. The elevation of the highest wave was 13.3 feet MSL; causing a break in the sand dune dike and the deposition of winter storm debris along the spit. There was no overtopping of the spit and no flooding resulted at Aberdeen.

The applicant estimated that a probable maximum tsunami (PMT) approaching the Chehalis River through Grays Harbor would result in only a 3.5 foot increase in water level at the mouth of the Chehalis River. The effects of the PMT would be reduced to negligible heights at the plant site because of attenuation in the river channel.

The staff has reviewed the applicant's tabulation of historical tsunami and its estimate of the PMT at the site. Based on procedures in SRP 2.4.6, the staff concludes that no credible tsunami event could threaten the plant. Thus, the requirements of GDC 2 as it relates to structures, systems and components important to safety being designed to withstand the effects of tsunami, have been met.

2.4.7 Ice Effects

The Pacific ocean, which is about 30 miles west of the site, greatly influences the climate in the WNP-3 area. The ocean acts to moderate the seasonal and daily variability in climate throughout the year such that winters are warmer than at other locations at similar latitudes. Because of this, there are no conditions which might produce a permanent ice cover or ice jam on the Chehalis River. In addition, because of the large difference in elevation between the Chehalis River and the plant (Section 2.4.3), even if ice jams did form, floods resulting from these jams would not affect the safe operation of the plant.

Water required for normal operation of WNP-3 will be supplied from groundwater infiltration-type structures (Ranney Wells). Therefore, potential icing will not affect the normal plant water supply. Emergency safe shutdown and cool down of WNP-3 can be accomplished using the ultimate heat sink which consists of dry cooling towers located adjacent to the reactor auxiliary building. Make-up water is not required for the dry cooling towers and during periods of low temperature the design of the towers prevents freezing of the tower or pipelines.

The staff has reviewed the information provided by the applicant concerning ice effects, in accordance with procedures in SRP 2.4.7. The staff concludes that icing will not affect the safe operation of the plant.

2.4.8 Cooling Water Canals and Reservoirs

There are no safety-related or other cooling water canals or reservoirs associated with WNP-3.

2.4.9 Channel Diversions

The Chehalis is a meandering river that shows a number of former channel locations, oxbows and sloughs in the vicinity of the plant (see Figure 2.4.1).

As described in Section 2.4.11.1, the source of makeup water for the WNP-3 cooling tower is the alluvial aquifer that underlies the Chehalis River flood-plain. Recharge to the aquifer occurs all across the river valley as well as

in the river channel and from the 70 to 80 inches of annual rainfall in the area. The aquifer reacts much like a reservoir by accepting and storing surface inflow during periods of high river flows and high rainfall and releasing the stored water when rainfall and river flows aren't as plentiful. Since the aquifer is recharged by both rainfall and the river across the entire valley it is possible for the river channel to meander considerably before the makeup water capability would be affected.

Water for plant use is withdrawn from the aquifer by means of two Ranney wells located as shown on Figure 2.4.1. As part of the design to place Ranney collectors in the floodplain, the adjacent river banks are being stabilized to minimize erosion. However, because the Chehalis is a meandering river, displaying oxbows and sloughs, diversion affecting the Ranney wells is still considered possible. Regardless of the availability of the Ranney wells, the safety of the plant will not be jeopardized because, as described in Section 2.4.11.2, emergency cooling of the plant can be accomplished using the ultimate heat sink, which consists of a dry cooling tower.

The staff thus concludes that potential channel diversions, although remote, present no safety-related hazard to the plant and that the requirements of 10 CFR Part 100, relative to channel diversions, have been met.

2.4.10 Flood Protection Requirements

As described in Section 2.4.3, the staff concluded that the plant is located considerably higher than any credible flood in the Chehalis River. However, in Section 2.4.2, the staff concluded that the applicant had not provided sufficient information to support its conclusion that local intense precipitation will not enter safety-related buildings. Additionally, it is not evident that a roof section adjacent to the steam tunnel has been designed to support potential ponded rainfall. The applicant will be required to provide additional information in support of its conclusions. Resolution of these issues will be addressed in a supplement to this SER.

2.4.11 Cooling Water Supply

2.4.11.1 Normal Water Supply

Under normal operating conditions, waste heat will be dissipated to the atmosphere by a natural draft cooling tower. Makeup water to replace the water lost by evaporation, blowdown and drift, will be supplied to the cooling tower by two Ranney wells located in the alluvial aquifer which underlies the Chehalis River valley (see Figure 2.4.1). To prevent excessive buildup of dissolved solids in the cooling system, a certain amount of cooling water must be continuously discharged to the Chehalis River after first being cooled down by a supplemental cooling facility. The blowdown discharge will be diluted in the river through the use of a submerged multiport diffuser.

The maximum makeup water requirement for WNP-3 is approximately 18,000 gpm (40.0 cfs), and a single Ranney well is capable of supplying this amount on a continuous basis. The capability of the Ranney wells to supply this quantity of water is independent of low flows in the Chehalis River. The State Energy Facility Site Evaluation Council (EFSEC), however, has administratively established that plant makeup withdrawal (except for a hot-standby maintenance flow of 2 cfs) must cease when the daily river flow goes below 550 cfs. Additionally, plant withdrawal may not exceed the difference between the river flow and 550 cfs. The long-term annual average flow of the Chehalis River at the site is estimated to be about 6820 cfs. The estimated average monthly flows vary from 730 cfs in August to 14,900 cfs in January. The minimum and maximum historical flows at the site are estimated to be about 400 and 97,100 cfs, respectively.

Because of the water withdrawal limitation established by the EFSEC, the plant will have to be shut down whenever the daily river flow goes below 550 cfs. The applicant estimates that, on the average, this will occur about four days a year.

The applicant has completed a contract with the city of Aberdeen to purchase releases of 62 cfs of flow from the Wynoocnee Reservoir to supplement the

Chehalis River during low flow periods. This water is to be used to mitigate adverse impacts associated with the consumptive use of river water.

The staff has reviewed the material presented by the applicant and concludes that neither drought periods nor the conditions established by the EFSEC, regarding withdrawal of water for plant use, will unduly restrict the availability of cooling water for normal operations as required by GDC 44.

2.4.11.2 Emergency Water Supply

Emergency safe shutdown and cooldown of WNP-3 can be accomplished using the ultimate heat sink which consists of a dry cooling tower located adjacent to the reactor auxiliary building. The UHS is also used for heat rejection during normal operation and shutdown. The UHS is comprised of two independent 100 percent capacity trains. Each train has a cooling tower of the aircooled heat exchange type, with Component Cooling Water System (CCWS) fluid passing through the tube side and air over the exterior surface of the tubes. The cooling tower fans cycle on and off and through low and high speeds automatically during all operating modes to maintain the desired temperature.

The UHS is capable of operating under a range of heat loads during normal and emergency conditions. With a dry bulb ambient temperature of 101.5°F, each train has a design heat rejection capability of 180 x 10^6 Btu/hr with 11,000 gpm CCWS flow entering the cooling tower at 153°F and leaving at 120°F.

During normal operation, the UHS in conjunction with the CCWS heat exchanger can reject the normal heat loads while maintaining CCWS temperatures at or below 95°F. For accident conditions the UHS is designed to reject the maximum accident heat loads without the need for additional cooling by the CCWS heat exchanger. By operating the fans at full speed, one cooling tower can maintain the CCWS temperature at or below 120°F during accident conditions.

As suggested in R.G. 1.27, the applicant analyzed the 30-day period following a design basis accident. This analysis showed that each cooling tower is capable of dissipating the maximum heat load following a LOCA. By maintaining the temperature of the water exiting the cooling tower at 120°F or less, the maximum

heat rejected during the 30 day analysis, was 133 x 10^6 Btu/hr. Since the cooling tower has a design heat rejection capability of 180 x 10^6 Btu/hr, the applicant concluded that the UHS is capable of providing adequate cooling for at least 30 days. The applicant thus concluded that the UHS meets the recommendations set forth in R.G. 1.27 and thus the requirements of GDC 44 with respect to thermal aspects of the heat transfer system.

The staff has not completed its review of the UHS so a conclusion on its acceptability cannot be made at this time.

2.4.12 Groundwater

2.4.12.1 Groundwater Conditions

The plant area is underlain by the Astoria Formation which is a thick (2500 to 3000 feet) deposit of relatively impermeable tertiary sandstones. This formation dips northward toward the Chehalis River and forms the most extensive geologic unit at the site. All Category I structures are founded in the Astoria sandstone approximately 2000 feet above its base. North of the plant and south of the Chehalis River, pleistocene terrace deposits overlie the Astoria Formation. These deposits consist mostly of sands, gravels and silts. Further north, the Chehalis River valley is underlain by alluvial materials. The Ranney wells which supply the cooling water required for normal operation of the plant are founded in these alluvial materials.

In the site area, groundwater is found in the Astoria Formation, the pleistocene deposits and in the alluvial materials in the Chehalis River floodplain. The Astoria Formation has very low permeability and permits only small amounts of recharge and minimal groundwater movement. Because of this, it is not a productive groundwater source. The groundwater table beneath the plant site area follows the ground topography and is parallel to the weathered and unweathered zones of the Astoria sandstone. The groundwater slopes northward toward the Chehalis River. Prior to construction of the plant, the groundwater level was at an elevation of about 380 feet MSL.

Groundwater also occurs in a discontinuous manner in the pleistocene terrace deposits. Recharge is derived from infiltration of rainfall on the areas above the terrace levels and infiltration from the Chehalis River. There are no known major aquifers within these deposits and only three domestic wells tap the terrace groundwater in small perched aquifers.

The only satisfactory source of groundwater in the site vicinity occurs in the alluvial aquifer that underlies the Chehalis River valley. This aquifer extends downward from about 10 feet below the ground surface to about 165 feet. The high permeability and transmissivity coefficients of this unconfined aquifer indicate that the aquifer reacts much like a reservoir and a hydraulic conduit. Recharge to the aquifer occurs all across the river valley as well as in the river channels from the 70 to 80 inches of annual precipitation in the area and from the high surface inflow from the widespread Chehalis River basin. The aquifer accepts surface water for storage during these periods until the underground storage is full. The permeable aquifer discharges readily into streams and rivers during periods of low flow. The alluvial aquifer is limited horizontally by tertiary sandstone sediments on the south side of the river and by by the southern edge of the Olympic Mountains on the north side of the valley. The aquifer extends two miles across the Chehalis River valley, about 14 miles downstream to Grays Harbor and about 15 miles upstream to the eastern limit of Grays Harbor County. As described in Section 2.4.11.1, this aquifer will be used to supply makeup water to the plant.

2.4.12.2 Dewatering System

A permanent groundwater drainage system (GWDS) that operates solely by gravity has been installed around the WNP-3 reactor auxiliary building (RAB). The GWDS consists of vertical 6 inch diameter half-round drain pipes spaced at 8.5 feet intervals around the RAB at the interface between the rock and exterior concrete walls. The vertical pipes, which drain the surrounding rock, rise from the base of the foundation mat to elevation 390 feet MSL except at the west side of the RAB where the vertical drain pipes extend into the turbine building, four feet above the floor elevation of 390 feet MSL. This elevation is above the highest level (3 feet) that the applicant has estimated circulating water could rise, in the event of a Circulating Water System break inside the building.

Thus any water released into the Turbine Building would be prevented from directly entering the GWDS.

Groundwater that seeps into the vertical drain pipes is conveyed to a 8 inch diameter horizontal drain pipe located along the periphery of the mat. Collected groundwater is then routed to a 6 foot diameter drainage tunnel that drains into a small tributary of Workman Creek south of the plant. In addition, 8 inch diameter perforated undermat drains have been placed diagonally beneath the foundation mat. These undermat drains are also connected to the drainage tunnels. Manholes are provided at each corner of the RAB to allow for periodic inspection and cleaning of the GWDS.

The GWDS is not classified as a seismic Category I system except for the manholes at the corners of the RAB and the upper 5 feet of the extended vertical drain pipes at the west side wall of the RAB. The manholes are seismic Category I to provide access to the horizontal drain pipe and to the drainage tunnel in the event of an earthquake. The upper portions of the vertical drain pipes are seismically qualified to resist the passive pressure of the sandstone on the embedded portion of the pipes and to resist the peak seismic acceleration of the RAB at grade elevation.

The applicant has stated that in the unlikely event of a complete blockage of the GWDS, the walls and foundation mat of the RAB are designed to withstand the resulting hydrostatic load of a groundwater level at elevation 365 feet MSL. This elevation is 39 feet above the bottom of the mat, elevation 326 feet MSL, and 24.5 feet below the plant grade elevation of 389.5 ft MSL. The applicant states that in the event of a complete blockage of the GWDS, there would be sufficient time to repair the system before the surrounding groundwater would rise to an elevation of 365 feet MSL. This time was estimated to be a minimum of 115 days.

Using the procedures in SRP 2.4.12 including Branch Technical Position (BTB) HGEB-1, the staff has reviewed the information provided by the applicant in the FSAR. The staff concludes that there is inadequate information with which to assess the applicant's conclusion that in the event of a complete failure of the dewatering system there will be sufficient time to repair the system before the surrounding groundwater would rise to an unacceptable level. The staff

has asked the applicant to provide additional information concerning the dewatering system. Until the staff receives the requested information, it cannot determine whether the plant design meets the criteria of BTP HGEB1 of SRP 2.4.12 or the requirements of GDC 4 with respect to the dewatering system.

The applicant considered a break in the underground Circulating Water System (CWS) pipe and its effect on the ability of the dewatering system to maintain water levels below elevation 365 feet MSL. The applicant stated that if a major break in the CWS pipe was to occur, water would be forced upward out of the trench in which it lies, and would be drained away from the surface by storm drains.

The staff has reviewed the information provided by the applicant according to procedures described in BTP HGEB1 in SRP 2.12, and concludes that insufficient information has been provided by the applicant, concerning the effect of postulated pipe breaks on the dewatering system. The staff has submitted questions to the applicant on the subject of pipe breaks. The staff will complete its review pending receipt of responses from the applicant concerning the effect of pipe breaks on the dewatering system.

In Section 2.4.4 of the CP-SER, the staff stated that the applicant had committed to monitor groundwater levels at the walls of the reactor auxiliary building and to radiologically monitor discharges through the groundwater drainage system. The applicant proposed to do this by installing one piezometer at each reactor auxiliary building wall, located between the manholes and 10 feet from the wall. This instrumentation was to provide an alarm in the control room if a specified level was exceeded. The monitoring program proposed by the applicant in the FSAR is considerably different than that proposed at the CP stage. The applicant did not indicate in the FSAR if any part or all of the CP stage monitoring program will be used during plant operation; instead, it stated that inservice surveillance of the vertical drains, horizontal headers, and the drain tunnels will be made at 90 day intervals during the wet season. The vertical drains will be tested by dropping a light down the drain to the horizontal header and observing the light from the manholes through the header. The horizontal headers will be inspected by shining a light at one end of each header

and observing it from the other end. The drainage tunnel will be inspected by walking along it from the manhole to the concrete plug and looking through the concrete plug to daylight.

The staff agrees that this surveillance program will effectively indicate whether the dewatering system is functioning and whether there is any blockage. However, it will not be effective if there is standing water in the vertical drains. The applicant should explain if any part or all of the monitoring program committed to in the CP-SER is to be used to monitor the performance of the dewatering system during operation. If it is not to be used, the applicant should explain why not. In any event, the applicant should describe the procedures to be used to monitor groundwater levels in the event that there is standing water in the vertical drains.

The staff has reviewed the information provided by the applicant and concludes that it cannot complete its review of the dewatering system because the applicant has not provided sufficient information to assess the following:

(1) The time for groundwater levels to rebound to the hydrostatic design level of 326 feet MSL in the event of a complete failure of the dewatering system.

(2) The adequacy of the in-service monitoring program.

(3) The effects of pipe-breaks on the dewatering system.

The staff has submitted questions to the applicant and will complete its review pending receipt of responses to those questions.

2.4.13 Accidental Release of Liquid Effluents

(To be provided later)

2.4.14 Technical Specifications and Emergency Operation Requirement

Because the staff has not yet completed its Hydrologic Engineering review, the need for technical specifications and/or emergency operation requirements has not been determined at this time.

2.4.15 Conclusions

According to procedures outlined in the SRP, the staff has reviewed the design of WNP-3 with regard to hydrologically and hydraulically related plant safety features. On the basis of this review, the staff concludes that large-scale river or stream floods do not pose a threat to the safe operation of the plant or the integrity of the site. The staff, however, is unable to conclude that local flooding will not threaten the plant. The staff concludes that WNP-3 meets the requirements of GDC 2 with respect to potential flood hazards except for the outstanding item concerning local flooding.

The staff has reviewed the availability of water for normal cooling purposes during diminished flow periods in the Chehalis River and the conditions imposed by the EFSEC. The staff concludes that there is sufficient water available to maintain safe plant operation over any reasonable drought period as required by GDC 44 with respect to normal cooling water availability.

The staff has reviewed the information on the dewatering system presented by the applicant and concludes that there is insufficient information concerning the time it would take for the groundwater level to rise to an unacceptable level in the event of a total failure of the dewatering system. There is also not enough information concerning the effect of pipe breaks on the dewatering system nor on the proposed in-service monitoring program. The staff will complete its review of the dewatering system once the applicant addresses the concerns stated in this draft SER and provides responses to the questions which have already been provided to the applicant.

In addition, the staff has not completed its review of the performance of the ultimate heat sink. This review is being conducted by Argonne National Laboratories and the results will be presented in the SER.

2.5 Geology and Seismology

The geology and seismology of the WNP-3 site were reviewed during the early and middle 1970's during the Construction Permit (CP) review.

As a result of the CP review the NRC staff concluded that:

(1) the inferred large deep seated fault blocks that have been associated with large earthquakes in the southern part of the Puget Sound are not present in the site area;

(2) movement of faults in the site vicinity most likely ceased in the late Tertiary, more than 2 million years before present (mybp), and are therefore not capable within the meaning of Appendix A, 10 CFR 100;

(3) there are no known structures in the immediate site vicinity that could be expected to localize earthquakes there;

(4) the applicant's assessment of the possible volcanic risks in the site region are adequate and that a problem of this type does not exist at the site; and

(5) the safe shutdown earthquake (SSE) with a maximum acceleration of 0.32, and the operation basis earthquake of 0.16g are conservative when applied to the foundation level.

During construction of the facility numerous minor faults were found in the excavation. The applicant investigated these faults and determined that they were at least 630,000 years old but most likely more than 2 million years old. The staff reviewed the applicant's data and made several visits to the site to examine the faults and concluded that the faults were not capable.

In 1974 through 1976, as a result of licensing activities for the Skagit Nuclear Power Plant, studies were initiated concerning the 1872 Earthquake (MMI IX, magnitude 7.0). New data from these studies raised the question as to whether or not an event of that size could occur at the WNP-3 site. The applicant investigated that earthquake, mostly in regard to its Hanford sites,

and determined that it was related to tectonic structure within a broad epicentral zone and therefore could not occur at the site. The NRC staff reviewed that data and data compiled by a panel of experts formed by Northwest US Utilities, and that of another panel of USGS and NOAA experts, and concluded that the 1872 Earthquake was centered in the region between Entiat, Washington and Chilliwack, British Columbia (NRC, 1978) and should not be expected to recur at the site. The most recent staff and USGS discussion of this earthquake can be found in the WNP-2 SSER.

On May 18, 1980, after several weeks of resurgent activity, Mount St. Helens erupted violently sending large quantities of ash several hundred miles downwind to the east. The NRC requested the applicant to reassess the volcanic hazards to the site based on the new data. The applicant concluded, based on that assessment, that the maximum potential ashfall that could be expected from such an eruption from the closest volcanoes during the worst meteorological conditions, would result in a maximum of 1.75 inches of ash at the site. They stated that the plant design could accommodate that kind of ashfall. The NRC staff reviewed the applicant's data and USGS data collected with partial NRC funding and concurred with the applicant's conclusion, but requested additional supporting data.

The staff has completed its review of the FSAR. It has held several meetings with its advisors, the U.S. Geological Survey and its geological consultant, Dr. David Slemmons, two technical meetings with the applicant and its consultants, and conducted a geological reconnaissance of the site and region around the site. On April 28, 1983 we transmitted questions, including those of our advisors to the applicant. Because of the June 1983 postponement of the WNP-3 site construction, those questions or outstanding issues have not been responded to. These open topics will be presented in Sections 2.5.2, 2.5.2, and 2.5.3.

Because of the extensive geologic and seismic information (mostly about subduction zones) that has come out since completion of the CP review, new staff concerns have arisen; however, the following CP conclusions of the staff are still valid:

(1) the inferred large deep-seated fault blocks that have been associated with large earthquakes in the southern part of Puget Sound are not present in the site area;

(2) movement on mapped faults in the site vicinity, including those in the excavation are ancient and are not capable; and

(3) the volcanic hazard to the site has been adequately addressed even in light of the recent eruption of Mt. St. Helens and has been appropriately considered in the design.

Based on new data since the CP, the adequacy of the SSE is in question for the following reasons which reflect our general concerns:

(1) The possibility of a large or great earthquake on a subduction zone beneath the site;

(2) the possibility of unrecognized low angle thrust faults in the site vicinity that could cause large close-in earthquakes or surface faulting at the site.

These issues will be addressed in greater detail in the following sections.

2.5.1 Basic Geologic and Seismic Information

2.5.1.1 Regional Geology

The WNP-3 site is located in the Pacific Border Physiographic Province of Washington State, about two miles south of the town of Satsop and 16 miles east of the city of Aberdeen. The site area lies in the Chehalis Lowlands, which comprise a physiographic zone separating the northern termination of the Oregon Coast Range from the Olympic Mountains.

The site and its environs are largely underlain by Cenozoic strata. Relative to more northern areas of the region, rocks of the site area are not highly deformed. Igneous rocks of Mesozoic and Cenozoic age, however, are more

abundant than either sedimentary or metamorphic units thoughout the region. The nearest outcrops to the site of Mesozoic and Paleozoic rocks (metamorphic, igneous, and sedimentary types) are found in the highly deformed area some miles to the north and northwest of the proposed plant area. Lithologically, the Cenozoic strata consists predominantly of marine clastic sediments deposited on a basement of Eocene oceanic basalts.

The tectonic history of the site region is complex, with eastward and westward directed low-angle thrusts, grabens, granitic plutons, and stratovolcanoes being best displayed and developed in the Northern Cascades. In the Northern Cascades, the Paleozoic Era is characterized by metamorphic and eugeosynclinal rocks. Eugeosynclinal sediments, granitic plutons, low-angle thrusts, and grabens were formed throughout the Mesozoic Era. During Cenozoic time, the formation of grabens, granitic plutons and basalt flows predominated tectonic activity. These events were followed by several orogenic periods which caused folding and faulting of the older formed rocks and general uplift of the region, and the stratovolcanoes of the Cascade range began to form. The structural features that were formed during these orogenies, and the region, were subsequently eroded during the Quarternary to produce the present day topography. While it appears that the last major period of deformation in the region ended in the Late Tertiary (Pliocene), evidence from Pleistocene deposits in the coastal areas west of the site, from 1100 year old fault dates in the Puget Sound area to the north, and from three active stratovolcanoes in the central part of the state to the east of the site, show that tectonism continued on a more minor scale through the Pleistocene into the Holocene.

The tectonic deformation of Western North America appears to be intimately related to the interaction of two major lithospheric plates, the North American Plate and the Pacific Plate. The interaction is principally along two major transcurrent faults, the San Andreas Fault in California and the Queen Charlotte Fault off Western Canada. However, in the area between Cape Mendocino in northern California and the southern extent of the Queen Charlotte Fault off the western p of Vancouver Island, the two major plates named above are separated from one another by the small Juan de Fuca Plate.

The interaction between the Juan de Fuca Plate and the North American Plate is not presently understood. The magnetic anomaly pattern east and west of the Juan de Fuca Ridge indicates that part of the Juan de Fuca Plate has been subducted beneath the North American Plate. Also, the chain of stratovolcanoes which forms the axis of the Cascade Mountains is believed to have been produced by magma from a subducting plate (Atwater, 1970). Several other types of data indicate that an episode of late Cenozoic subduction occurred in this region of western North America. Seismic reflection surveys off the coast show a sediment-filled trench at the base of the Continental slope (Hays and Ewing, 1970). Anomalously high gravity values on Vancouver Island are suggestive of a remnant subducting slab beneath the region (Stacey, 1973). Seismic wave velocities indicate that a high velocity slab exists beneath the Puget Sound Basin (McKenzie and Julian, 1971; Crosson, 1972) which is indicative of a subducting lithospheric plate.

The applicant has thoroughly reviewed the above-mentioned items and other types of data related to the current interaction of the lithospheric plate boundaries, including studies of plate kinematics (Silver, 1971; Atawater, 1970). While the available data are not clearly definitive, the applicant concludes that the data tends to support the interpretation that subduction is no longer occurring along the Juan de Fuca-North American Plate boundary or is occurring asseismically.

Available evidence examined during the CP review indicated that subduction along the Juan de Fuca Plate-North American Plate boundary was not currently occurring. In particular, earthquake activity indicative of a Benioff zone (a characteristic of subducting plates) was absent in this region. Also, the orientation of the present regional stress field was inconsistent with active subduction. Analysis of earthquake source mechanisms showed that the maximum principal stress is north-south compressional and the minimum principal stress varies from east-west to nearly vertical (Dehlinger and Couch, 1969; Couch and McFarlane, 1971; Crosson, 1972; Malone, et al., 1975).

New information has been developed since publication of the CP SER and the FSAR, however, which may require a modification of the above conclusions. This new information may indicate that subduction is continuing and that the

two plates may be coupled. That information and NRC staff's concerns are presented in Sections 2.5.2 and 2.5.3.

Numerous reverse faults of a generally northwest or northeast trend, marking elongated basement uplifts, occur throughout the basaltic rocks of the region. These structural features are cut by east to northeast trending normal faults bounding areas showing different amplitudes of folding. Some of these faults significantly displace Tertiary strata in the region. The above described faults are thought to be the result of northeast compression of the crust, which was recurrent several times throughout the early Tertiary, until at least the middle Miocene. The basaltic basement complex shows the highest degree of faulting, with the intensity of faulting declining with the decreasing ages of the overlying rock units.

A line of stratovolcanoes extends along the Cascade Mountains from northern California to southern British Columbia. Eight of the volcanoes are within 200 miles of the Satsop site, the nearest being Mt. Rainier and Mt. St. Helens, each about 80 miles away. All of the volcanoes are believed to have been active within the past 15,000 years and three of them, Mt. St. Helens, Mt. Rainier, and Mt. Baker are considered active at the present time.

Prior to 1980 Mt. Rainier had received the most study. The studies show that it has been intermittently active during the last 10,000 years. This activity has been mainly of pyroclastic type, but includes at least one flow which extended nine miles from the mountain. Three of the tephra eruptions deposited about one inch of material up to 25 miles east of the mountain. The last major eruption occurred about 2000 years ago, but minor eruptive activity occurred 120 years and 150 years ago.

In addition to the eruptions of tephra, numerous mud flows have occurred at Mt. Rainier. The largest of these, the Osceola mud flow, occurred 5700 years ago. It extended about 70 miles down-valley from the volcano. None of the river valleys which could be potential mud flow pathways pass near the Satsop site. We conclude, therefore, that no mud flow hazard exists at the site.

A reassessment of the volcanic hazard was made after the May 18, 1980 eruption of Mt. St. Helens. It was found that downwind of the prevailing winds from the volcano at about 80 miles (plant's distance) there was an accumulation of 6 inches of tephra. The applicant reduced that value of 1.75 inches because the WNP-3 plant is upwind from the nearest Cascade volcanoes. This is a reasonable assumption but we require more data about the maximum thickness of tephra landfall and maximum rate of ash fall to support it.

In summary, it can be said that, while the geologic conditions of the Satsop site and its environs are very complex, and the area is still tectonically active, based on our review of the applicant's work to date, there are no known faults or other structures in the immediate vicinity of the site which could be expected to localize earthquakes; however, because of recent findings about the tectonics of the region, we require additional information to support that conclusion. The outstanding items concerning faulting in the region are discussed more fully in Section 2.5.3.

2.5.1.2 Site Geology

The WNP-3 site is located on a ridge in the Willapa Hills, 1 mile south of the intersection of the Satsop and Chehalis Rivers. The site elevation was +595 MSL prior to excavation. Elevations rise to +1,768 MSL at Minot Peak, 4 miles to the south. The floodplain of the Chehalis River Valley is about 1 mile wide and has a general elevation of +25 MSL. Drainage patterns in the site area form a modified dendritic pattern that is structurally controlled to some extent by the regional Tertiary folding and jointing. Slopes are generally moderate, but range from nearly flat to vertical. The abundant weathering profiles, relict erosion surfaces and Pleistocene terraces in the area were used extensively to determine an upper limit to areal tectonic events.

The site vicinity is underlain by Quarternary deposits which consist of weathered gravels of the Wedekind Creek and Logan Hill formations of early to middle Pleistocene age; glacio-fluvial sands, silts, and gravels of middle to late Pleistocene age; loess of late Pleistocene age; colluvium and landslide deposits of late Pleistocene to Holocene age; and Holocene colluvium.

Approximately 15,000 feet of Tertiary rocks are present in the site vicinity, the oldest of which is the middle Eocene Crescent formation, a submarine basalt. Late Eocene Skookumchuck and McIntosh formations siltstones, tuffs, breccias, and sandstones overlie the basalt. The late Eocene to early Miocene Lincoln Creek formation of tuffaceous siltstone overlies the older four formations and is overlain by early to middle Miocene sandstones of the Astoria formation. The uppermost rocks in the site area are siltstones, sandstones, and conglomerates of the Montesano formation of late Miocene to early Pleistocene age. The plant site is founded on massively bedded sandstone of the Astoria formation.

Structurally the site is located on the nose of a broad poorly defined anticline, which is an extension of one of the areas several uplifts, the Minot Peak uplift. Typical of other anticlines in the region, the Minot Peak uplift has the basaltic basement rocks exposed in its core. Several significant faults (some with several thousand feet of displacement) in the site area can be shown by various means (e.g., terrace dating, saprolitization rates, erosion rates) to be associated with deformations no younger in age than Middle Quaternary (more than 630,000 years ago). Thus, they are not considered to be capable faults within the meaning of 10 CFR Part 100, Appendix A.

Numerous landslides have been mapped on the site locality. Many of these, though not most commonly, have been identified in the Astoria formation, which is the foundation bedrock. These slides in the Astoria formation are related to slippage along weathered siltstone interbeds. Based on a detailed investigation of local landslides, the applicant determined the geologic and geomorphic conditions necessary for sliding to occur: strong weathering of the Astoria rock, the presence of siltstone beds in the Astoria, topographic slopes inclined in the direction of bedding dip, and undercutting of bedding beneath dip slopes. Site investigations showed that these conditions do not exist at the site. The staff concludes that landsliding does not represent a problem at the site.

2.5.2 Vibratory Ground Motion

As a result of regional and site investigations performed by the applicant and others since the issuance of the CP-SER for WNP-3 in February 1975, the knowledge

of the area has been greatly enhanced. The applicant has, and is continuing to undertake numerous studies and investigations that will provide an extensive amount of new information and interpretation. The s:aff anticipates that our review of this new information will lead to an understanding and resolution of many issues relating to the site vibratory ground motion determination.

The increasing amount of new information, however, may require the reinterpretation of some previous positions of the staff, the USGS, and the applicant. Presently the open seismological items have been transmitted in the form of questions (Q230.1 through Q230.6) to the applicant. The applicant and the staff have met to discuss these open issues, and it is anticipated that the applicant will undertake a rigorous program of investigations to collect the information which will allow the staff to resolve the open issues. A summary of these issues follows.

The most significant seismologic issues involves the seismogenic potential of the subducting Juan de Fuca plate beneath WNP-3. The staff concluded in the CP-SER for WNP-3 in February 1976 that "while the available data are not clearly definitive, we believe that they tend to support the interpretation that subduction is no longer occurring along the Juan de Fuca - North American Plate boundary." Since that time additional recordings of small earthquakes have revealed an inclined zone of seismicity dipping to the east-northest (Crosson, 1980). In addition, based upon the work of Ruff and Kanamori (1980) and Kanamori (1983) regarding the seismogenic potential of subduction zones, a number of questions regarding the Juan de Fuca zone have been raised. It is the applicant's position as discussed in FSAR sections 2.5.1.1.4.2 and 2.5.2.4.2.2, that the interface between the Juan de Fuca and North American plates will not be the location of a large magnitude earthquake. The staff has indicated via the review questions that the applicant must document in greater detail their position.

In particular the staff has requested at the applicant document the following information regarding the Juan de Fuca plate. This includes the applicability of Kanamori (1983) relationship, and examples of aseismic subduction zones which share the same characteristics witn tne Juan de Fuca zone. The magnitude of the largest shock in the plate or along the plate interface that could occur

without exceeding the SSE and ground motion attenuation from subduction zones that can be used for the WNP-3 site will also be documented. The magnitude of the maximum credible earthquake on the subduction zone, along with estimates of vertical and horizontal response spectra, depth and configuration of the subducting plate based upon earthquake locations cross-sections, fault plane solutions, and historic earthquake re-locations will also be provided by the applicant and reviewed by the staff.

The staff has also requested that the applicant calculate site specific response spectra for the maximum historical earthquake, not associated with known geologic structure, in the tectonic province of the site, and for the maximum earthquake on the Olympia Lineament. The applicant has also been asked to estimate the annual exceedance probability of the SSE using all possible seismologic source including the subduction zone.

The staff, the USGS, and Dr. Slemmons will undertake and participate in meetings and probably several site visits to review the applicant's additional information and field investigations. Upon the applicant's submission and the staff's review of the new information, the staff will issue its Final SER. This SER will discuss in detail all the relevant geologic and seismic issues including the regional and site geology, capable faulting, seismicity, operating and safe shutdown earthquakes, and the vibratory ground motion. Reports by the USGS and Dr. Slemmons will be incorporated as appendices and will be discussed in the SER.

2.5.3 Surface Faulting

The applicant has determined that the structural geology of the site and regions around the site is characterized by large uplifts and faults and folds related to those uplifts that were formed by regional northeast directed compression during the Tertiary period. Three of these uplifts are present within the site vicinity, the Minot Peak uplift, the Blue Mountain uplift, and the Black Hills uplift. The site is located on an anticline which is the northern extension of the Minot Peak uplift. All of the uplifts are bounded primarily on the southwest sides and soutneast sides by high angle faults that strike north-northwest, and east-northeast, respectively, with offsets ranging

from several thousand feet to several hundred feet. The closest faults of this kind to the site are the Weikswood fault on the southwest side of the Minot Peak uplift and the Gibson Creek fault on the southeast side of the uplift. Offsets on both faults exceed 2000 feet. The Weikswood fault is approximately 1 mile south of the site at its closest approach, and the Gibson Creek fault is about 5 3/4 miles south of the site.

The applicant investigated all of the faults in the site vicinity by means of a literature search, mapping, borings, trenching, and remote sensing techniques. The applicant determined an upper limit of age of last movement on the faults by analyzing cross-cutting relationships between faults and stratigraphic contacts, relict erosion surfaces, Quarternary deposits, paleosols and weathering profiles. By determining the ages of these features the applicant was able to show an upper limit of movement on these faults of at least 630,000 years before present and more likely 2 million years before present. The staff has reviewed the data that is the basis for the conclusion and concludes that the faults mapped in the site vicinity are not capable within the meaning of Appendix A. Numerous minor faults were encountered in excavations for the plant. Most of these faults are northwest to northeast striking reverse faults. The applicant has made a good case in the FSAR for relating these faults to the regional faults and to the Late Tertiary northeast directed compression. NRC staff geologists examined these faults on several occasions. The NRC concludes that the faults mapped on and around the site are not capable (Appendix A).

On the other hand, considerable new geological information regarding the tectonics of the site region has been developed since the FSAR was published. Although we hold to our position that the faults in the site locality are not capable, some of the new data raises some concern. For example, it is not clear what happens to the faults at depth. If they are indeed related to Late Tertiary tectonics which are no longer in existence that is one thing, but if they are tied to large eastward dipping thrust faults that flatten downward (eastward), which are related to an active subduction tectonic style of the Juan de Fuca plate, then additional analyses and possibly investigations, will have to be carried out. A major northwest-trending fault in the Humptulips River area (Tabor and Cady, 1978) is a possible fault of this kind. It projects northwestward under Quaternary deposits to an outcrop of steeply

dipping Pleistocene deposits (op. cit) on the west Fork of the Humptulips River. The capability of this fault may be important to the site in light of the following. Offshore studies by Silver (1972) and Snavely and Wagner (1982) indicate a subduction tectonic style characterized by eastward (landward) dipping thrust faults that generally steepen westward (upwards) and that have offset sediments as young as Quaternary. Considering this structural framework, we have asked the applicant to evaluate the possibility that the Humptulips fault, if capable, extends southeastward as a continuous fault or fault zone along the steepened west limb of the Wynoochee anticline (Rau, 1976) and on into the less well-defined Melbourne anticline (Gower and Pease, 1965) or alternatively to the southeast of these structures. We have requested the applicant to determine whether or not the Humptulips fault is throughgoing and capable, and, if so to evaluate the effects on the site.

Recent seismic reflection, remote sensing, and geophysical data covering the area has been gathered that post dates the FSAR publication and therefore has not been evaluated with respect to the site. We have recommended that the applicant assess these data with respect to the site.

Many of the natural drainage features in the site vicinity occur along projections of mapped faults although the faults are shown to terminate away from the stream valleys but along projections of their trends. Also many drainages are oriented in a pattern that is paralleled to the north-northwest and northeast striking fault pattern, yet the streams are not considered to be fault controlled by the applicant. Evidence that supports the conclusion that the drainage features are not fault controlled is needed before the staff can complete its review.

The applicant has dismissed offset magnetic anomalies KK and HH on the Juan de Fuca plate as probably due to episodic jumping of short transform faults connecting offset segments of the spreading ridge as suggested by Hey (1977). (FSAR 2.5-44). Provided that successive jumps are in the same direction and occur after equal increments of spreading, the jumps should produce a V-shaped wake consisting of a pair of lineaments intersecting at the ridge. Although KK seems to form such a wake, mirrored in the Pacific plate, HH is less convincingly matched (c.f. Barr, 1974 and Elevers and others, 1973). Considering the

difficulty of i entifying the mirror image of HH, the applicant has been requested to evaluate the hypothesis that HH is a fault as suggested by Pavoni (1966), and that the on-shore subcrustal extension of HH could be the source of deep-seated major earthquakes in the Puget Sound region (Fox, 1983), and to evaluate the response at the site of a major earthquake on fault HH.

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2.5.4 Stability of Subsurface Materials and Foundations

2.5.4.1 Site Conditions

2.5.4.1.1 General Plant Description

The WNP-3 plant site is located on a ridge at the northern edge of Willapa Hills, in the southeastern portion of Grays Harbor County in the State of Washington. The site is approximately 16 miles east of Aberdeen and approximately three miles south of the town of Satsop. Prior to the start of the plant construction, the ridge at the plant location was at an elevation of approximately 480 feet above mean sea level. During early construction, the general plant grade was excavated to an elevation of approximately 390 feet.

The common foundation mat for the WNP-3 reactor building and reactor auxiliary building is supported on essentially fresh sandstone at an elevation of approximately 326 feet. All other Seismic Category I structures are founded at plant grade on weathered sandstone. These structures include UHS Dry Cooling Tower Train A and Train B, Dry Cooling Tower Control Building, Condensate and Refueling Water Storage Tank Enclosure, two Diesel Oil Storage Tank Enclosures and Manholes for the gravity drainage system. About 1000 feet northeast of the powerblock there is a cooling tower which is used for normal cooling operations. It is not a seismic Category I structure. The ultimate heat sink function for WNP-3 plant is performed by two dry cooling towers; no makeup water is required for safety-related cooling of the plant.

There are permanent natural slopes in the north-south direction and man-made excavation slopes in the east-west direction whose failure could affect the safe operation of the plant. The natural rock slope south of the power block dips at an average slope of 3 horizontal to 1 vertical (3:1), with a maximum slope of 1-1/2:1. The crest of the natural slope is more than 250 feet from the edge of the powerblock. The man-made rock slope east of the powerblock is about 180 feet high, rising at an average slope of 4-1/2:1, with a maximum slope of 3:1. The toe of the slope to the east is more than 350 feet from the edge of the dry cooling towers.

2.5.4.1.2 Subsurface Investigation

The subsurface investigation program at the site consisted of drilling and trenching. A total of 95 borings were drilled at the site and soil and rock samples were recovered. Most of the borings were drilled with mud and tricone bit. Rock cores were obtained by using NX double tube coring barrels and diamond bits. Core recovery and rock quality designation (RQD) were recorded. Piezometers were installed in 52 boreholes after completion of the drilling. In addition, 65 trenches of various lengths and depths were excavated.

The geophysical investigations consisted of seismic refraction surveys and cross-hole shear wave velocity measurements. The geophysical surveys show good spatial correlation with the test boring results.

2.5.4.1.3 Subsurface Materials

2.5.4.1.3.1 Investigation of Subsurface Materials Properties

The plant area is underlain by Astoria Formation, a sandstone in various degrees of alteration including residual soil, weathered sandstone, fresh (unweathered) sandstone and tuff. This formation also contains some siltstone strata and several tuff beds. Fresh sandstone is differentiated from weathered sandstone on the basis of color change. Fresh sandstone is light to dark gray, and weathered sandstone is yellowish-brown from the oxidation of iron minerals in the sandstone. Joints in sandstone show two dominant trends; one set ranges in strike from N34°E to N65°E and dips generally from 50°-70° SE and the second set strikes about N43°W and dips nearly vertical. The general spacing of joints within a joint set is approximately 1 to 5 feet and the distance between joint sets is 40 to 100 feet.

During construction all residual soils and most of the tuff were excavated from the plant area, and the plant grade was established at elevation 390 feet above mean sea level on the exposed weathered sandstone surface. At the location of WNP-3 site, the weathered sandstone extends to a depth ranging from 40 to 60 feet below final plant grade level and is underlain by fresh sandstone. A 7 to 11 foot thick tuff bed exists in the vicinity of the northern

edge of the plant location at 0 to 15 feet below the finished plant grade. The general subsurface profiles under the plant location are shown on FSAR Figures 2.5-72, 2.5-75, and 2.5-76.

The Reactor Buildings and Reactor Auxiliary Buildings are founded on a common mat over essentially fresh sandstone at Elevation 326. Based on subsurface exploration results (RQD values ranging from 90 to 100% and core recovery of 85 to 100%) the applicant has concluded and the staff concurs that there are no zones of alteration or irregular weathering or zones of structural weakness below Elevation 326.

The unconfined compression strengths of intact fresh standstone core samples range from 300 psi to 850 psi. The test values showed tangent modulus values at 50% ultimate strength for this material to range from 1×10^5 to 2.5×10^5 psi. The strength and compressiblity results for fresh sandstone are listed in Table 2.5-16 of the FSAR. We find these test results to be generally inconsistent with the higher values reported in the literature (e.g., Foundation Engineering Handbook by Winterkorn and Fang) for sandstone. However, since these values result in conservative analyses, we find them to be acceptable. The applicant has, also, listed on Table 2.5-16 and used in his analysis of some structures, Poisson's ratio value of 0.35 to 0.50 for fresh sandstone. We find this value of Poisson's ratio for rock material to be too high. The applicant should either provide further substantiation of his reasons for using this range of values or use an appropriate value of Poisson's ratio in his analysis.

The permeability of the fresh sandstone was determined by means of in-situ packer tests in two borings. In addition, field falling head and rising head permeability tests were conducted. The test results generally agreed with the in-situ packer test results. The highest permeability recorded in fresh sandstone was 7.5×10^{-6} cm/sec. We find this value of permeability for fresh sandstone to be reasonable.

The applicant determined the dynamic properties of the fresh sandstone from laboratory sonic velocity measurements and from field seismic refraction surveys and cross-hole shear-wave velocity measurements. The P-wave velocity

values recorded ranged from about 6000 to 8000 ft/sec, and the S-wave velocity values ranged from about 2500 to 4000 ft/sec. These results are shown on Table 2.5-16 of the FSAR. The staff finds these results to be reasonable and acceptable.

The foundations of Category I structures other than the Reactor Building and the Reactor Auxiliary Building are founded at or slightly below grade level (El. 390) and rest on weathered sandstone. The weathered sandstone has low to moderate hardness and is found by the applicant not to have any zones of structural weakness at the location of the plant site (as evidenced by core recovery of 85 to 100 percent and RQD of 90 to 100 percent).

The results of strength and compressibility tests on weathered sandstone are summarized in Table 2.5-15 of the FSAR. The unconfined compression strength of test samples are shown to range from 350 to 800 psi and the tangent modulus values at 50% ultimate strength range from 2.5×10^5 to 7.0×10^5 psi. We find these test results to be inconsistent with those reported in the literature. However, since the values result in conservative analyses, we find them to be acceptable. The field permeability tests indicate that the highest permeability coefficient recorded for weathered sandstone is 7.5×10^{-6} cm/sec which is the same as for the fresh sandstone. We find these permeability test results to be reasonable and acceptable.

The dynamic properties for weathered sandstone were investigated by the applicant through field siesmic refraction surveys and cross hole testing. These tests were supplemented by laboratory sonic velocity measurements on representative weathered sandstone samples. The field and laboratory dynamic test results are given in Table 2.5-15 of the FSAR. The results indicate that the P-wave velocity values range from about 5000 to 9000 ft/sec, and the S-wave velocity values for weathered sandstone range from about 2,300 to 4,000 ft/sec. We find these values to be reasonable.

The tuff beds below plant grade are composed of coarse to fine grained material and are about 7 to 11 feet thick. Based on strength testing, the applicant

determined that the relative hardness and engineering properties of the tuff do not differ significantly from those of the fresh sandstone. The applicant also did not find any evidence of bedding planes between the ruff layers and the sandstone.

2.5.4.1.3.2 Design Values of Subsurface Materials

Fresh and Weathered Sandstone

In Section 2.5.4.11 of the WNP-3 FSAR, the applicant has stated that representative values of compressive strengths of the fresh and weathered sandstone cores were based on the statistical method using results of the laboratory compression tests; however, the details of the method are not given. The applicant has also stated that he used reduction factors (RF) to arrive at the design compressive strengths for the two materials. Although the procedure to compute the RF value is briefly described, the computed values of (RF) and the corresponding final design values of the compressive strength are not given. In view of the wide variation in the results of the test data shown in Tables 2.5-15 and 2.5-16, the staff requires the applicant to provide, (i) the various steps used in statistical analyses to arrive at the representative compressive strengths along with assumptions and the results of the analyses, (ii) the values of $(V_p)_1$ and the computed values of (RF) along with the method of deriving values of $(V_p)_f$ and $(V_p)_1$ in these computations, and (iii) the computed values of design compressive strengths.

The applicant stated in the FSAR that a statistical method was used to calculate values of the representative tangent moduli from unconfined compression test results on selected core specimens; however, the details of these analyses are not given. Reduction factors (RF), similar to those used in computing the design compressive strengths were used to compute the design elastic moduli. Again, the details are not given and the procedures are not justified. The staff requires that the applicant explain in detail the statistical analyses, justify the bases for assumptions and provide the results of analyses. The computed values of design elastic moduli for fresh and weathered sandstone materials should also be presented.

In describing the procedure for arriving at the design values for Poisson's ratio, the applicant has stated in the FSAR that the values were "selected based on the comparative method using results of laboratory compression tests on selected core specimens." The staff requires the applicant to provide details of the so called 'comparative method' and to justify this method's use in selecting design Poisson's ratio values for the fresh and weathered sandstone materials. Although the range is given, the applicant should document and justify the values of the design Poisson's ratio used for these materials in the various plant designs and analyses.

In response to staff questions 241.10 and 241.23, the applicant stated that a selected value of modulus of subgrade reaction of 500 lb/in^3 was used in the static analysis (using MSC/NASTRAN computer program) for the Tank Enclosure Structure. In analytically deriving the value of this modulus of subgrade reaction the applicant used a shear modulus value of 330 ksi corresponding to strain value of 10^{-2} in./in. in his dynamic shear modulus versus strain curve. Further, the expression used for modulus determination is indirectly derived from an equation given by Barkan for vibratory loads and for the purpose of machine foundations design. We find that the applicant's use of the subject equation, the assumptions made in utilizing the equation and the resulting value of the modulus used in MSC/NASTRAN need further justification.

The applicant has selected design permeability values of 2×10^{-6} cm/sec and 6×10^{-6} cm/sec for fresh and weathered sandstone materials, respectively. These values are very close to the maximum recorded permeability measurements in the field and laboratory for these materials and, therefore, are reasonable and acceptable values to be used in design and analysis.

Dynamic shear wave velocities of 3,800 ft/sec for the fresh sandstone and 3,200 ft/sec for the weathered sandstone have been selected by the applicant to be used in the design and analysis of the plant. Little or no explanation is given in the FSAR in support of the selection of the values. In view of the wide variation, exhibited by test results, in the values of S-wave velocity data presented in FSAR Appendix 2.5 D, Tables 2.5-15 and 2.5-16, the applicant needs to further justify the use of these values for design.

The applicant performed one strain-controlled cyclic triaxial test to establish the design values and shape of shear modulus versus strain curve for fresh sandstone. To account for end effects and non-uniform strains within the test specimen, the average axial strain was obtained by dividing the measured axial strain by a correction factor of 4. Values of shear modulus were obtained at four different strain levels. Damping ratio for fresh sandstone was calculated for one strain level of 3×10^{-2}. Based on the results of measurements and interpretation of the test data, the applicant found that the shape of the shear modulus versus strain curve is different and steeper than the published curves for rock (e.g., Schnabel et al. 1972) and the calculated damping at 3×10^{-2} strain is also higher than corresponding values in the published curves. We find the applicant has not provided bases for selecting a damping ratio versus strain curve and has not sufficiently justified deviations from the published dynamic properties of fresh sandstone. Moreover, the applicant's results are based on only one dynamic test on fresh sandstone. Only a few different strain values were used to measure the shear modulus and damping. The correction factors used to divide measured axial strain have not been properly justified. For these reasons, applicant needs to provide further bases for selecting the shear modulus and damping ratio versus strain curves shown in Figures 2.5.121 and 2.5-122 of the FSAR.

For the weathered sandstone material, the applicant has used the same dynamic shear modulus versus strain curve as for the fresh sandstone; however, no explanation for this assumption is provided. A curve for weathered sandstone damping ratio versus strain is not given. The applicant should justify the design curves for dynamic properties of weathered sandstone and provide the bases for assumptions used in deriving the design curves.

Tuff Beds

Since the relative hardness and engineering properties of the tuff beds below grade were found by the applicant to be similar to those of the weathered sandstone, the static and dynamic design values for the tuff were selected as identical to those of the weathered sandstone. We find this assumption to be reasonable and acceptable.

Residual Soils

Although all residual soils were excavated from underneath the location of the plant structures, their engineering properties were established by the applicant for use in the evaluation of stability of man-made slopes east of the WNP-3 plant structures.

Based on interpretation of laboratory test results of seventeen unconsolidated-undrained triaxial tests, six consolidated-undrained triaxial tests and three unconfined compression tests, the applicant established three sets of strength parameters for the residual soils: maximum strength parameters of c = 1000 psf, Ø = 40°, minimum strength parameters of c = 750 psf, Ø = 26° and average strength parameters of c = 1000 psf, Ø = 30°. The applicant used all three sets in slope stability evalua' on. We consider the design c and Ø values for residual soils used by the applicant in analyses to be reasonable and the approach of using an appropriate variation of solid properties in the slope stability analyses to be acceptable.

2.5.4.1.4 Groundwater Conditions

The pre-construction groundwater conditions at the site were determined using borehole piezometer readings taken from April 1973 to September 1974 during the subsurface exploration. At that time, the groundwater levels in the WNP-3 powerbloc' area ranged from El 385 feet to El 411 feet.

During construction, the plant level was excavated to an elevation of 390 feet and the bottom of the common mat foundation was excavated to an approximate elevation of 326 feet. For this situation the temporary groundwater flows from relatively impermeable sandstone (permeability coefficient of approximately 2×10^{-6} cm/sec) were handled with a drainage system within the common mat excavation. Groundwater collected was drained from the excavation through a gravity flow to the slope south of the plant location. The groundwater conditions were monitored from October 1977 to December 1979 using piezometers around the WNP-3 excavation; the recorded levels during this period ranged from El 330 feet to 390 feet as shown on FSAR Figure 3.4.1-5.

The plant, has a permanently lowered watertable level to an elevation below the foundation mat level (below 326 feet) along the exterior faces of the category structures by means of a gravity drainage system. The drainage system is not Category I, as the walls and the mat of the Reactor Auxiliary Building are designed to withstand full hydrostatic loads (water table at El 365) that may be caused by a complete blockage of the drainage system.

The staff's evaluation of groundwater is presented in Section 2.4 of this SER.

2.5.4.2 Excavation and Backfill

All residual soil was excavated from the plant area. The excavation into rock for the powerblock area extended about 64 feet below the final grade level (390 feet above mean sea level). It penetrated the overlying weathered sandstone for a depth of about 60 feet and exposed the fresh sandstone surface at elevation 326 feet. Vertical cuts were made in sandstone formations. The vertical rock sides were cleaned by air jetting and protected against weathering by shortcreting over welded wire fabric. The bottom of excavation was covered by concrete mud mat.

The Category I structures other than the powerblock structures are located slightly below plant grade on weathered sandstone. All cuts for these structures were vertical. No backfill was required beneath or around seismic Category I structures since they were placed directly against fresh or weathered sandstone.

Class A1 Structural Fill, consisting of a well graded sand and gravel having a maximum size of 6 inches and a maximum of 15 percent passing the number 200 sieve, was used to backfill beneath, around and above seismic Category I buried pipe. When used as a bedding material for the pipes, backfill with the maximum particle size of 3/4 inch was used.

In the FSAR Section 2.5.4.2.6, the applicant nas stated that the Class A1 structural fill was compacted to a specification of at least 95 percent of the maximum modified Proctor density (ASTM 1557-78). The in-place density tests

were performed in each lift. The results indicated that less than 10 percent of the tested densities fell below 90 percent of the specified density. The staff finds these as-placed densities of the Class A1 structural fill beneath, around and above seismic Category I buried piping to be reasonable and acceptable.

In response to the staff question 241.22, the applicant has stated that there are other areas (including areas under and adjacent to Diesel Generator Fuel Oil Storage and Transfer System, and Class IE Duct Lines from Reactor Auxiliary Building to Dry Cooling Tower and Refueling Water Storage Tank Area) where placement of Class I structural fill has not been completed. The applicant has committed to provide for staff review the results of all Category I field density and moisture content tests performed under and adjacent to safety related structures as construction progresses. We find this commitment to be acceptable. The staff will review and evaluate the information when provided by the applicant.

The applicant used soil-cement (concrete sand mixed with 10 percent Type II Portland cement and 10.4% ± 2% moisture) compacted to a specification of 95 percent Standard Proctor Density (ASTM D558-57) as a backfill in the construction access ramps adjacent to the Reactor Auxiliary Building. In these areas the in-place density test results showed almost 100% compliance with the specified compaction requirement. We find these results to be acceptable.

2.5.4.3 Response of Rock to Seismic Loading

In Sections 2.5.2 and 2.5.4.7 of the WNP-3 FSAR, the applicant has stated that since the shear wave velocities in the underlying sandstone at the plant site is greater than 3000 ft/sec, there will be no amplification or modification of the input acceleration time histories at the plant site and the design earthquake would be defined by Regulatory Guide 1.60 response spectra anchored to the maximum design ground acceleration (0.32g) at the site. We concur with the applicant's assessment, given in FSAR Section 2.5.4.7, that the response of rock to seismic loading would not result in a modification of the input motion and find it acceptable. However, in Sections 3.7 of the same FSAR, the applicant stated that he used a deconvolution analysis through 570 ft of rock column which resulted in a substantial modification of the design motion and

resulted in base slab response spectra lower than R.G. 160 spectra in frequency range of interest. Thus, the information presented in the Section 3.7 of the FSAR is inconsistent with the information presented in Section 2.5.4.7. The procedure given in FSAR Section 3.7 for evaluating response of rock to seismic loading is not acceptable to the staff because input motion would not be substantially altered within the firm rock surrounding structures.

In response to staff questions 241.9 and 241.24, the applicant informed the staff that the depth of rock for deconvolution was based on the results of a sensitivity study in which the depth of rock column was gradually increased to determine the lower boundary of the analytical model until no difference in response of the building could be detected. However, the analytical parameters and the results of this study have not been provided to the staff for review. During an audit of the applicant's calculations on September 26 to September 30, 1983, the staff was verbally informed by the applicant's A/E (Ebasco) that the calculations and results pertaining to the said sensitivity study were not saved by the applicant's A/E.

Based on a review of the information provided in the FSAR Sections 2.5.2, 2.5.4.7, and 3.7, the applicant's response to our Q's and our audit findings, we conclude that the procedure used by the applicant in determining the response of the rock using deconvolution through 570 foot rock column is not acceptable. an assessment of the results of its application to structural seismic design and analyses is presented in Section 3.7 of this SER.

During the audit on September 26 - 30, 1983, the staff found that, in the NASTRAN computer program input for the soil-structure interaction analysis of the Reactor Auxiliary Building (RAB), a value of Poisson's ratio of 0.5385 was used. This is inconsistent with the design parameters shown on Table 2.5-16 of the FSAR. Moreover, a value of Poisson's ratio greater than 0.5 is not theoretically possible. The applicant should provide justification of this issue and assess the impact on analysis.

2.5.4.4 Foundation Stability

The Category I structures on a common mat (Reactor Building and Reactor Auxiliary Building/Fuel Handling Building) are supported on firm, fresh sandstone at a depth of 65 feet below finished plant grade (El 390 feet). The foundations for other Category I Structures, (Refueling Water Storage and Condensate Storage Tank Enclosure Structure Dry Cooling Tower Train A Structure and Control Building, Dry Cooling Tower Train B Structure, and Two Diesel Oil storage Tank Enclosures) rest on weathered sandstone slightly below El 390. The Category I buried pipelines are located on Class A1 structural fill overlying either sandstone or soil-cement underlain by sandstone.

Bearing Capacity

In FSAR Section 2.5.4.10.1, the applicant has stated that Terzaghi's bearing capacity formula (Modified for rock) was used to compute the ultimate bearing capacities for the Category I foundation mat. We find this state-of-the-art procedure to be acceptable for static bearing capacity calculations. However, the applicant has not provided any information about the procedure and assumptions for dynamic bearing capacity calculations. The staff requires this information along with the bases for arriving at the dynamic loads used in these analyses.

For the Reactor Auxiliary Building mat foundation static bearing capacity calculations, the applicant neglected the effect of cohesion and used a rupture angle of 20°. We find these rock properties assumptions to be reasonable and acceptable. In view of the foundation being supported on firm fresh sandstone 64 feet below grade, sufficient margin of safety against bearing capacity failure exists. However, we find that applicant's calculation results given in the FSAR Section 2.5.4.11.7 are inconsistent with those given in response to staff Question 241.3; the minimum factor of safety in the FSAR is stated to be 34, whereas it is stated to be 6.3 in response to Question 241.3. The applicant needs to appropriately amend his submittals to make them consistent and correct. The applicant should also provide for staff review, adequate information on the methods, assumptions and results of dynamic bearing capacity calculations.

The applicant has provided the results of static bearing capacity calculations for the Condensate/Refueling Water Tank foundation and Dry Cooling Tower and Control Building Foundation in response to staff Question 241.3. The values given for the factors of safety of the two buildings are 20.0 and 23.8, respectively, which we find to be adequate. However, for these computations, as well as for the dynamic bearing capacity calculations, the applicant has not provided the procedure and the rock properties used in the analysis. We require the applicant to provide the necessary details of the procedures used for static and dynamic bearing capacity calculations and the assumptions made in these analyses for staff review.

The applicant has not provided any information on the static and dynamic bearing capacity calculation procedures, assumptions and results of analyses for the following seismic Category I structures: Dry Cooling Tower Train B Structure, Two Diesel Oil Storage Tank Enclosure Structures and Category I Drainage Manholes. The necessary bearing capacity calculation results for these structures along with the procedures and assumptions should be provided for staff review.

The attached Table 2.2, provided by the applicant in response to staff question 241.3, shows pertinent details of static and dynamic loads on three seismic Category I structures. The staff requires the applicant to modify this table to make it consistent with the information provided in the FSAR. In addition, the applicant should include information on other seismic Category I structures to complete this Table, and submit the amended Table for staff review.

Settlement

In response to staff question 241.4, the applicant informed us that, since all seismic Category I structures were founded directly on either fresh or weathered sandstone, settlement or rebound was not considered by the applicant to be a factor in the design of the plant. The applicant did not and does not have any settlement monitoring of the plant foundations. Also, no information has been submitted for staff review on the potential or actual differential settlements between plant foundations and buried piping or duct run penetrations.

Using Boussinesq equation, the applicant has computed the value of estimated post-construction total settlement of the Reactor Auxiliary Building mat to be less than half inch. However, allowable settlements are not given in the FSAR. The applicant has also not provided for staff review estimated and allowable settlement values for other seismic Category I structures.

We do not agree with the applicant that construction and post-construction settlements for rock supported structures need not be considered in the design of structures, piping, and duct run penetrations because the stresses induced due to differential settlements may be significant. We require the applicant to provide for staff review the values of allowable differential settlements that the Category I buildings can withstand (in combination with other appropriate loads) and still meet code allowable stresses of the FSAR. In addition, since the applicant is not monitoring the actual total and differential settlements between various Category I foundations, he should assume a minimum of one-half inch differential settlement to check the design of piping running between rock supported structures. Piping and duct run penetrations should also be assessed for a minimum of one-half inch differential settlement. The results of these confirmatory analyses should be provided for staff review.

Lateral Pressures

The exterior walls of the Category I structures were placed directly against the vertically excavated rock face. The applicant computed the static lateral pressures resulting from (a) hydrostatic pressure due to possible failure of the groundwater drainage system around Category I structure (b) long term creep of sandstone causing active lateral pressures on exterior walls, and (c) the effect of the adjacent building surcharge at the ground surface causing lateral pressure on embedded walls. The applicant has stated in the FSAR that the Category I structure walls are designed for full hydrostatic pressure up to an elevation of 365 feet, incorporating a coefficient of active earth pressure of 0.22, corresponding to $\phi = 40°$, and using Boussinesq stress distribution to evaluate the effect of surcharge on the embedded exterior walls. We find these computational procedures and assumptions to be reasonable and acceptable. The

applicant stated in response to staff Question No. 241.17 that total static lateral pressure was determined to be . kips/ft^2. The applicant has, however, not provided the distribution of static lateral pressure along the depth of the walls. We require that the applicant submit this information for review.

The applicant computed the dynamic lateral pressure based on the effects of rock-structure interaction analysis. The dynamic lateral pressure obtained from this analysis and used in design was 10.27 kips/ft^2; the distribution of this pressure with depth of wall was not provided by the applicant for review. This information is needed to complete our SER.

As stated earlier in Section 2.5.4.3 of this SER, it is the staff's position that the modification of rock response due to deconvolution is not acceptable. Since the applicant used this procedure to compute seismic lateral earth pressures, the staff requested the applicant (Question 241.18) to re-compute these pressures without using deconvolution and by utilizing the state-of-the-art Seed and Whitman (1976) approach. The applicant has responded to this question (informally received by the staff on September 23, 1983). The applicant's computations using the Seed and Whitman approach and incorporating the effect of static lateral pressure, hydrostatic pressure and effect of surcharge load show the total dynamic lateral pressures to be 4.7 kips/ft^2. This value is less than one-half of the value used by the applicant in the design. These computation results are reasonable and acceptable to the staff. The applicant should, however, provide a comparison of the dynamic lateral pressure distribution along the depth of the seismic Category I walls using the above mentioned two approaches (rock-interaction analysis and Seed and Whitman procedure) for staff review.

Liquefaction Potential

There is no potential for liquefaction of sandstone that supports structures, systems and components. The compacted Class A1 backfill placed under, around and over the seismic Category I buried piping is well graded, has a maximum particle size of 6 inches and has been compacted to 95 percent modified Proctor density. The soil cement backfill placed at the location of the construction ramps (at 95% Standard Proctor density) has unconfined compressive strength of

approximately 600 lb/in², which is similar to that of the sandstone. The applicant has not considered the liquefaction potential of Class A1 structural fill or the soil cement backfill. We consider this approach to be reasonable, because as a result of the high compaction and compressive strength of these materials they can be considered to be not susceptible to liquefaction.

2.5.4.5 Conclusion

Based on the applicant's design criteria and construction reports and on the results of the applicant's site investigations, laboratory and field tests, and analysis, the staff has concluded that the site and plant foundations will be adequate to safely support the WPPSS Nuclear Project No. 3 (WNP-3) in accordance with the requirements of Appendix A to 10 CFR Part 100, pending satisfactory resolution of the open and confirmatory items identified above.

2.5.5 Stability of Slopes

The WNP-3 plant site is surrounded by natural rock slopes gently dipping to the south of the plant and cut rock slopes rising to the east of the site.

The natural rock slope to the south of the plant has an average slope away from the plant of 3 horizontal:1 vertical with a maximum slope of 1.5H:1V. The crest of the slope is more than 250 feet away from the edge of the power-block structures. A typical cross-section through the slope is shown on FSAR Figure 2.5-110. The material forming the slope essentially consists of weathered and fresh sandstone. The presence of a thin bed (<<10 feet) of residual soil near the toe of the slope was neglected by the applicant in the stability analysis of the natural slopes.

Typical cross-sections analyzed for stability analyses of cut rock slopes east of the plant are shown on FSAR figures 2.5-111 and 2.5-112. The slope generally rises at an average slope of 4.5H:1V with maximum slope of 3:5H:1V. The toe of the slope is more than 350 feet away from the edge of the Category I structures. The slope consists of weathered sandstone and residual soil.

As evidenced by the results of the site exploration, the applicant has determined that the sandstone at the plant site is massive, without continuous joints, seams or layers of weaker material. The applicant determined the static strength parameters of the weathered and fresh sandstone on the basis of 13 uniaxial compression tests. Based on these test results, a cohesion of 23 kips/ft^2 and an angle of internal friction of 0° were selected for analyses. We find these rock strength parameters to be reasonable and acceptable.

Shear strength parameters for residual soil to be used in stability analyses of cut slopes were obtained from seventeen unconsolidated-undrained triaxial tests, six consolidated-undrained triaxial tests and three unconfined compression tests. Based on these test results, the applicant selected the following properties for residual soil:

	Cohesion, C	Angle of internal friction, /
High	1000 lb/ft^2	40°
Average	1000 lb/ft^2	30°
Low	750 lb/ft^2	26°

We consider the applicant's use of these residual soil properties to be reasonable and acceptable.

<u>Natural Slope</u>

The static stability of the natural rock slope has been investigated by the applicant using the Simplified Bishop Method of Slices and the Sliding Wedge Method of Analysis. Two different groundwater conditions, viz., (i) normal groundwater level elevation of 320 feet (with drainage system operating) and (ii) groundwater level elevation of 365 feet (with drainage system blocked), are considered. The result of the applicant's analyses indicate that the natural slope has a minimum factor of safety of 5.5 for Slip Circle Method of Analysis and minimum factor of 7.6 for the Sliding Wedge Method of Analysis.

Based on these results, the staff concludes that for static design loads, the natural slope south of the plant is stable.

The applicant has made a seismic stability evaluation of the natural slope for SSE condition using Slip Circle and Sliding Wedge analysis approaches. In these analyses, a horizontal seismic coefficient of 0.32 and a vertical seismic coefficient of 0.22 were used. The applicant's results for these analyses indicate minimum factors of safety of 2.0 for Slip Circle and 2.64 for dynamic Wedge Method of Analysis. We consider that the margin of safety is adequate and acceptable.

Cut Slopes

The applicant has analyzed the cut rock slopes east of the plant structures using Bishop's Slip Circle Method and the Sliding Wedge Method of Analysis for static and dynamic cases. Two cross-sections shown on FSAR figures 2.5-111 and 2.5-112 have been analyzed. For seismic stability analyses, a seismic coefficient corresponding to 0.32 horizontal and 0.22 vertical were used for SSE.

The following minimum factors of safety were computed by the Applicant from these analyses of cut slopes:

	Factor of safety	
Method of analysis	Static case	Dynamic case
Slip circle	3.36	1.45
Wedge method	3.13	1.40

We find these factors of safety for stability of cut rock slopes to be acceptable.

The staff concludes that the natural and man-made slopes around the plant site have been analyzed by the applicant in an appropriate and reasonable manner,

and, based on the results of analyses presented by the applicant, the staff concludes these slopes have an adequate margin of safety, and meet the requirements of 10 CFR 100. The natural and cut slopes are, therefore, acceptable.

2.5.6 Embankments and Dams

There are no embankments or dams associated with the WNP-3 plant used for plant flood protection or for impounding cooling water required for operation of the plant.

Table 2.1 Resident population versus distance

year	0-1 miles	0-2 miles	0-3 miles	0-4 miles	0-5 miles	0-10 miles
1980	15	109	906	3061	5867	15165
1990	16	119	1000	3363	6519	16915
2030	20	163	1506	4989	10130	27103

Figure 2.1 Low population zone and area within 4 miles (_ m) of site

Figure 2.2 Principal plant features in relation to exclusion area and property lines

Figure 2.3 Area within 50 miles of site

Figure 2.4 Transportation routes and pipelines within 5 miles (_ m) of site

Table 2.2

Category I structure	Foundation dimensions (as-built)	Foundation elevations (MSL)	Description of foundation base and depth to fresh sandstone	Foundation loading		Allowable bearing capacity		Factor of safety	
				Static	Dynamic	Static	Dynamic	Static	Dynamic
Condensate/ refueling water tank foundation	171'-0 x 66'-0	Top El 390.00'	5'-0 thick mat on weathered sandstone 40' to fresh sandstone	2.5	6.0	50.0	50.0	20.0	8.3
Dry cooling tower and control bldg.	85'-0 x 272'-0	Top El 390.00'	4'-0 thick mat on weathered sandstone 35' to fresh sandstone	2.1	4.6	50.0	50.0	23.8	10.5
Shield bldg. containment, internal structure, fuel handling bldg. on common mat	298'-0 x 310'-0	Top El 315.00'	9'-0 thick mat located in fresh sandstone	13.6	17.7	85.0	85.0	6.3	4.8

3 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

3.1 General

3.2 Classification of Structures, Systems, and Components

3.2.1 Seismic Classification

3.2.2 System Quality Group Classification

3.3 Wind and Tornado Criteria and Loadings

3.3.1 Wind Design Criteria

All Category I structures exposed to wind forces were designed to withstand the effects of the design wind. The design wind specified has a velocity of 105 mph based on a recurrence of 100 years.

The procedures that were 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 ASCE Paper 3269 and ANSI - A58.1-1972. These documents are acceptable to the staff.

The staff concludes that the plant design is acceptable and meets the requirements of General Design Criterion 2. This conclusion is based on the following:

The applicant has met the requirements of GDC 2 with respect to the capability of the structures to withstand design wind loading so that their design reflects

(1) appropriate consideration for the most severe wind recorded for the site with an appropriate margin;

(2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena; and

(3) the importance of the safety function to be performed.

The applicant has met these requirements by using ANSI A58.1 and ASCE paper No. 3269, which the staff has reviewed and found acceptable, to transform the wind velocity into an effective pressure on structures and for selecting pressure coefficients corresponding to the structures geometry and physical configuration.

The applicant has designed the plant structures with sufficient margin to prevent structural damage during the most severe wind loadings that have been determined appropriate for the site so that the requirements of Item 1 listed above are met. In addition, the design of seismic Category 1 structures, as required by Item 2 listed above, has included in an acceptable manner load combination which occur as a result of the most severe wind load and the loads resulting from normal and accident conditions.

The procedures used to determine the loadings on structures induced by the design wind specified for the plant are acceptable since these procedures have been used in the design of conventional structures and proven to provide a conservative basis which together with other engineering design considerations assures that the structures will withstand such environmental forces. The use of these procedures provides reasonable assurance that in the event of design basis winds, the structural integrity of the plant structures that have to be designed for the design wind will not be impaired and, in consequence, safety-related systems and components located within these structures are adequately protected and will perform their intended safety functions if needed, thus satisfying the requirement of Item 3 listed above.

3.3.2 Tornado Design Criteria

All Category I structures exposed to tornado forces and needed for the safe shutdown of the plant were designed to resist a tornado of 240 mph tangential wind velocity and a 60 mph translational wind velocity. The simultaneous

atmospheric pressure drop was assumed to be 2.25 psi in 1.9 seconds. Tornado missiles are also considered in the design as discussed in Section 3.5 of this report.

We conclude that the plant design is acceptable and meets the recommendations of Standard Review Plan 3.3.2 and the requirements of General Design Criterion 2. This conclusion is based on the following:

The applicant has met the recommendations of Standard Review Plan 3.3.2 and the requirements of GDC 2 with respect to the structure capability to withstand design tornado wind loading and tornado missiles so that their design reflects

(1) appropriate consideration for the most severe tornado recorded for the site with an appropriate margin;

(2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena; and

(3) the importance of the safety function to be performed.

The applicant has met these requirements by using ANSI A58.1 and ASCE Paper No. 3269, which the staff has reviewed and found acceptable, to transform the wind velocity generated by the tornado into an effective pressure on structures and for selecting pressure coefficients corresponding to the structures geometry and physical configuration.

The applicant has designed the plant structures with sufficient margin to prevent structural damage during the most severe tornado loadings that have been determined appropriate for the site so that the requirements of Item 1 listed above are met. In addition, the design of seismic Category 1 structures, as required by Item 2 listed above, has included in an acceptable manner load combinations which occur as a result of the most severe tornado wind load and the loads resulting from normal and accident conditions.

The procedures utilized to determine the loadings on structures induced by the design basis tornado specified for the plant are acceptable since these

procedures have been used in the design of conventional structures and proven to provide a conservative basis which together with other engineering design considerations assures that the structures will withstand such environmental forces.

The use of these procedures provides reasonable assurance that in the event of design basis tornado, the structural integrity of the plant structures that have to be designed for tornadoes will not be impaired and, in consequence, safety-related systems and components located within these structures will be adequately protected and may be expected to perform necessary safety functions as required, thus satisfying the requirement of item 3 listed above.

3.4 Water Level (Flood) Design

3.4.1 Flood Protection

The design of the facility for flood protection was reviewed in accordance with Section 3.4.1 of the Standard Review Plan (SRP), NUREG-0800. An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for the staff's evaluation of the design of the facility for flood protection with respect to the applicable regulations of 10 CFR 50.

To ensure conformance with the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," the staff reviewed the overall plant flood protection design including all systems and components whose failure due to flooding could prevent safe shutdown of the plant or result in the uncontrolled release of significant radioactivity. The applicant has provided protection from inundation and the static and dynamic effects of flooding for safety-related structures, systems, and components by providing "hardened protection" in accordance with the guidelines of Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants." The plant site is a "dry site" as defined in Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants," Position C.1.

The source of flooding at the site is the probable maximum flood (PMF) in the Chehalis River. The water level at the site vicinity resulting from the PMF in the river is 76.2 ft MSL, approximately 313 ft below plant grade. Because all safety-related systems and components are located at the plant grade that is well above the highest PMF level, they are not subjected to flooding concerns resulting from the PMF. Refer to Section 2.4.2 of this SER for further discussion of site flooding caused by local intense precipitation.

The reactor auxiliary building (RAB), fuel handling building, and reactor building are protected against flooding as a result of groundwater seepage by the installation of a permanent groundwater drainage system (GWDS). The GWDS permanently lowers the groundwater in the vicinity of the plant. Watertight seals are also provided on all below-grade penetrations of the RAB to further limit groundwater seepage into the building. The dry cooling towers and refueling water storage tank structures are at plant grade and thus are not susceptible to flooding as a result of groundwater seepage.

The GWDS is not classified as seismic Category I. The applicant has stated that this classification is adequate since a failure of the GWDS (clogging of the drain pipes) during a seismic event would not cause an appreciable rise in the groundwater level for a minimum of 115 days. In addition, the GWDS will be inspectable to ensure proper functioning at any time, including after an earthquake. Refer to Section 2.4.12 of this SER for further discussion regarding the GWDS.

Within safety-related plant structures, protection against flooding from failures in fluid piping systems as identified in the guidelines of Branch Technical Position ASB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," is provided by equipment location and drainage as described under Sections 3.6.1 and 9.3.3 of this SER.

On the basis of its review of the design criteria and bases, and the safety classification of safety-related systems, structures, and components necessary for a safe plant shutdown during and following flood conditions, the staff concludes that the design of the facility for flood protection conforms to the requirements of General Design Criterion 2 with respect to protection against natural phenomena and conforms to the guidelines of Regulatory Guides 1.59 and

1.102 concerning flood protection. Therefore, the flood protection design meets the acceptance criteria of SRP Section 3.4.1 and is acceptable. The staff further concludes that the CESSAR interface requirements are satisfied by the above described design.

3.4.2 Water Level (Flood) Design Procedures

The design flood level resulting from the most unfavorable condition or combination of conditions that produce the maximum water level at the site is discussed in Section 2.4, Hydrology. The hydrostatic effect of the flood was considered in the design of all Category I structures exposed to the water head.

With the exception noted at the end of this section we conclude that the plant flood structural design procedures are acceptable and meet the recommendations of Standard Review Plan 3.4.2 and the requirements of General Design Criterion 2. This conclusion is based on the following:

The applicant has met the recommendations of Standard Review Plan 3.4.2 and the requirements of GDC 2 with respect to the capability to withstand the effects of the floor or highest groundwater level so that their design reflects

(1) appropriate consideration for the most severe flood recorded for the site with an appropriate margin;

(2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena; and

(3) the importance of the safety function to be performed.

The applicant has designed the plant structures with sufficient margin to prevent structural damage during the most severe flood or groundwater and the associated dynamic effects that have been determined appropriate for the site so that the requirements of Item 1 listed above are met. In addition, the design of seismic Category I structures, as required by Item 2 listed above, has included in an acceptable manner loac combinations which occur as a result of the most severe flood or groundwater-related loads and the loads resulting from normal and accident conditions.

The procedures used to determine the loadings on seismic Category I structures induced by the design flood or highest groundwater level specified for the plant are acceptable since these procedures have been used in the design of conventional structures and proven to provide a conservative basis which together with other engineering design considerations assures 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, in consequence, seismic Category I systems and components located within these structures will be adequately protected and may be expected to perform necessary safety functions, as required, thus satisfying requirement of item 3 listed above.

The applicant has used PVC as a Waterstop material in the RAB (question 220.11 and reply). Since such material is subject to radiation deterioration and attendant production of destructive chemical reactions, the staff will require additional studies to show that these materials, as used, will not be hazardous to the plant structures and will perform their intended functions throughout the lifetime of the plant. It was noted that the reply to question 220.11 is not complete. The applicants answer was compared with information shown on drawing WPSS-3240, G-2520-S1, "RAB Internal Structures - Sheet 1." In that drawing reference is made to silicone rubber joint sealer, non-specific material premolded joint filler, polyethylene and epoxy grout. A complete discussion of this issue, which demonstrates that all of the organic materials used in waterstop and structural joint filler applications are satisfactory to resist chemical and radiation deterioration, is required.

3.5 Missile Protection

3.5.1 Missile Selection and Description

3.5.1.1 Internally Generated Missiles (Outside Containment)

The design of the facility for providing protection from internally generated missiles (outside containment) was reviewed in accordance with Section 3.5.1.1

of the Standard Review Plan (SRP), NUREG-0800. An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria except as noted below formed the basis for the staff's evaluation of the design of the facility for providing protection from internally generated missile outside containment with respect to the applicable regulations of 10 CFR 50.

The acceptance criteria for the design of the facility for providing missile protection includes meeting Regulatory Guide 1.115, "Protection Against Low-Trajectory Turbine Missiles." The review of turbine missiles is discussed separately in Section 3.5.1.3.

General Design Criterion 4, "Environmental and Missile Design Bases," requires protection of plant structures, systems, and components, whose failure could lead to offsite radiological consequences or that are required for safe plant shutdown, against postulated missiles associated with plant operation. The missiles considered in this evaluation include those missiles generated by rotating or pressurized (high-energy fluid system) equipment.

Protection is provided by any one or a combination of compartmentalization, barriers, separation, orientation, and equipment design. The primary means of providing protection to safety-related equipment from damage resulting from internally generated missiles is through the use of plant physical arrangement. Safety-related systems and components of safety-related systems are physically separated from their redundant components.

The applicant has provided an evaluation of potential missile sources from rotating equipment failures and pressurized component failures. The potential missiles resulting from this analysis are valves in high energy systems. The applicant's evaluation has verified that plant design features such as walls or separation of redundant systems will prevent these missiles from causing adverse effect on safety-related systems and components. Other missile sources are precluded by the design of the equipment itself. The staff concurs with the applicant's assumptions and evaluation for potential missiles outside containment. Protection of safety-related equipment and stored fuel from the effects of turbine missiles is discussed in Section 3.5.1.3 of this SER.

The potential sources of missiles which were evaluated in the fuel handling building are considered to be generated from failure of either a pressurized component or a rotating component. There are no high energy systems located within the fuel handling building and therefore missiles from pressurized components are not postulated. The only rotating pieces of equipment in the fuel handling building are the component cooling water pumps, fuel pool cooling pumps, and the fuel pool cleanup pumps. All of these pumps and their motors are located at elevations below the spent fuel pool and are separated by seismic Category I barriers which prevent any missiles from penetrating the spent fuel pool.

In addition, the staff requested the applicant to provide assurance that turbine driven
pumps would not become a source of missiles or that missiles from the pump turbine could not damage safety-related equipment. There are two types of turbine driven pumps at the plant, the steam generator feedwater pumps (nonsafety-related) and the auxiliary feedwater pumps (safety-related). The steam generator feedwater pumps incorporate redundant overspeed protection devices and both the turbine and pump casings are designed of sufficient strength to prevent the release of missiles generated by failure of the rotor or impeller. In the unlikely event that a missile penetrated the casing, the steam generator feedwater pumps are oriented such that the path of the missile would be away from safety-related components. Further, each of the two trains of the auxiliary feedwater system is located in a separate concrete cubicle containing one motor driven and one turbine driven pump. Thus, the plant design incorporates physical separation of trains A and B components with sufficient redundancy to ensure safe shutdown of the plant.

The staff concludes that the above described design satisfies CESSAR interface requirements. [However, the applicant's analysis does not address pressurized tanks and gas cylinders (≥275 psig) from becoming potential missiles. Therefore, the staff cannot conclude that the design is in conformance with the requirements of General Design Criterion 4 as it relates to protection against internally generated missiles until the applicant provides additional information in this regard. Resolution of this item will be reported in the final SER.]

3.5.1.2 Internally Generated Missiles (Inside Containment)

The design of the facility for providing protection from internally generated missiles inside containment was reviewed in accordance with Section 3.5.1.2 of the Standard Review Plan (SRP), NUREG-0800. An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for the staff's evaluation of the design of the facility for providing protection from internally generated missiles with respect to the applicable regulations of 10 CFR 50.

All plant structures, systems and components (SSC) inside containment whose failure could lead to offsite radiological consequences or that are required for safe plant shutdown must be protected against the effects of internally generated missiles in accordance with the requirements of General Design Criterion 4, "Environmental and Missile Design Bases." Potential missiles that could be generated inside containment are from failures of rotating components, pressurized components (high-energy fluid system) failures and gravitational effects.

With regard to potential missiles from pressurized high-energy systems inside the containment, the applicant has analyzed the primary missiles that can be generated in the reactor vessel head area. The missiles considered in this context were the closure head nut, closure head nut and stud and the control rod drive assembly. The applicant's analysis verified that structures and shields provide protection for safety-related equipment from the above primary missiles. Also, potential gravitational missiles inside the containment resulting from seismic events are prevented by either designing the structures, systems and components located inside the containment as seismic Category I or by designing them to withstand seismic Category I loads without falling.

With regard to potential missile sources from rotating equipment, the applicant has verified that all HVAC rotating equipment located inside containment is designed to withstand the impact of self generated missiles such as fans or impeller blades by fabricating the equipment housing with sufficient material thickness. Also, either duct reinforcement or missile barriers have been

provided at the discharge of the fans to contain the generated missiles and additionally prevent the generation of secondary missiles outside the HVAC rotating equipment housing. For a discussion of compliance with the criteria of Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity," as it relates to potential missile sources, refer to Section 5.4.1.1 of this SER and of the CESSAR SER.

The applicant has stated that temperature sensors or other detectors installed on pipes or in wells, nuts, bolts, studs, and combinations thereof contribute insignificantly to missile hazards due to the low amount of stored energy. [However, the applicant has not provided specific information regarding protection against other potential primary system high-energy missile sources identified in CESSAR FSAR Table 3.5-1 and in the CESSAR SER Section 3.5.1.2. Additionally, the applicant has not provided information on secondary missiles generated by the impact of primary missiles associated with high-energy systems. Therefore, the staff cannot conclude that the WNP-3 design is in conformance with the requirements of General Design Criterion 4 as it relates to protection against internally generated missiles inside the containment. Resolution of this item will be reported in the final SER.]

3.5.1.3 Turbine Missiles

We have reviewed the WNP Unit 3 facility with regard to the turbine missile issue and conclude that the probability of unacceptable damage to safety-related systems and components due to turbine missiles is acceptably low (i.e., less than 10^{-7} per year) provided that the turbine missile generation probability is maintained to be 10^{-4} per reactor year or less for the life of the plant by an acceptable maintenance program. In reaching this conclusion, the staff has factored into consideration the favorable orientation of the turbine generator.

The staff considers the turbine missile issue as a confirmatory item if the applicant agrees to:

(1) submit for NRC approval, within three years of obtaining an operating license, a turbine system maintenance program based on the manufacturer's calculations of missile generation probabilities, or

(2) volumetrically inspect all low pressure turbine rotors at the second refueling outage and every other (alternate) refueling outage thereafter until a maintenance program is approved by the staff, and conduct turbine steam valve maintenance, (following initiation of power output) in accordance with present NRC recommendations as stated in SRP Section 10.2 of NUREG-0800.

3.5.1.4 Missiles Generated by Natural Phenomena

The tornado missile spectrum was reviewed in accordance with Section 3.5.1.4 of the Standard Review Plan (SRP), NUREG-0800. An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section except as noted below. Conformance with the acceptance criteria formed the basis for the staff's evaluation of the tornado missile spectrum with respect to the applicable regulations of 10 CFR 50.

The portions of the "Review Procedures" concerning the probability per year of damage to safety-related systems due to missiles was not used in the staff's review. The staff's review for this section of the SRP is concerned with establishing the missile spectrum, not with calculating the probability of damage.

General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," requires that structures, systems, and components important to safety be designed to withstand the effects of natural phenomena, and General Design Criterion 4, "Environmental and Missile Design Bases," requires that these same plant features be protected against missiles. The missiles generated by natural phenomena of concern are those resulting from tornadoes. The applicant has identified a spectrum of missiles for a tornado Region III site as identified in Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants," Positions C.1 and C.2. The spectrum includes the weight, velocity, kinetic energy, impact area, and height in accordance with current tornado missile criteria. The staff has reviewed this spectrum and concludes that it is representative of missiles at the site and is, therefore, acceptable. Discussion of the protection (barriers and structures) afforded to safety-related equipment from the identified tornado missiles including compliance with the guidelines of

Regulatory Guide 1.117, "Tornado Design Classification," is provided in Section 3.5.2 of this SER. Discussion of the adequacy of barriers and structures designed to withstand the effects of the identified tornado missiles is provided in Section 3.5.3 of this SER.

On the basis of its review of the tornado missile spectrum, the staff concludes that the spectrum was properly selected and meets the requirements of General Design Criteria 2 and 4 with respect to protection against natural phenomena and missiles and the guidelines of Regulatory Guide 1.76 with respect to identification of missiles generated by natural phenomena and is, therefore, acceptable. The tornado missile spectrum meets the acceptance criteria of SRP Section 3.5.1.4. The staff further concludes that the above described design satisfies the CESSAR interface requirements.

3.5.2 Structures, Systems, and Components to be Protected from Externally Generated Missiles

The design of the facility for providing protection from tornado-generated missiles was reviewed in accordance with Section 3.5.2 of the Standard Review Plan (SRP), NUREG-0800. An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for the staff's evaluation of the design of the facility for providing protection from tornado generated missiles with respect to the applicable regulations of 10 CFR 50.

General Design Criterion (GDC) 2, "Design Basis for Protection Against Natural Phenomena," requires that all structures, systems, and components important to safety be protected from the effects of natural phenomena, and GDC 4, "Environmental and Missile Design Bases," requires that all structures, systems, and components important to safety be protected from the effects of externally generated missiles. The WNP-3 site is located in tornado Region III as identified in Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants." The tornado missile spectrum is discussed in Section 3.5.1.4 of this SER. Protection from low-trajectory turbine missiles including compliance with RG 1.115, "Protection Against Low-Trajectory Turbine Missiles", is discussed in Section 3.5.1.3 of this SER.

The applicant has identified all safety-related structures, systems, and components requiring protection from externally generated missiles. All safety-related structures are designed to withstand postulated tornado generated missiles without damage to safety-related equipment. Safety-related systems and components and stored fuel and spent fuel pool are located within tornado-missile protected structures or are provided with tornado missile barriers. The two dry cooling towers which constitute the ultimate heat sink for WNP-3 are enclosed in structures designed to prevent tornado and missile impact damage to any vital component of the towers. The cooling tower fans, particularly, are protected from tornado generated missiles by missile grating. Therefore, the staff concludes that the guidelines of Regulatory Guides 1.13, "Spent Fuel Storage Facility Design Basis," 1.27, "Ultimate Heat Sink for Nuclear Power Plants," and 1.117, "Tornado Design Classification," concerning tornado missile protection for stored fuel, ultimate heat sink and the spent fuel pool are met. With regard to HVAC openings, the outside air HVAC intakes for the control room, the fuel building, diesel generator (DG) area, and the electrical equipment and battery rooms are all protected from tornado-missiles by protective missile grating. Also, the component cooling water system dry cooling towers electrical equipment room outside air HVAC intake and exhaust openings are protected against tornado-missiles by missile grating. Additionally, the applicant states that the DG combustion air intake opening is protected from external missiles by shield bars and that both the normal and emergency combustion air exhaust path openings are protected against externally generated missiles. [However, the applicant has not provided assurance that HVAC exhaust openings such as those for the control room, fuel building, ECCS area/fuel building emergency filtration system, diesel generator area, and the electrical equipment and battery rooms are protected from tornado-missiles. Also, the staff is unable to conclude that possible damage to the DG combustion air exhaust assembly (for example, the silencer) from tornado-missile, such as crimping will not disable the DG. Therefore, the staff cannot conclude that the requirements of GDCs 2 and 4 with respect to missile protection and the guidelines of RG 1.117 concerning tornado-missile protection for safety-related structures, systems and components are met. Resolution of this concern will be reported in the final SER.]

[On the basis of the preceding, the staff concludes that, except as noted above, the applicant's list of safety-related structures, systems and components to be

protected from externally generated missiles and the provisions in the plant design providing this protection are in accordance with the requirements of GDCs 2 and 4 with respect to missile protection and the guidelines of Regulatory Guides 1.13, 1.27, and 1.117 as they relate to tornado missile protection for safety-related structures, systems and components including stored fuel and ultimate heat sink. The staff therefore concludes that the design meets the acceptance criteria of SRP Section 3.5.2 except as noted above. Also, the staff cannot conclude that the design meets the intent of CESSAR interface requirements. Resolution of the concerns identified above will be reported in the final SER.]

3.5.3 Barrier Design Procedures

The plant Category I structures, systems and components are shielded from, or designed for, various postulated missiles. Missiles considered in the design of structures include tornado generated missiles and various containment internal missiles, such as those associated with a loss-of-coolant accident.

Information has been provided indicating that the procedures that were used in the design of the structures, shields and barriers to resist the effect of missiles are adequate. The analysis of structures, shields and barriers to determine the effects of missile impact was accomplished in two steps. In the first step, the potential damage that could be done by the missile in the immediate vicinity of impact was investigated. This was accomplished by estimating the depth of penetration of the missile into the impacted structure. Furthermore, secondary missiles are prevented by fixing the target thickness well above that determined for penetration. In the second step of the analysis, the overall structural response of the target when impacted by a missile is determined using established methods of impactive analysis. The equivalent loads of missile impact, whether the missile is environmentally generated or accidentally generated within the plant, are combined with other applicable loads as is discussed in Section 3.8 of this report.

We conclude that the barrier design is acceptable and meets the recommendations of Standard Review Plan 3.5.3 and the requirements of General Design Criteria 2 and 4 with respect to the capabilities of the structures, shields, and barriers to provide sufficient protection to equipment that must withstand the effects

of natural phenomena (tornado missiles) and environmental effects including the effects of missiles, pipe whipping and discharging fluids. This conclusion is based on the following:

The procedures utilized to determine the effects and loadings on seismic Category I structures and missile shields and barriers induced by design basis missiles selected for the plant are acceptable since these procedures provide a conservative basis for engineering design to assure that the structure or barriers are adequately resistant to and will withstand the effects of such forces.

The use of these procedures provides reasonable asurance that in the event of design basis missiles striking seismic Category I structures or other missile shields and barriers, the structural integrity of the structures, shields and barriers will not be impaired or degraded to an extent that will result in a loss of required protection. Seismic Category I systems and components protected by these structures are, therefore, adequately protected against the effects of missiles and will perform their intended safety function, if needed. Conformance with these procedures is an acceptable basis for satisfying in part the requirements of General Design Criteria 2 and 4.

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

3.6.1 Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment

The design of the facility for providing protection against postulated piping failures outside containment was reviewed in accordance with Section 3.6.1 of the Standard Review Plan (SRP), NUREG-0800. An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for the staff's evaluation of the design of the facility for providing protection against postulated piping failures outside containment with respect to the applicable regulations of 10 CFR 50.

The staff's guidelines for meeting the requirements of General Design Criterion 4, "Environmental and Missile Design Bases," concerning protection against postulated piping failure in high-energy and moderate-energy fluid systems outside containment are contained in Branch Technical Position (BTP) ASB 3-1, "Protection Against Postulated Failures in Fluid Systems Outside Containment." The applicant has identified high- and moderate-energy piping systems in accordance with these guidelines and has also identified those systems requiring protection from postulated piping failures (refer to Section 3.6.1 of the CESSAR SER for a discussion of the high- and moderate-energy fluid systems outside containment which are in the CESSAR scope).

The plant design accomodates the effects of postulated pipe breaks and cracks including pipe whip, jet impingement and environmental effects. The means used to protect essential (safety-related) systems and components include physical separation, enclosure within suitably designed structures, pipe whip restraints, and equipment shields. To be consistent with BTP ASB 3-1, the applicant has utilized separation as the primary means of protection, and where separation was not feasible, one of the other acceptable methods of protection was used.

The plant design includes the ability to sustain a high-energy pipe break accident coincident with a single active failure and retain the capability for safe cold shutdown. For postulated pipe failures, the resulting effect will not cause the loss of function of power supplies or controls and instrumentation needed to complete a safety action and will not preclude the habitability of the control room as indicated in BTP ASB 3-1.

The applicant has also analyzed the effects of moderate-energy line breaks outside containment on safety-related systems by postulating cracks in moderate energy lines at any location. For moderate-energy essential system piping cracks in other than dual purpose moderate-energy essential systems which satisfy the guidelines of BTP ASB 3-1, Position 3.b.(3), a single active failure in the redundant train or trains of the essential system was also considered and it was shown that safe shutdown will not be affected or the functional capability of the essential systems will not be compromised. [However, the staff cannot accept the applicant's assumption that a seismic event concurrent with a crack in non-seismically designed moderate-energy piping is not a credible event since

the seismic event by itself can cause a pipe break in a non-seismically designed piping system. Also, it cannot grant credit for mitigation of flooding consequences resulting from postulated seismically induced pipe failures by non-seismically designed systems, components or equipment such as floor drainage systems, sump pumps etc. Therefore, the staff cannot conclude that moderate-energy systems have been designed to meet the intent of the guidelines set forth in BTP ASB 3-1. Resolution of this concern will be reported in the final SER.]

The main steam and feedwater systems up to the first restraint outside containment are classified as part of the break exclusion (BEX) boundary as defined in item B.1.6 of BTP MEB 3-1, "Postulated Breaks and Leakage Locations in Fluid System Piping Outside Containment." At the staff's request, the applicant provided the results of a subcompartment analysis of a nonmechanistic break in these lines to determine the environmental effects in the compartments housing the main steam and feedwater lines. The applicant determined that the structural integrity of the applicable steam tunnel (there are 2 steam tunnels) which houses the BEX portion of these lines will not be affected by the pressure increase from the resulting blowdown. The tunnel is vented to relieve the pressure effects. Main steam isolation and feedwater isolation valves (MSIVs and FWIVs) functional capability will be maintained by assuring that they are environmentally qualified to conservative bounding conditions determined by the analysis. The staff concurs with this analysis. Environmental qualification of essential auxiliary feedwater (AFW) system pumps and flow control/isolation valves and AFW turbine steam supply valves and essential equipment located in the steam tunnel including the MSIVs and FWIVs and the atmospheric dump valves is discussed in Section 3.11 of this SER. [The applicant has not provided a pressure and environmental analysis for the other subcompartments outside containment which house high-energy piping (the CVCS charging and letdown, steam generator (SG) blowdown and auxiliary steam lines). The staff evaluation of the results of the analysis to assure that safety-related equipment is protected from the postulated failure in these piping systems will be provided in the final SER.]

[On the basis of its review described above, the staff cannot conclude that the applicant has adequately designed and protected areas and systems required for safe plant shutdown following postulated failures in high- and moderate-energy

piping outside containment as required by GDC 4 until it has completed its review of the subcompartment pressure and environmental analysis for the CVCS charging and letdown lines, SG blowdown lines and auxiliary steam lines, and until its concern relating to moderate-energy piping identified above is resolved. The resolution of all these concerns will be discussed in the final SER. CESSAR interface requirements (refer to CESSAR SER, Section 3.6.1) specify that safety-related equipment must be protected from the effects of high- and moderate-energy pipe failures. Therefore, the staff cannot conclude that these requirements have been met by the applicant until the review of applicant's responses to the concerns identified above has been completed.]

3.6.2 Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

3.7 Seismic Design

3.7.1 Seismic Input

The peak accelerations associated with SSE were selected based on the seismicity evaluation described in Section 2.5 of the FSAR. The earthquake on which SSE is modeled has a Richter magnitude of 7-1/2, and originates at a distance of approximately 22 miles from the site. The peak horizontal baserock acceleration at the site associated with this earthquake is 0.32 g (SSE). The duration of strong motions for this earthquake is estimated to be approximately 30 seconds. The operating basis earthquake (OBE) is chosen to be 1/2 SSE or 0.16 g. Vertical accelerations are 2/3 of horizontal, that is, 0.22 g SSE and 0.11 g OBE.

Horizontal design response spectra conform to the recommendations of Regulatory Guides 1.60 and 1.61. The vertical design response spectra does not comply with the recommendations of Regulatory Guide 1.60 in the 33 to 50 Hz range. As discussed in DSER Section 2.5.2, the Geosciences Branch is reviewing the above seismic input assumptions and will report conclusions in the SER.

Although the duration of the earthquake record was to be 30 seconds it was found that the duration of earthquake was truncated to 20 seconds in the horizontal direction (Structural Design Audit Finding #5). This matter is under investigation at this writing.

The horizontal and vertical acceleration time histories were derived by applying a deconvolution methodology to a finite element model of the rock site. The staff disagrees with the results of this analysis (question 220.13, and Structural Design Audit Finding #1). Essentially, the applicant has allowed a significant reduction in the earthquake excitation at the base mat based on his deconvolution analysis of a rock site. This issue remains unresolved as of this writing.

3.7.2 Seismic System Analysis

3.7.3 Seismic Subsystem Analysis

The scope of review of the Seismic System and Subsystem Analysis for the plant included the seismic analysis methods for all 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 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 Category I structures and piping and effects of parameter variations on floor response spectra. The review has also included criteria and seismic analysis procedures for reactor internals and Category I buried piping outside the containment.

The system and subsystem analyses were performed by the applicant on an elastic bases. Modal time history methods form the bases for the analyses of all major Category I structures. Response spectra methods were used in the design of Category I systems and components.

The finite element approach is used to evaluate soil-structure interaction and structure to structure interaction effects from seismic excitation.

As noted above in Section 3.7.1 of this report, the seismic acceleration input at the base mat used by the applicant is in question. (See questions 22J.15

and 220.16). In addition several issues regarding the construction and use of floor response spectra are also unresolved at this time. These issues are:

(1) Derivation of floor response spectra without consideration of out-of-plane acceleration factors (See question 220.18 and Audit Funding #17). It is noted that the torsional-effects analysis described in paragraph 3.7.2.11 of the FSAR was not made available to the audit team at the Structural Design Audit. It is hereby requested that the applicant make this analysis available to the staff as part of his resolution of this issue.

(2) Actual application of methods for peak- roadening and smoothing of response spectra are not in accordance with methods recommended by Regulatory Guide 1.122 (to which the applicant is committed). See question 220.22, 220.23, and Structural Design Audit Findings #11 and #19.

3.7.4 Seismic Instrumentation Program

The type, number, location and utilization of strong motion accelerographs to record seismic events and to provide data on the frequency, amplitude and phase relationship of the seismic response of the containment structure were compared with Regulatory Guide 1.12 requirements. Supporting instrumentation is to be installed on Category I structures, systems and components in order to provide data for the verification of the seismic responses determined analytically for such Category I items.

The ranges of the types of instrumentation as well as the readout locations have not been provided. A seismic surveillance scheme as outlined in the SRP was not provided although it is said to be incorporated in the technical specifications for the plant (See question 220.19). The technical specifications for the plant were not available, as of this writing, in order that the surveillance scheme could be verified.

3.8 Design of Seismic Category I Structures

3.8.1 Concrete Containment

3.8.2 Steel Containment

The containment consists of a free-standing steel shell located within a reinforced concrete shield building. These are founded on a common mat with (but separated by seismic gaps above the mat) a reactor auxiliary building. The containment was designed, fabricated, constructed and tested as a Class MC vessel in accordance with Subsection NE of the ASME Boiler and Pressure Vessel Code, Section III. Loads include an appropriate combination of dead and live loads; thermal loads; seismic and loss-of-coolant accident-induced loads including pressure and jet forces.

The analysis of the containment was based on elastic thin shell theory. The allowable stress and strain limits are in accordance with those delineated in the applicable sections of Subsection NE of the ASME Code, Section III, for the various loading conditions.

The following issues are unresolved at this time:

(1) Question 220.25, which is a request for drawings, remains unanswered.

(2) Question 220.26 which requests validation of computer programs used in the containment design remains unanswered. An advance copy of the applicants proposed answer was provided to the staff in reply to portion (c) of Structural Design Audit Finding #4. However, the staff considers this proposed answer to be incomplete and lacking in specific details.

(3) Portions (a) and (b) of Structural Design Audit Finding #4 regarding compliance with ASME code requirements and buckling analysis remain unresolved.

(4) The answer to question 220.28 is not responsive. Neither a complete description nor results of calculations for the containment static analysis was provided.

(5) Question 220.27 was a staff request for an ultimate capacity analysis of the containment in accordance with the criteria contained in the SRP. The answer provided by the applicant indicated that the ultimate capacity analysis would not be prepared.

(6) According to Table 1.8-3 of the FSAR a Design Report meeting the guidelines set forth in Appendix C to SRP Section 3.8.4 was prepared for the containment. The staff was to examine the design report at the Structural Design Audit but did not do so. Therefore, the applicant is requested to make the Design Report available for staff review. This comment also applies to SER sections 3.8.3, 3.8.4 and 3.8.5.

(7) Question 220.30 regarding use of plastic filler materials remains unanswered.

3.8.3 Concrete and Structural Steel Internal Structures

The containment interior structures consist of walls, compartments and floors. The major code used in the design of concrete internal structures is ACI 318-71. For steel internal structures the AISC Specification, "Specification for the Design, Fabrication and Erection of Structural Steel for Building," is used. (For equipment supports, Subsection NF of the ASME Code is used.)

The containment concrete and steel internal structures were designed to resist various combinations of dead and live loads, accident induced loads, including pressure and jet loads, and seismic loads. The load combinations used cover those cases likely to occur and include all loads which may act simultaneously. The design and analysis procedures that were used for the internal structures are the same as those on previously licensed applications and, in general, are in accordance with procedures delineated in the ACI 318-71 Codes and in the AISC Specification for concrete and steel structures, respectively.

The containment internal structures were designed and proportioned to remain within limits established by the Regulatory staff under the various load combinations. These limits are, in general, based on the ACI 318-71 Code and on the AISC Specification for concrete and steel structures, respectively, modified as appropriate for load combinations that are considered extreme.

The material of construction, their fabrication, construction and installation, are in accordance with the ACI 318-71 Code and AISC Specification for concrete and steel structures, respectively.

The response to question 220.35 is unsatisfactory. Differences between the application of ACI 318-71 and the staffs referenced design standard consisting of Reg. Guide 1.142 and ACI 349 were not documented. All that is stated in Table 1.8-3 is that ACI 318-71 was used in accordance with PSAR commitments. This is not considered to satisfy the applicant's obligation to document deviations from NUREG 0800. This comment applies to SER sections 3.8.4 and 3.8.5 as well.

3.8.4 Other Seismic Category I Structures

Category I structures other than containment and its interior structures are all of structural steel and concrete. The structural components consist of slabs, walls, beams and columns. The major code used in the design of concrete Category I structures is the ACI 318-71, "Building Code Requirements for Reinforced Concrete." For steel Category I structures, the AISC, "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," is used.

The concrete and steel Category I structures were designed to resist various combinations of dead loads; live loads, environmental loads including winds, tornadoes, OBE and SSE; and loads generated by postulated ruptures of high energy pipes such as reaction and jet impingement forces, compartment pressures, and impact effects of whipping pipes.

The design and analysis procedures that were used for these Category I structures are the same as those approved on previously licensed applications and, in general, are in accordance with procedures delineated in the ACI 318-71 Codes and in the AISC Specification for concrete and steel structures, respectively.

The various Category I structures are designed and proportioned to remain within limits established by the Regulatory staff under the various load

combinations. These limits are, in general, based on the ACI 318-71 Code and not the AISC Specification for concrete and steel structures, respectively, modified as appropriate for load combinations that are considered extreme.

The materials of construction, their fabrication, construction and installation are in accordance with the ACI 318-71 Code and the AISC Specification for concrete and steel structures, respectively.

With respect to Other Category I structures, the following issues remain outstanding:

(1) Structural Design Audit Finding #18 regarding the arbitrary reduction of seismic accelerations obtained from the NASTRAN analysis of the shield building remains unresolved at this time.

(2) Structural Design Audit Finding #21 regarding use of negative signs on load factors in the shield building analysis is not resolved.

(3) Structural Design Audit Finding #22 regarding computer printouts with compressive stresses undefined in the output is not resolved.

(4) Structural Design Audit Finding #23 regarding verification of the static vs. dynamic analysis of the dry cooling tower is not resolved.

3.8.5 Foundations

Foundations of Category I structures are described in Section 3.8.5 of the SAR. Primarily, these foundations are reinforced concrete of the mat type. The major code used in the design of these concrete mat foundations is ACI 318-71. These concrete foundations have been designed to resist various combination of dead loads, live loads, environmental loads including winds, tornadoes, OBE and SSE, and loads generated by postulated ruptures of high energy pipes.

The design and analysis procedures that were used for these Category I foundations are the same as those approved on previously licensed applications and,

in general, are in accordance with procedures delineated in the ACI 318-71 Code. The various Category I foundations were designed and proportioned to remain within limits established by the Regulatory staff under the various load combinations. These limits are, in general, based on the ACI 349 Code modified as appropriate for load combinations that are considered extreme. The materials of construction, their fabrication, construction and installation, are in accordance with the ACI 318-71 Code.

3.8.6 Structural Audit

Structural Design Audit Finding #24 regarding foundation uplift in the stability analysis remains outstanding.

General Comments (Section 3.8)

(1) Structural Design Audit Finding #25 regarding the applicants documentation of load and load combination values in the various analyses remains unresolved.

(2) The staff has not completed its review of all of the answers provided by the applicant to staff questions. Further issues may therefore become apparent as these reviews are completed.

3.11 Environmental Qualification of Electrical Equipment Important to Safety and Safety-Related Mechanical Equipment

3.11.1 Introduction

Equipment that is used to perform a necessary safety function must be demonstrated to be capable of maintaining functional operability under all service conditions postulated to occur during its installed life for the time it is required to operate. This requirement, which is embodied in GDC 1 and 4 and in Sections III, XI, and XVII of Appendix B to 10 CFR 50, is applicable to equipment located inside as well as outside containment. More detailed requirements and guidance relating to the methods and procedures for demonstrating this capability for electrical equipment have been set forth in 10 CFR 50.49,

"Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants," and NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety Related Electrical Equipment." NUREG-0588 supplements IEEE Standard 323 and various NRC Regulatory Guides and industry standards.

3.11.2 Background

NUREG-0588 was issued in December 1979 to promote a more orderly and systematic implementation of electrical equipment qualification programs by industry and to provide guidance to the NRC staff for use in ongoing licensing reviews. The positions contained in this section provide guidance on (1) how to establish environmental service conditions, (2) how to select methods that are considered appropriate for qualifying equipment in different areas of the plant, and (3) other factors such as margin, aging, and documentation.

In February 1980, the NRC requested the Washington Public Power Supply System to review and evaluate the environmental qualification documentation for each item of safety-related electric equipment which could be exposed to a harsh environment and to identify the degree to which their qualification program complies with the staff positions described in NUREG-0588. IE Bulletin 79-01B, "Environmental Qualification of Class 1E Equipment," issued January 14, 1980, and its supplements dated February 29, September 30, and October 24, 1980, established environmental qualification requirements for operating reactors. This bulletin and its supplements were provided to the applicant for consideration in his review.

A final rule on environmental qualification of electric equipment important to safety for nuclear power plants became effective on February 22, 1983. This rule, Section 50.49 of 10 CFR 50, specifies the requirements to be met for demonstrating the environmental qualification of electrical equipment important to safety located in a harsh environment. In accordance with this rule, equipment for WNP-3 may be qualified to the criteria specified in Category I of NUREG-0588.

The qualification requirements for mechanical equipment are principally contained in Appendices A and B of 10 CFR 50. The qualification methods defined in NUREG-0588 can also be applied to mechanical equipment.

In response to the above requirements, the applicant has provided some preliminary equipment qualification information in Section 3.11 of the FSAR and a submittal dated April 28, 1981.

3.11.3 Completeness of the Environmental Qualification Program

In order to demonstrate compliance with the final rule, 10 CFR 50.49, the following information must be submitted by the applicant before an operating license can be granted.

In accordance with the scope defined in 10 CFR 50.49, provide:

(1) A list of all nonsafety-related electrical equipment located in a harsh environment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions by the safety-related equipment. A description of the method used to identify this equipment must be included. The nonsafety-related equipment identified must be included in the environmental qualification program.

(2) A statement that all safety-related electric equipment in a harsh environment, as defined in the scope of 10 CFR 50.49, is included in the equipment qualification program (including equipment required for MELB, fuel handling accident, etc.).

(3) A list of all Category 1 and 2 post-accident monitoring equipment currently installed, or to be installed before plant operation, in response to Regulatory Guide 1.97, Revision 2. The equipment identified must be included in the environmental qualification program.

Also provide information demonstrating qualification of all equipment in a harsh environment within the scope of 10 CFR 50.49, or provide justification for interim operation pending completion of qualification as required by 10 CFR 50.49. This material should be submitted to allow sufficient time for staff review and approval before issuance of an operating license.

Although there are no detailed requirements for mechanical equipment, GDC 1 and 4; Sections III and XVII of Appendix B to 10 CFR 50; and SRP 3.11,

Revision 2, contain the following requirements and guidance related to equipment qualification:

- Components shall be designed to be compatible with the postulated environmental conditions, including those associated with LOCAs.

- Measures shall be established for the selection and review for suitability of application of materials, parts, and equipment that are essential to safety-related functions.

- Design control measures shall be established for verifying the adequacy of design.

- Equipment qualification records shall be maintained and shall include the results of tests and materials analyses.

In order to demonstrate compliance with General Design Criterion 4 of Appendix A to 10 CFR Part 50 for mechanical equipment, the staff requires that the applicant perform a review and evaluation that includes the following:

(1) Identification of safety-related mechanical equipment located in harsh environmental areas, including required operating time.

(2) Identification of the nonmetallic subcomponents of this equipment.

(3) Identification of the environmental conditions for which this equipment must be qualified. The environments defined in the electrical equipment program are also applicable to mechanical equipment.

(4) Identification of nonmetallic material capabilities.

(5) Evaluation of environmental effects.

The list of equipment identified should be submitted. From this list the staff will select approximately three items of mechanical equipment for which documentation of their environmental qualification should be provided for review. Also, the results of the review should be provided for all

mechanical equipment in harsh environment areas and corrective actions identified. Justification for interim operation must be submitted prior to fuel load for any mechanical equipment whose qualification cannot be established.

For mechanical equipment, the staff review will concentrate on materials which are sensitive to environmental effects for example; seals, gaskets, lubricants, fluids for hydraulic systems, diaphragms, etc.

Additionally, all safety-related equipment should be subjected to a maintenance, surveillance and periodic testing program in accordance with Reg. Guide 1.33, to detect any age-related degradation that could affect the qualification of the equipment and to maintain the equipment in a qualified condition.

Upon receipt of a final submittal, the staff will review the environmental qualification program for compliance and request any additional information needed to establish its acceptability. The staff will then perform an audit review of electrical equipment environmental qualification files and associated installed equipment. Following this audit, an SER supplement will be prepared documenting the results of the review and evaluation. Prior to granting of an operating license, the staff must be able to conclude that full compliance with 10 CFR 50.49 and all applicable rules and regulations has been demonstrated.

4 REACTOR

4.1 Introduction

4.2 Fuel Design

WNP-3 is a CESSAR System 80 plant. We have reviewed the CESSAR fuel system and concluded (Rubenstein, June 8, 1983) that the CESSAR fuel system has been designed so that (a) the fuel system will not be damaged as a result of normal operation and anticipated operational occurrences, (b) fuel damage during postulated accidents would not be severe enough to prevent control rod insertion when it is required, and (c) core coolability will always be maintained, even after severe postulated accidents, and thereby meets the related requirements of 10 CFR Part 50.46; 10 CFR Part 50, Appendix A, General Design Criteria 10, 27, and 35, 10 CFR Part 50, Appendix K, and 10 CFR Part 100. This conclusion is based on the following:

(1) Combustion Engineering has provided sufficient evidence that these design objectives will be met based on operating experience, prototype testing, and analytical predictions. Those analytical predictions dealing with control rod ejection and fuel densification have been performed in accordance with the guidance of Regulatory Guide 1.77 and with an acceptable alternative of Regulatory Guide 1.126 (Ref. 56).

(2) Combustion Engineering has provided for testing and inspection of new fuel to ensure that it is within design tolerances at the time of core loading. The NRC will require (Rubenstein, February 17, 1983) that applicants (a) make commitments to perform CEA reactivity checks and post-irradiation surveillance to detect anomalies or confirm that the fuel has performed as expected and (b) provide assurance of adequate shoulder gap clearance.

(3) Combustion Engineering has described methods of adequately predicting fuel rod failures during postulated accidents so that radioactivity releases are not underestimated and thereby meets the related requirements of 10 CFR Part 100. In meeting these requirements, C-E has (a) used the fission-product release assumptions of Regulatory Guides 1.25 (Ref. 57) and 1.77, and an acceptable (more conservative) alternative to 1.4 (Ref. 58), and (b) performed the analysis for fuel rod failures for the rod ejection accident in accordance with the guidelines of Regulatory Guide 1.77.

On the basis of the NRC's review of the fuel system design, the NRC concluded (Rubenstein, June 8, 1983) that the CESSAR fuel system design has met all the requirements of the applicable regulations, regulatory guides, and current regulatory positions.

All applicants referencing the CESSAR FSAR (including the WPPSS application for WNP-3) must supply the following applicant-specific information (Rubenstein, June 8, 1983):

(1) A CEA surveillance program (see paragraphs 4.2.1.1(j) and 4.2.3.1(j)). In Q490.2 (Berlinger, April 14, 1983), WPPSS was asked to provide this information. No response has been received.

(2) A fuel assembly loads analysis due to combined seismic and LOCA forces (see paragraph 4.2.3.3(d)). In Q490.2, WPPSS was asked to provide this information. We have not yet received a response.

(3) A commitment to perform a general fuel surveillance program (see paragraph 4.2.4.3). In Q490.2, WPPSS was asked to provide this information. We have not yet received a response.

(4) Certification that the mechanical fracturing analysis result conforms to the acceptance criterion (see paragraph 4.2.3.2(g)). In Q490.2, WPPSS was asked to provide this information. We have not yet received a response.

In addition, the following license condition will be required to address the concern discussed in Section 4.2.3.1(g) related to axial growth.

> "The licensee shall confirm that adequate shoulder gap clearance will be maintained during the first two refueling outages (Cycles 1/2 and 2/3). This may be done either by analysis or hardware modification and shall be based on measurements taken on a sufficient number of fuel assemblies irradiated in WNP-3."

4.3 Nuclear Design

The Nuclear design of WNP-3 is identical to the corresponding item in the CESSAR (Combustion Engineering Standard Safety Analysis Report).

4.4 Thermal-Hydraulic Design

The CESSAR SER discusses all items in this section except for the staff review of: (1) the loose parts monitoring system; (2) the instrumentation for detection of inadequate core cooling requirements as described in item II.F.2 of NUREG-0737 and; (3) the plant-specific information on the Core Protection Calculators (CPCs) and statistical combination of uncertainties (SCU).

4.4.1 Loose Parts Monitoring System

The acceptance criteria for the Loose Parts Monitoring System (LPMS) are set forth in Regulatory Guide 1.133, Revision 1, "Loose-Part Detection Program for the Primary System of Light Water Reactors," May 1981. The applicant has provided a description of the LPMS which will be used at WNP-3. The design will include pairs of pieso-electric accelerometers located at the following natural collection regions: (1) hot leg nozzles adjacent to steam generator inlet plenum; (2) incore instrument nozzle located on the vessel bottom beam, adjacent to vessel inlet plenum; (3) Control Element Drive Mechanism (CEDM) nozzle located on vessel head. The system will provide eight channels with the capability of monitoring and detecting loose parts impacting within 3 feet of a sensor having a kinetic energy of 0.5 ft-lb and weighing from 0.25 to 30 lb. The system will remain functional following an Operating Basis Earthquake (OBE).

In response to the staff's request for additional information on the LPMS system, the applicant responded (Letter from G. C. Sorensen to G. W. Knighton, September 2, 1983) that the LPMS will be in full compliance with the requirements of Regulatory Guide 1.133. This will include the submission of a training program for plant personnel prior to power operation which will address Technical Specifications, purpose and operation of the LPMS system. The applicant has committed to provide Technical Specifications for the LPMS in accordance with Position C.5 of Regulatory Guide 1.133 which will specify the limiting conditions for operation and the surveillance requirements. We will also require the applicant to commit to provide prior to power operation a final design report showing conformance with positions C.1 through C.6 of Regulatory Guide 1.133 which contains the following:

(1) An evaluation of the LPMS for conformance to Regulatory Guide 1.133.

(2) A description of the system hardware, operation and implementation of the loose parts detection program after start-up testing. This should also include the baseline data and alarm settings.

(3) A description and evaluation of diagnostic procedures used to confirm the presence of a loose part.

A sample table of contents of the LPMS description is enclosed in Table 4-1.

4.4.2 Inadequate Core Cooling Requirements (SRP-4.4-Section II.9)

The applicant has not responded to questions asked relative to inadequate core cooling requirements as specified in Item II.F.2 of NUREG-0737 and this is therefore an open item.

4.4.3 Plant Specific Information (SRP 4.4-Section II.1)

The applicant has not responded to a question asked relative to plant-specific information on: (1) the application of statistical combination of uncertainties (SCU) and plant-specific instrument uncertainties; and (2) digital core protection calculator (CPC) including the Reactor Power Cutback System (RPCS) for

which information is needed on the plant-specific data base constants, software implementation testing and effects of SCU on DNBR. This is therefore an open item.

4.6 Functional Design of Reactivity Control Systems

The functional design of reactivity control systems is within the scope of CESSAR. Refer to Section 4.6 of the CESSAR SER for this discussion.

Table 4.1 Loose Part Detection Program Description

I. System Description

A. Scale piping diagram showing LPM sensor locations.
B. Sensor specifications (type, manufacturer, sensitivity, temperature rating, etc.).
C. Sensor mounting details (drawing and procedure).
D. Preamplifier or line driver (type, manufacturer, location and specifications).
F. Functional description of LPMS.
   1. Theory of operation, detection logic, alarm display.
   2. Data recorder specifications (No. of channels, length of recording, frequency range, and conditions under which recording is initiated).

II. Operational Procedures

A. System Calibration Procedures and Results
   1. Initial and subsequent calibrations
   2. Functional check, as defined in Regulatory Guide 1.133
   3. Channel check, as defined in Regulatory Guide 1.133

B. Plant Operator Instructions for Use of LPMS
   1. Procedures for routine operation
   2. Procedures to be used following indication of a loose part
      a. Method to confirm existence of loose part
      b. Method of diagnose a loose part (size and location)

III. Evaluation for Conformance to Regulatory Guide 1.133

A. Loose Part Detection Program
B. Loose Part Detection System

5 REACTOR COOLANT SYSTEM

5.1 Summary Description

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.4 Reactor Coolant Pressure Boundary Inservice Inspection and Testing

This section was prepared with the technical assistance of DOE contractors from the Idaho National Engineering Laboratory.

5.2.4.1 Compliance with the Standard Review Plans

The July 1981 Edition of the "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (NUREG-0800) includes Section 5.2.4, "Reactor Coolant Pressure Boundary Inservice Inspection and Testing." The Washington Nuclear Project No. 3 (WNP-3) review is continuing because the Applicant has not submitted a Preservice Inspection (PSI) Program and has not completed the PSI examinations. In FSAR Table 1.8-3, the Applicant has committed to comply with the Standard Review Plan (SRP) 5.2.4 acceptance criteria. The staff review to date was conducted in accordance with SRP Section 5.2.4 except as discussed below.

Paragraph II.3, "Acceptance Criteria, Examination Categories and Methods," will be reviewed when the complete PSI Program has been received from the Applicant.

Paragraph II.4, "Acceptance Criteria, Inspection Intervals," has not been reviewed because this area applies only to inservice inspections (ISI), not to PSI. This subject will be addressed during review of the ISI program after licensing.

Paragraph II.5, "Acceptance Criteria, Evaluation of Examination Results," has been reviewed. The Applicant committed in the FSAR to incorporate ASME Code Section XI, Article IWB-3000, "Standards for Examination Evaluation," into the PSI Program. However, ongoing NRC generic activities and research projects indicate that the presently specified ASME Code procedures may not always be capable of detecting the acceptable size flaws specified in the IWB-3000 acceptance standards. For example, ASME Code procedures specified for volumetric examination of reactor vessels, bolts and studs, and piping have not proven to be capable of detecting the acceptable size flaws in all cases. The staff will continue to evaluate the development of new or improved procedures and will require that these improved procedures be made a part of the inservice examination requirements. The Applicant's repair procedures based on ASME Code Section XI, Article IWB-4000, "Repair Procedures," have not been reviewed. Repairs are not generally necessary in the PSI program. This subject will be addressed during the staff review of the ISI program.

Paragraph II.7, "Acceptance Criteria, Code Exemptions," will be reviewed when the completed PSI Program Plan is submitted by the Applicant.

Paragraph II.8, "Acceptance Criteria, Relief Requests," has not been completed because the Applicant has not identified all limitations to examination. Specific areas where ASME Code examination requirements cannot be met will be identified as performance of the PSI progresses. The complete evaluation of the PSI program will be presented in a supplement to this Safety Evaluation Report (SER) after the Applicant submits the required examination information, identifies all plant-specific areas where ASME Code Section XI requirements cannot be met, and provides a supporting technical justification.

5.2.4.2 Examination Requirements

General Design Criterion 32, "Inspection of Reactor Coolant Pressure Boundary," Appendix A of 10 CFR Part 50 requires, in part, that components which are part of the reactor coolant pressure boundary be designed to permit periodic examination and testing of important areas and features to assess their structural and leak-tight integrity. To ensure that no deleterious defects develop during service, selected welds and weld heat-affected-zones (HAZ) will be examined

periodically. The design of the ASME Code Class 1 and 2 components of the reactor coolant pressure boundary incorporates provisions for access for inservice examinations, as required by Paragraph IWA-1500 of Section XI of the ASME Code. Section 50.55a(g), 10 CFR Part 50, defines the detailed requirements for the preservice and inservice programs.

Based upon the construction permit date of April 11, 1978, this section of the regulations requires that a preservice inspection program be developed and implemented using at least the Edition and Addenda of Section XI of the ASME Code applied to the construction of the particular components. The components (including supports) may meet requirements set forth in subsequent editions of this Code and Addenda which are incorporated by reference in 10 CFR 50.55a(b) subject to the limitations and modifications listed therein.

The initial ISI program must comply with the requirements of the latest Edition and Addenda of Section XI of the ASME Code in effect twelve months prior to the date of issuance of the operating license, subject to the limitations and modifications listed in Section 50.55a(b) of 10 CFR Part 50.

5.2.4.3 Evaluation of Compliance with 10 CFR 50.55a(g)

Review has been completed on the information presented in the FSAR through Amendment 3 dated April 1983. The preservice examination on the piping and components, except NSSS components, will be examined in accordance with the requirements of the 1977 Edition of ASME Code Section XI with Addenda through Summer 1978.

The NSS components will be examined in accordance with the requirements of the 1974 Edition of ASME Code Section XI with Addenda through Summer 1975 except that the steam generator tubing will be examined in accordance with ASME Code Section XI 1980 Edition with Addenda through Winter 1980.

The Preservice Inspection (PSI) Program for systems and components within the reactor coolant pressure boundary has not been received. However, the Applicant has stated in the FSAR that these systems and components will be examined per

the applicable Code requirements. Based on the review of the FSAR, the staff has established technical positions that should be included in the PSI Program.

The Applicant has committed to identify all plant-specific areas where the Code requirements cannot be met after the examinations are performed and provide a supporting technical justification for requesting relief. The SER input will be completed after the Applicant:

(1) Dockets a complete and acceptable PSI Program,

(2) Submits the requested additional information regarding the PSI/ISI program, and

(3) Submits all relief requests with a supporting technical justification.

The staff considers the review of the PSI Program an open issue subject to the Applicant providing an acceptable response to the above requirements.

The initial Inservice Inspection Program has not been submitted by the Applicant. This program will be evaluated after the applicable ASME Code Edition and Addenda can be determined based on Section 50.55a(b) of 10 CFR Part 50, but before inservice inspection commences during the first refueling outage.

5.2.4.4 Conclusions

The conduct of periodic examinations and hydrostatic testing of pressure-retaining components of the reactor coolant pressure boundary, in accordance with the requirements of Section XI of the ASME Code and 10 CFR Part 50, will provide reasonable assurance that structural degradation or loss of leak-tight integrity occurring during service will be detected in time to permit corrective action before the safety functions of a component are comprised. Compliance with the preservice and inservice examinations required by the Code and 10 CFR Part 50 constitutes an acceptable basis for satisfying the inspection requirements of General Design Criterion 32.

5.2.4.5 References

1. NUREG-0800, Standard Review Plans, Section 5.2.4, "Reactor Coolant Boundary Inservice Inspection and Testing," July 1981.

2. Code of Federal Regulations, Volume 10, Part 50.

3. American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI, Division 1.

   1974 Edition, through Summer 1975 Addenda
   1977 Edition, through Summer 1978 Addenda
   1980 Edition, through Winter 1980 Addenda

5.2.5 Reactor Coolant Pressure Boundary Leakage Detection

The reactor coolant pressure boundary (RCPB) leakage detection systems were reviewed in accordance with Section 5.2.5 of the Standard Review Plan (SRP), NUREG-0800. An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for the staff's evaluation of the reactor coolant pressure boundary leakage detection systems with respect to the applicable regulations of 10 CFR 50.

A limited amount of leakage is to be expected from components forming the reactor coolant pressure boundary (RCPB). Means are provided for detecting and identifying this leakage in accordance with the requirements of General Design Criterion (GDC) 30, "Quality of Reactor Coolant Pressure Boundary." Leakage is classified into two types - identified and unidentified. Components such as valve stem packing, pump shaft seals, and flanges are not completely leak tight. Since this leakage is expected, it is considered as identified leakage and is monitored, limited, and separated from other leakage (unidentified) by directinq it to closed systems as identified in the guidelines of Position C.1 of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems." Refer to CESSAR SER Section 5.2.5 for discussion on the sources, disposition, and indication of identified leakage.

Unidentified leakage, which includes steam generator tube sheet and intersystem leakage, is monitored by several devices as identified in the guidelines of Positions C.1, C.3 and C.4 of Regulatory Guide 1.45. Steam generator tube leakage is monitored by the condenser air removal system radiation monitors, steam generator blowdown system radiation monitors, or routine steam generator water samples. The method of detection of intersystem leakage depends on the particular interfacing system. Leakage of reactor coolant through the safety injection system can be identified by high pressure alarms in the control room. In the event seat leakage takes place past two shutdown cooling isolation valves, the leakage will pressurize the shutdown cooling lines and lift the two relief valves. The discharge from the relief valves is directed to the safety injection system recirculation sump and monitored as an unidentified leakage source.

The means of detecting intersystem leakage of primary coolant to the component cooling water system through the letdown heat exchanger, reactor coolant pump seal heat exchanger and thermal barriers is as follows. Heat exchanger leaks will produce inleakage of reactor coolant and fission products into the cooling water. Such inleakage will increase the radioactivity content of the cooling water. The increase will be detected by the component cooling water system radiation monitors located in the recirculation lines across the component cooling water pumps of each train. Leakage of reactor coolant also increases the inventory in the component cooling water system, causing an increase in the surge tank level which would result in a high level alarm in the main control room.

Leakage to the primary rector containment from unidentified sources is collected and the flow rate monitored with an accuracy of 1 gpm or better. Indication of unidentified leakage into the containment is monitored by four independent methods:

(1) Sump Level and Flow Monitoring

Unidentified leakage inside the containment including condensate from the containment fan coolers will flow to the containment drain sump. Leakage from the reactor coolant system (RCS) will result in either an increase in humidity in containment (which will cause condensation on the fan cooler

coils) or water on the floor. Thus, RCS unidentified leakage will pass to the containment sump. All flow entering the sump is routed first to a measurement tank. The tank is fitted with a level transmitter that sends a signal proportional to the tank level to the main control room. An alarm occurs whenever the equivalent of one gpm in one hour is exceeded as prescribed by Regulatory Guide 1.45, Positions C.2 and C.5. Sump level and flow monitoring equipment will remain functional after being subject to an SSE.

Unidentified leakage inside the reactor cavity will be collected in the reactor cavity sump and will be pumped directly into the measurement tank at the containment drain sump. Pump start alarm, sump level alarm, and flood detection alarms are provided for the reactor cavity area to alert the operator in case of any leakage into the area.

(2) Airborne Particulate Radioactivity Monitoring

The containment atmosphere is monitored for radioactive particulates by the containment atmosphere/containment purge airborne radiation monitors. These monitors are a pair of identical and redundant units. The particulate channel in each monitor is capable of detecting the airborne radioactive particulates resulting from an increase of one gpm in the leakage rate from the primary coolant pressure boundary into the containment atmosphere within one hour. In addition, the particulate filter tape and the downstream iodine filters may be removed for laboratory analysis. The monitoring equipment used for leakage detection has been designed to remain functional following an SSE as indicated in guidelines of Regulatory Guide 1.45, Position C.6.

(3) Airborne Gaseous Radioactivity Monitoring

The containment atmosphere is monitored for radioactive gases by the containment/atmosphere purge airborne radiation monitors. These monitors are a pair of identical and redundant units. The gaseous channel in each monitor is capable of detecting the airborne radioactive gases resulting from an increase of one gpm in the leakage rate from the primary coolant

pressure boundary into the containment atmosphere within one hour. The airborne gas monitoring equipment used for leakage detection has been designed to remain functional following an SSE.

(4) <u>Reactor Coolant Inventory Monitoring</u>

Abnormal leakage from the reactor coolant system is also detected through measurement of the net amount of makeup flow to the system (refer to CESSAR SER Section 9.3.4 and Section 5.2.5 for further discussion).

As described above, the RCPB leakage flow and radioactive monitors within containment are seismic Category I, testable, and may be calibrated as identified in the guidelines of Positions C.6, C.7, and C.8 of Regulatory Guide 1.45. Further, their accuracy meets the guidelines of Position C.5 of Regulatory Guide 1.45.

Additional sources of indication of unidentified leakage include containment pressure, temperature, and humidity indicators, pressurizer level indicators, and low pressure safety injection header pressure. Technical specifications will include limiting conditions for identified and unidentified leakage and will also address availability of the various leakage detection systems to assure adequate coverage of all times as indicated in Regulatory Guide 1.45, Position C.9.

On the basis of the preceding, the staff concludes that the RCPB leakage detection systems are diverse and provide reasonable assurance that primary system leakage (both identified and unidentified) will be detected. The systems meet the requirements of GDC 30 with respect to provisions for RCPB leak detection and identification, and the guidelines of Regulatory Guide 1.45 with respect to RCPB leakage detection system design. The staff therefore concludes that the design is acceptable and that it meets the acceptance criteria of SRP Section 5.2.5. It further concludes that the CESSAR interface requirements, as discussed in CESSAR SER Section 5.2.5, are satisfied by the above-described design.

5.3 Reactor Vessel

5.3.1 Reactor Vessel Materials (Fracture Toughness)

In this section we have reviewed the fracture toughness of ferritic reactor vessel and reactor coolant pressure boundary materials, and the materials surveillance program for the reactor vessel beltline. The acceptance criteria and references which are the basis for this evaluation are set forth in Paragraph 11.3.a of Standard Review Plan (SRP) Section 5.2.3 and Paragraphs II.5, II.6 and II.7 (Appendices G and H, 10 CFR Part 50) of SRP Section 5.3.1 in NUREG-0800 Rev. 1, dated July 1981. A discussion of this review follows.

GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," of Appendix A to 10 CFR Part 50, requires that the reactor coolant pressure boundary be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and anticipated transient conditions, the boundary behaves in a non-brittle manner and the probability of rapidly propagating fracture is minimized. GDC 32, "Inspection of Reactor Coolant Pressure Boundary," of Appendix A to 10 CFR Part 50, requires, in part, that the reactor coolant pressure boundary be designed to permit an appropriate material surveillance program for the reactor pressure vessel.

The staff reviews the materials selection, toughness requirements, and extent of materials testing conducted by the applicant in accordance with the above criteria subject to the rules and requirements of 10 CFR Part 50, Paragraph 50.55a, "Codes and Standards"; 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements"; and 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements."

(1) Compliance to Section 50.55(a), 10 CFR 50

The Edition and Addenda of the ASME Code which are applicable to the design and fabrication of any reactor vessel are specified in Section 50.55a of 10 CFR Part 50 and are based on the construction permit date. The construction permits for WNP-3 was issued on April 11, 1978. Based on the construction permit date, Section 50.55a of 10 CFR Part 50 requires that the WNP-3 reactor

vessel meet, as a minimum, the requirements of the 1971 edition of the ASME Code, Summer 1972 Addenda. The WNP-3 FSAR states that the reactor vessels were designed, fabricated, tested, inspected, and stamped according to the 1971 ASME Code, Summer 1973 addenda. Therefore, the applicant complied with the explicit requirements of Section 50.55a, 10 CFR Part 50.

(2) Compliance to Appendix G, 10 CFR 50

The staff review of this section is discussed in CESSAR SER Section 5.3.1. In that SER, the staff determined that the requirements of Appendix G, 10 CFR 50 had been met, with the exception of Sections III.B.5 and III.C.2. However, subsequent to that review, Appendix G, 10 CFR 50 was revised. The revised Appendix G, 10 CFR 50 was published on May 27, 1980 and became effective July 26, 1983. The applicant must review the material test program and pressure-temperature limits to determine whether the test program and pressure-temperature limits comply with the explicit requirements of Appendix G, 10 CFR 50. If they do not, the applicant must propose an alternative test program and pressure-temperature limits which will provide a safety margin equivalent to that required by Appendix G, 10 CFR 50.

(3) Compliance to Appendix H, 10 CFR 50

The toughness properties of the reactor vessel beltline materials must be monitored throughout the service life of WNP-3 by a material surveillance program that meets the requirements of Appendix H, 10 CFR 50. CESSAR indicates that all System 80 reactor vessel material surveillance programs will satisfy the requirements of Appendix H, 10 CFR 50. Since each reactor vessel material surveillance program is designed based on the actual reactor vessel beltline properties, the staff in its SER for CESSAR indicated that the actual reactor vessel beltline fracture toughness properties and surveillance data must be reviewed for each reference plant. The applicant has provided actual reactor vessel beltline fracture toughness properties for WNP-3, but has not reported the materials or withdrawal schedule for the WNP-3 surveillance capsules. A revised Appendix H, 10 CFR 50 was published on May 27, 1983 and became effective on July 26, 1983. As a result, the capsule withdrawal schedule requirements were revised to require that the applicant's withdrawal schedule be submitted

for approval and include its technical justification. Until the materials and withdrawal schedules for the WNP-3 surveillance capsules have been identified, we cannot complete our review of the applicant's compliance to Appendix H, 10 CFR 50.

(4) Conclusions of Compliance to Appendices G and H, 10 CFR 50

Appendix G, "Protection Against Nonductile Failure," Section III of the ASME Boiler and Pressure Vessel Code, will be used, together with the fracture toughness test results required by Appendices G and H, 10 CFR Part 50, to calculate the reactor coolant pressure boundary pressure-temperature limitations for WNP-3.

The fracture toughness tests required by the ASME Code and Appendix G of 10 CFR Part 50 will 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 boundary. The use of Appendix G, Section III of the ASME Code, as a guide in establishing safe operating procedures, and the use of the results of the fracture toughness tests performed in accordance with the ASME Code and NRC regulations, will provide adequate safety margins during operating, testing, maintenance, and anticipated transient conditions. Compliance with these Code provisions and NRC regulations constitutes an acceptable basis for satisfying the fracture toughness requirements of GDC 31.

The materials surveillance program, required by Appendix H, 10 CFR Part 50, will provide information on material properties and the effects of irradiation on material properties so that changes in fracture toughness of material in the WNP-3 reactor vessel beltline caused by exposure to neutron radiation can be properly assessed and adequate safety margins against the possibility of vessel failure can be provided.

Compliance with ASTM E-185-73 and Appendix H, 10 CFR Part 50, assures that the surveillance program constitutes an acceptable basis for monitoring radiation-induced changes in the fracture toughness of the reactor vessel material and satisfies the materials surveillance requirements of GDC 31 and GDC 32.

5.3.2 Pressure-Temperature Limits

In this section we review the applicant's pressure-temperature limits for operation of their reactor vessels. The acceptance criteria and list of references which are the basis for this evaluation are set forth in the Standard Review Plan (SRP) Section 5.3.2 of NUREG-0880 Rev. 1, dated July 1981. A discussion of this review follows.

Appendix G, "Fracture Toughness Requirements," and Appendix H, "Reactor Vessel Material Surveillance Program Requirements," 10 CFR Part 50, describe the conditions that require pressure-temperature limits for the reactor coolant pressure boundary and provide the general bases for these limits. These appendices specifically require that pressure-temperature limits must provide safety margins for the reactor coolant pressure boundary at least as great as the safety margins recommended in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G, "Protection Against Nonductile Failure." Appendix G, 10 CFR Part 50 requires additional safety margins whenever the reactor core is critical except for low-level physics tests.

The following pressure-temperature limits imposed on the reactor coolant pressure boundary during operation and tests are reviewed to ensure that they provide adequate safety margins against nonductile behavior or rapidly propagating failure of ferritic components as required by GDC 31:

(1) Preservice hydrostatic tests,
(2) Inservice leak and hydrostatic tests,
(3) Heatup and cooldown operations, and
(4) Core operation.

The applicant has not provided actual pressure-temperature limits for WNP-3, but has referenced the methodology outlined in the CESSAR FSAR, which the staff found to be acceptable (see the CESSAR SER, Section 5.3.2). However, pressure-temperature limits depend upon the fracture toughness properties of the ferritic reactor pressure vessel materials. Until the applicant submits actual WNP-3 pressure-temperature limit curves and provides the information

requested in Section 5.3.1 of this SER, the staff will not be able to complete its review of the WNP-3 pressure-temperature limits.

The pressure-temperature limits to be imposed on the reactor coolant system for all operating and testing conditions to ensure adequate safety margins against nonductile or rapidly propagating failure must be in conformance with established criteria, codes, and standards acceptable to the staff. The use of operating limits based on these criteria, as defined by applicable regulations, codes, and standards, will provide reasonable assurance that nonductile or rapidly propagating failure will not occur, and will constitute an acceptable basis for satisfying the applicable requirements of GDC 31.

5.3.3 Reactor Vessel Integrity

The staff has reviewed the FSAR sections related to the reactor vessel integrity of WNP-3. Although most areas are reviewed separately, reactor vessel integrity is of such importance that a special summary review of all factors relating to reactor vessel integrity is warranted.

The staff has reviewed the information in each area to ensure that it is complete with respect to the BOP scope and to ensure that no inconsistencies exist that would reduce the certainty of vessel integrity. In addition, the staff has reviewed the CESSAR FSAR sections related to reactor vessel integrity to ensure that it is complete with respect to the NSSS scope and to ensure that all interface requirements have been met by the applicant. A discussion of this review is contained in the CESSAR SER (NUREG-0852). The areas reviewed are:

(1) Design (Section 5.3.1)
(2) Materials of construction (Section 5.3.1)
(3) Fabrication methods (Section 5.3.1)
(4) Operating conditions (Section 5.3.2)

The acceptance criteria and references which are the basis for the evaluation are set forth in Paragraphs II.1, II.6 and II.7 (Appendices G and H, 10 CFR

Part 50) of Standard Review Plan (SRP) Section 5.3.3 in NUREG-0880 Rev. 1, dated July 1981.

Until the applicant supplies the information necessary to complete our evaluation of compliance to Appendices G and H, 10 CFR 50, and reactor vessel pressure-temperature limits, we cannot complete our evaluation of the strucutral integrity of the WNP-3 reactor vessel.

MATERIALS ENGINEERING BRANCH
MATERIALS APPLICATION SECTION

Question 251.1

Appendices G and H, 10 CFR Part 50 were revised in the Federal Register on May 27, 1983 and became effective on July 26, 1983.

a. Identify ferritic reactor coolant pressure boundary materials that do not comply with the fracture toughness requirements of Section 50.55a and Appendices G and H of 10 CFR Part 50.

b. For materials that cannot meet the fracture toughness requirements of Section 50.55a and Appendices G and H of 10 CFR Part 50, provide alternative fracture toughness data and analyses to demonstrate their equivalence to the requirements of 10 CFR Part 50.

c. To demonstrate conformance to Appendices G and H, 10 CFR Part 50;

   (1) Provide pressure temperature limit curves for hydrostatic pressure and leak tests, heat-up cooldown and core operations.

   (2) Identify the withdrawal schedule, lead factor, test samples and materials in the Reactor Vessel Materials Surveillance Program.

   (3) Provide technical justification for the capsule withdrawal schedule.

(4) Report the projected peak end of life neutron fluence (E > 1 MeV) at the inside surface and 3/4 T location of the reactor vessel.

Question 251.1

Are all weld materials, which were used in fabrication of the reactor vessel beltline, identified in FSAR Tables 5.2-4e and 5.2-4F? Is the copper chemical composition for these beltline weld materials less than the copper chemical composition for the limiting beltline plate (M-4305-6)? For all beltline weld metals, which have a copper chemical composition greater than the limiting beltline plate, identify the weld metal (heat and lot of flux and wire), its location in the beltline, its copper and nickel chemical composition and provide its Charpy V-notch test data.

5.4 Component and Subsystem Design

5.4.1 Reactor Coolant Pump Flywheel Integrity

5.4.1.5 Pump Flywheel Integrity

The safety objective of this review is to assure that the integrity of the primary reactor coolant pump flywheel is maintained to prevent failure at normal operating speeds and speeds that might be reached under accident conditions and thus preclude the generation of missiles.

The basis for review is outlined in Standard Review Plan (SRP), Section 5.4.1.1 and the Regulatory Guide 1.14, which describes and recommends a method acceptable to the NRC staff in implementing General Design Criterion 4, "Environmental and Missile Design Bases," of Appendix A of 10 CFR Part 50 with regard to minimizing the potential for failure of flywheels of the reactor coolant pump.

(1) Materials and Fabrication

Flywheels are fabricated from ASTM, A-543, Grade B, Class 2 plate. The material is produced by a process that minimizes flaws and improves fracture toughness properties. The materials as well as finished flywheels are subjected to 100

percent volumetric ultrasonic inspection using procedures and acceptance standards specified in Section III of the ASME Code.

The nil-ductility transition temperature (NDTT) of the flywheel material is obtained by two drop weight tests (DWT) which exhibit "no break" performance at -5°F in accordance with ASTM E-208. The Charpy V-notch energy level is at least 50 foot-pounds in the WR orientation at 70°F. Hence, the RT_{NDT} of 10°F can be assumed. The above drop weight tests also demonstrate that the NDTT of the material is no higher than 10°F.

(2) Design Basis

The calculated stresses at the operating speed, due to centrifugal forces and the interference fit on the shaft, are within the Regulatory Guide limits. The pump runs at 1190 rpm and may operate briefly at overspeed of 110 percent during the loss of outside load. The design speed is 125 percent of the operating speed. The flywheels are also tested at 125 percent of the maximum synchronous speed of the motor. The combined stresses at the design overspeed, due to interference fit and centrifugal forces, are within the Regulatory Guide limit.

The flywheels can be inspected by removing the cover. Hence, any crack that develops can be noticed. The critical crack length at the key-ways, where the stress concentration is high, is about 6 inches at the design overspeed.

(3) Evaluation

We have reviewed the material fabrication, design and inspection aspects of the pump flywheels for compliance with the Regulatory Guide 1.14. We have concluded that the structural integrity of the flywheels is adequate to withstand the forces imposed by overspeed transients without the loss of function, and the integrity will be verified periodically by inspection to assure that the integrity is maintained.

5.4.2 Steam Generator

5.4.2.2 Steam Generator Tube Inservice Inspection

This section was prepared with the technical assistance of DOE contractors from the Idaho National Engineering Laboratory.

5.4.2.2.1 Compliance with the Standard Review Plan

The July 1981 edition of the "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (NUREG-0800) includes Section 5.4.2.2, "Steam Generator Tube Inservice Inspection." The FSAR was reviewed in accordance with this section of the Standard Review Plan (SRP). The results of this review are summarized below.

The SRP Acceptance Criteria recommend that the Applicant perform examinations based on Regulatory Guide 1.83 and the applicable Standard Technical Specifications. The FSAR Table 1.8-3 states that compliance with Section 5.4.2.2 of NUREG-0800 is under review and a compliance statement will be provided in a subsequent amendment.

5.4.2.2.2 Evaluation of the Inspection Program

General Design Criterion 32, "Inspection of Reactor Coolant Pressure Boundary," Appendix A of 10 CFR Part 50 requires, in part, that components which are part of the reactor coolant pressure boundary be designed to permit periodic examination and testing of important areas and features to assess their structural and leak-tight integrity. The steam generators have been designed to meet the ASME Boiler and Pressure Vessel Code requirements for Class 1 and 2 components. Provisions also have been made to permit inservice inspection of the Class 1 and 2 components, including individual steam generator tubes. The design aspects that provide access for examination and the proposed inspection program must comply with the requirements of Section XI of the ASME Code with respect to the examination methods to be used, provisions for a baseline examination,

selection and sampling of tubes, inspection intervals, and actions to be taken in the event that defects are identified.

The proposed inspection program must also follow the recommendations of Regulatory Guide 1.83, Revision 1, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes," and NUREG-0212, Revision 2, "Standard Technical Specifications for Combustion Engineering Pressurized Water Reactors."

The Applicant, in Chapter 16 of the FSAR, has committed to develop the Technical Specifications using the guidance in the latest revision of NUREG-0212 as well as the approved CESSAR-F Technical Specifications. Based on the above, the staff considers the preservice examination of the steam generators an open issue subject to the Applicant docketing an inspection program that complies with the latest revision of NUREG-0212.

5.4.2.2.3 Conclusions

Conformance with Regulatory Guide 1.83, NUREG-0212, and the inspection requirements of Section XI of the ASME Code constitutes an acceptable basis for meeting, in part, the requirements of General Design Criterion 32.

5.4.2.2.4 References

1. NUREG-0800, Standard Review Plans, Section 5.2.4, "Reactor Coolant Boundary Inservice Inspection and Testing," Section 5.4.2.2, "Steam Generator Tube Inservice Inspection," July 1981.

2. Code of Federal Regulations, Volume 10, Part 50.

3. American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI, Division 1, 1980 Edition through Winter 1980 Addenda.

4. NUREG-0212, Revision 2, "Standard Technical Specifications for Combustion Engineering Pressurized Water Reactors."

5. Regulatory Guide 1.83, Revision 1, "Inservice Inspection of Pressurized Water Steam Generator Tubes."

(Editor's Note: See end of 6.6.5 for continuation of 5.4.2 discussion.)

5.4.11 Pressurizer Relief Tank (Reactor Drain Tank)

The reactor drain tank is within the scope of CESSAR. Refer to Section 5.4.11 of the CESSAR SER for this discussion.

6 ENGINEERED SAFETY FEATURES

6.1 Materials

6.1.1 Post-Accident Emergency Cooling Water Chemistry

Introduction

This review is related to providing and maintaining the proper pH of the containment sump water and recirculated containment spray water following a design basis accident to reduce the likelihood of stress corrosion cracking of austenitic stainless steel.

The applicant will use borated water with a concentration of 4000 - 4400 ppm boron from the refueling water storage tank during the initial injection phase of containment spray. The borated water will be mixed with a 40 percent by weight sodium hydroxide solution from the chemical storage tank.

The resulting solution will have a pH greater than 7, and will drain to the containment sump. Mixing is achieved as the solution is continuously recirculated from the sump to the containment spray nozzles during the recirculation phase of containment spray.

Evaluation

The post-accident emergency cooling water chemistry has been reviewed in accordance with Section 6.1.1 of Standard Review Plant (NUREG-0800, July 1981).

We evaluated the pH of the water (mixture of refueling water storage tank and sodium hydroxide solution) in the containment sump. We verified by independent calculations that sufficient sodium hydroxide is available to raise the containment sump water pH above the minimum 7.0 level to reduce the probability of stress-corrosion cracking of austenitic stainless steel components. The

removal effectiveness of the chemical additive for fission products in containment is reviewed in Section 6.5.2. We will review the surveillance requirements in the plant Technical Specifications to verify that sufficient sodium hydroxide is maintained in the containment spray additive tank.

Conclusion

On the basis of the above evaluation, we conclude that the postaccident emergency cooling water chemistry meets the minimum pH acceptance criterion of Standard Review Plan Section 6.1.1, the positions of Branch Technical Position MTEB 61, the requirements of General Design Criterion 14 of Appendix A to 10 CFR 50, and the CESSAR interface requirements, and is therefore acceptable.

6.1.2 Organic Materials

Introduction

This evaluation is conducted to verify that protective coatings applied inside containment meet the testing requirements of ANSI N101.2, "Protective Coatings (Paints) for Light Water Reactor Containment Facilities," American National Standards Institute (1972), and the quality assurance guidelines of RG 1.54 "Quality Assurance Requirements for Protective Coatings Applied to Water Cooled Nuclear Power Plants." Compliance with these requirements provides assurance that the protective coatings will not fail under design basis accident conditions and generate significant quantities of solid debris that would adversely affect the engineered safety features.

Evaluation

We have reviewed the organic materials in accordance with SRP 6.1.2 (NUREG-0800). In the FSAR, the applicant states that the coating system used on exposed surfaces inside the containment have been qualified in accordance with ANSI N101.2. The applicant also states that the protective coating system for the containment are applied in accordance with RG 1.54.

The applicant meets the positions of RG 1.54 and the testing requirements of ANSI N101.2. These measures demonstrate their suitability to withstand a postulated design-basis accident environment.

The consequences of solid debris that can potentially be formed from unqualified paints are reviewed in Section 6.2.2. The control of combustible gases that can potentially be generated from the organic materials and from qualified and unqualified paints is reviewed in Section 6.2.5.

Conclusions

On the basis of the above evaluation, we conclude that the organic materials meet the testing requirements of ANSI N101.2 and the positions of RG 1.54 and are, therefore, acceptable.

6.2 Containment Systems

The containment systems for the WNP, Unit 3 include dual containment structures, containment heat removal systems, a shield building ventilation system, a containment isolation system and a containment combustible gas control system. The primary and secondary containment structures and their associated systems all function to prevent or control the release of radioactive fission products which might be released following a postulated loss-of-coolant accident (LOCA), secondary system pipe rupture, or any other accident releasing radioactive material into the containment atmosphere.

The staff has reviewed the applicant's design, design bases, and safety analyses for the containment and the containment systems provided in the FSAR. The acceptance criteria used as the basis for our evaluation are contained in Sections 6.2.1, "Containment Functional Design," 6.2.2, "Containment Heat Removal Systems," 6.2.3, "Secondary Containment Functional Design," 6.2.4, "Containment Isolation System," 6.2.5, "Combustible Gas Control In Containment," and 6.2.6, "Containment Leakage Testing," of the Standard Review Plan (SRP), NUREG-0800, dated July 1981. These acceptance criteria include the applicable General Design Criteria (Appendix A of 10 CFR Part 50), Regulatory Guides, Branch Technical Positions, and industry codes and standards as specified in the above cited sections of the SRP. The results of the staff review are discussed below.

6.2.1 Containment Functional Design

6.2.1.1 Containment Structure

The reactor containment is a free-standing steel structure with a net free volume of 3,218,000 ft^3. The containment structure houses the nuclear steam supply system including the reactor, reactor coolant pumps, pressurizer and steam generators, as well as certain components of the plant's engineered safety feature systems. The structure is designed for an internal pressure of 44 psig and a temperature of 367°F. The reactor containment is completely enclosed by a shield building with an annulus region between the structures.

Maximum Pressure and Temperature Analysis

The applicant has analyzed the containment pressure and temperature responses for postulated reactor coolant system and secondary system pipe ruptures to establish the containment design bases and the conditions for environmental qualification of safety related equipment in the containment. The most limiting single active failure, from the standpoint of predicting the highest containment pressure and temperature, was assumed in the containment analyses.

The applicant's postulated spectrum of breaks in the reactor coolant system (i.e., loss-of-coolant accidents), as described in FSAR Table 6.2.1-1, include double ended slot breaks in the hot leg and in the cold leg at the reactor coolant pump suction and discharge. For the cold leg breaks both minimum and maximum emergency core cooling system (ECCS) flow cases were considered. The reactor coolant system pipe break spectrum is based on that prescribed in CESSAR Section 6.2.1.3, which the staff found acceptable in the CESSAR SER of November, 1981. The design basis LOCA at WNP-3 was determined to be a hot leg slot break of 19.2 ft^2; the failure of one train of the containment spray system was the worst single active failure.

The spectrum of breaks postulated for the secondary system includes slot breaks in the main steam line at five different power levels, from 0% to 102% of full power. Slot breaks were determined to be more severe than guillotine breaks. The maximum break area that allowed a pure steam blowdown was chosen as the

most limiting break for each power level. Main feedwater line breaks were not included because they result in two-phase blowdowns and thus are not as severe as main steam breaks. Again, the spectrum of main steam line breaks (MSLB) analyzed for WNP-3 is the same as that prescribed in CESSAR Section 6.2.1.4, which the staff found acceptable in the CESSAR SER of November, 1981. The MSLB resulting in the highest containment pressure was found to be the four-square-foot slot break at 0% power with a failure of one spray train. The peak containment temperature of 367°F was calculated to occur following a 8.78 ft^2 slot break in the main steam line at 102% power, with a failure of one spray train.

The applicant has performed the containment pressure and temperature analyses using the CONTEMPT-LT computer mode. Initial conditions and input data, including passive and active heat sink parameters, were conservatively chosen to produce the highest containment pressure. The highest containment pressure was calculated to be 39.4 psig (versus the containment design pressure of 44 psig), which occurred for the design basis LOCA identified above.

For the long term containment pressure response, a double ended slot break at the pump suction (with minimum safety injection) was analyzed. .The analysis showed the containment pressure would be reduced to approximately 17 psig; i.e., less than 50% of the peak calculated value (38.2 psig), in 24 hours, in accordance with staff guidelines. The design basis main steam line break was analyzed to establish the peak containment temperature to be used in developing the temperature profiles for environmentally qualifying safety-related equipment located in containment. The peak temperature was calculated to be 367°F.

We have reviewed the applicant's selection of initial conditions, input parameters, and analytical assumptions and find them to be acceptable and in conformance with staff guidelines. Staff confirmatory ahalyses were performed for the design basis reactor coolant system break and the design basis main steam line break using the CONTEMPT-LT/28 computer code. The results of the confirmatory analysis are in close agreement with the applicant's results, and confirm their acceptability.

Based on our review of the applicant's containment functional analysis, as discussed above we conclude that the applicant has satisfactorily demonstrated

the adequacy of the containment functional design, and has appropriately determined the containment temperatures and pressures to which safety-related equipment in containment must be environmentally qualified.

Protection Against Damage From External Pressure

To demonstrate the adequacy of the containment against the maximum expected external pressure, the applicant has analyzed the consequences of a postulated inadvertent actuation of the containment heat removal system during normal plant operation. The operation of two spray trains and two containment fan coolers was assumed to occur during normal operation. One of the two vacuum breakers was assumed failed. The applicant calculated a maximum pressure differential of 0.58 pounds per square inch, which is less than the containment vessel design external (differential) pressure of 0.7 psid. The initial conditions and assumptions used in the analysis were chosen to maximize the differential pressure load on the containment. Based on our reveiw of the applicant's analysis we find the containment design has sufficient margin to accommodate the maximum postulated external load.

6.2.1.2 Subcompartment Analysis

Subcompartment analyses were performed to determine the acceptability of the design differential pressure loadings on containment internal structures from high-energy line rupture accidents. The applicant's subcompartment analyses included the reactor cavity, and pressurizer and steam generator compartments, where high energy line ruptures were postulated to occur. A spectrum of pipe breaks was analyzed by the applicant to determine the limiting break that resulted in peak loads on each of the subcompartment walls.

The reactor coolant system (RCS) breaks, considered in the WNP-3 subcompartment analysis (except for one), and the mass and energy release data, were obtained from CESSAR Section 6.2.1.2. We find this approach acceptable based on the staff CESSAR SER. The one break that differs from those in CESSAR is the guillotine break of the discharge leg in the reactor coolant system; a 350-in^2 break size is specified in CESSAR, where as the break size has been reduced to 100-in^2 in the WNP-3 safety analysis report. The inherent stiffness of the system,

together with pipe whip restraints, limits the postulated pipe rupture to this break area. The 100-in² break mass and energy data were calculated based on the methodology described in CESSAR. The staff has reviewed the applicant's analysis and find the mass and energy release data acceptable, contingent upon the acceptability of the limiting pipe break size (see SER Chapter 3.0).

The applicant used in the RELAP-4 MOD6 computer program to analyze the pressure transients in the reactor cavity, and the steam generator and pressurizer compartments. Separate discussions for each subcompartment are presented below:

Reactor Cavity Analysis

The reactor activity is a heavily reinforced concrete structure that performs the dual function of providing reactor vessel support and radiation shielding. The reactor cavity is essentially a cylindrical annular air space between the reactor vessel and the primary shield wall. The cavity is bounded at the top and bottom by a neutron shield. The major vent paths for the reactor cavity are the six piping penetrations (two hot legs and four cold legs) through the primary shield wall to the steam generator compartments.

The applicant postulated 100 in², discharge and hot leg guillotine breaks; the design basis break was found to occur in the discharge leg. The peak differential pressure across the reactor cavity wall was calculated by the applicant to be 29.6 psid, versus a design value of 211.4 psid.

The applicant chose to use the reactor cavity nodalization sensitivity study of Carolina Power & Light on their Shearon Harris Plant (Docket Nos. 50-400, -401, -402, -403) as a basis for verification of the WNP-3 reactor cavity nodalization scheme. It was concluded in the Shearon Harris nodalization study that subcompartment nodalization models were determined principally by physical flow restrictions within each compartment. These flow restrictions consider the presence of steel and concrete obstructions, doorways, vent shafts, grating, reactor coolant pumps, piping, the steam generator, the pressurizer, the reactor vessel, and the reactor cavity missile and neutron shields. The subcompartment models in WNP-3 take into account all physical flow obstructions. All assumptions utilized by the applicant in the reactor cavity subcompartment analysis

have been reviewed and found to be appropriate. In addition, the staff performed a confirmatory analysis using the COMPARE-MOD 1A computer code, and the same nodal model as the applicant. Although the staff's analysis predicted a higher peak differential pressure (34.6 psid), the reactor cavity design is adequate for the differential pressure loads from the worst postulated pipe rupture within the reactor cavity.

The applicant has not provided in the FSAR an analysis of the forces and moments on the reactor vessel due to the differential pressure across the vessel caused by a reactor coolant system break within the reactor cavity. The applicant has indicated that the methodology presented in CESSAR Chapter 3 was used for the force/moment analysis. However, it is not clear that the generic analysis in the CESSAR is applicable to WNP-3. This matter will remain an open item until further justification or analysis is provided by the applicant.

<u>Steam Generator Subcompartment (SGS) Analysis</u>

The walls of the steam generator compartment are constructed of reinforced concrete. The applicant considered a spectrum of primary coolant system pipe breaks for resultant load impact on the SGC walls. A steam line break was not postulated because the routing of this line does not pass through the steam generator compartment. The staff has reviewed the spectrum of breaks postulated by the applicant and finds it acceptable. The SGC has been nodalized as 20 volumes. The staff has accepted this nodalization based on the above cited Shearon Harris subcompartment nodalization sensitivity study.

For the spectrum of breaks analyzed for the steam generator compartment, the applicant's analysis shows the design basis break is the 592-in^2 guillotine break in the suction leg. The analysis results a peak differential pressure of 13.1 psid on the steam generator subcompartment walls versus the design value of 25.9 psid. Our confirmatory calculation, using the COMPARE-MOD 1A computer code, confirms the acceptability of the applicant's results.

Based on our review of the applicant's analysis and the results of our confirmatory analysis, we find the design basis of the steam generator subcompartment walls is adequate.

Pressurizer Subcompartment Analysis

The pipe breaks considered for the pressurizer subcompartment include surge line, spray line, and safety relief valve line breaks. The most limiting case is the double-ended guillotine break in the surge line. Based on our review of these breaks, we find the applicant's choice acceptable for the pressurizer compartment analysis.

The pressurizer compartment nodalization consisted of 13 volumes. The applicant calculated a peak differential pressure of 24.3 psid across the walls of the subcompartment, compared to the design value of 84 psig. We have performed a confirmatory analysis using the COMPARE-MOD 1A computer code and our results confirm the acceptability of the applicant's results. We therefore find the applicant's pressurizer compartment analysis to be acceptable.

6.2.1.3 Mass and Energy Release Analyses for Postulated Loss-Of-Coolant Accidents

For the containment functional analysis, the applicant obtained mass and energy release data for postulated loss-of-coolant accidents from CESSAR Section 6.2.1.3. We find this approach acceptable based on the staff findings in the CESSAR SER dated November, 1981.

6.2.1.4 Mass and Energy Release Analysis for Postulated Secondary System Pipe Breaks Inside Containment

The applicant used the mass and energy release data provided in CESSAR, Section 6.2.1.4 for main steam line breaks. The CESSAR MSLB mass and energy release data may be used if certain interface requirements (for example, main steam and feedwater isolation valve closure times and maximum steam line and feedwater line volumes) are met by the applicant. We have confirmed that the WNP-3 MSLB analysis satisfies these interface requirements and, therefore, conclude that use of the CESSAR MSLB mass and energy release data is acceptable.

However, the applicant has not addressed the concerns of IE Bulletin 80-04, regarding the impact of runout flow from the feedwater system. We have requested

additional information in this regard; this matter will remain an open item until we can review the applicant's response.

6.2.1.5 Minimum Containment Pressure Analysis for Performance Capability Studies on the Emergency Core Cooling System (ECCS)

The applicant has not adequately demonstrated the applicability of the CESSAR minimum containment pressure analysis to the WNP-3 design. Therefore, a CESSAR interface requirement has not been met. The applicant has indicated that a WNP-3 plant-specific analysis will be performed. This matter will remain an open item until we have had the opportunity to review the applicant's analysis.

6.2.1.6 Summary and Conclusions

We have evaluated the WNP-3 containment functional capability with respect to the requirements of General Design criteria 16 and 50 of Appendix A to 10 CFR Part 50. We have found the applicant's analyses of the dynamic pressure loads that act on the containment vessel and subcompartment structures from postulated pipe breaks acceptable, with the following qualifications:

(1) The forces and moments acting on the reactor vessel as a result of the design basis reactor coolant system break within the reactor cavity are needed to complete our review. The guidelines of SRP 6.2.1.2 and Section 3.2 of NUREG-0609 should be followed.

(2) A minimum containment pressure analysis that is applicable to the WNP-3 plant is needed to complete our review.

(3) The applicant's response to IE Bulletin 80-04 is needed to complete our review.

6.2.2 Containment Heat Removal Systems

The function of the containment heat removal systems is to remove heat from the containment atmosphere to limit, reduce and maintain at acceptably low levels both the containment pressure and temperature following a LOCA or secondary system pipe rupture. The containment spray system (CSS) is the only active

containment heat removal system at WNP-3 and also serves as a fission product removal and control system (see SER Section 6.5).

The CSS consists of two redundant and independent 100% - capacity trains, each containing a containment spray pump, a shutdown cooling heat exchanger, a spray header, and associated valves, piping and instruments. Each of the two containment spray pumps has a design flow rate of 5000 gpm of water at a head of 645 feet during the injection mode, and a flow rate of 6000 gpm at a head of 600 feet during the recirculation mode. The CSS is automatically started by the containment spray actuation signal (CSAS) which is initiated by high-high containment pressure. The CSAS may also be initiated manually in the control room. Upon receipt of a CSAS the containment spray pumps are started, the system isolation valves are opened, and the borated water from the refueling water storage tank (RWST) flows into the containment. Full spray flow from the nozzles is established in about 50 seconds after receipt of the CSAS, assuming loss of offsite power; this satisfies the CESSAR interface requirement of establishing full spray flow in less than 58 seconds. When the water level in the RWST reaches a specified low level, a recirculation actuation signal (RAS) automatically realigns containment spray pump suction from the RWST to the safety injection system (SIS) recirculation sump. The operator must then verify that the appropriate amount of water has been discharged into the containment, the flow path from the SIS recirculation sump to the containment spray pumps is open, and the minimum flow lines are isolated. Finally, the operator will manually close the RWST isolation valves from the control room to complete the transition from the injection mode to the recirculation mode.

The CSS is designed to Quality Group B and Seismic Category I requirements. The applicant has also provided a failure mode and effects analysis (FMEA) and other information demonstrating the ability of the CSS to function following postulated single active failures.

We have reviewed the applicant's net positive suction head (NPSH) calculations and find that sufficient NPSH will be available for the CSS pumps during both the injection and the recirculation modes of operation. The applicant's evaluation of the available NPSH is consistent with the guidelines of Regulatory Guide 1.1, Rev. 0, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pump," and is acceptable.

Regulatory Guide 1.82, "Sumps for Emergency Core Cooling and Containment Spray System," provides guidelines to be met by reactor building sumps that are designed to be sources of water for the ECCS and the CSS following a LOCA. The guidelines address redundancy, location, and arrangement of sumps as well as provisions to screen out debris and ensure adequate pump performance. The applicant's sump design conforms to Regulatory Guide 1.82, revision 0, and we find the design acceptable.

Periodic testing and inspection of the system active components, i.e., pumps, valves, etc., will be performed in accordance with the in-service inspection requirements of the ASME Code Section XI to assure the operability and performance of the system.

Based on our review, we conclude that the containment spray system satisfies the requirements of General Design Criteria 38, 39, 40 and the provisions of Regulatory Guides 1.1 and 1.82, Rev. 0, and, therefore, is acceptable.

6.2.3 Secondary Containment Functional Design

The secondary containment encompasses the annual space between the concrete shield building and the steel primary containment vessel, the ECCS and mechanical penetration areas, and the fuel handling building. The shield building is a seismic Category I structure which provides biological shielding, controlled release of airborne radioactive materials following an accident, and environmental protection. The shield building ventilation system (SBVS) is an engineered safety feature designed to maintain a negative pressure in the annulus following a LOCA and to filter the airborne fission products which may leak from the primary containment to the annulus following a LOCA. (The submittals do not fully describe the fission product removal mechanisms as required in a discussion of the modeling of design basis accidents.) During normal operation, the annular subatmospheric pressure of minus 10 inches (water gage) is maintained by the annulus vacuum maintenance systems (AVMS). The regions comprising the secondary containment, other than the annulus, are designed to maintain a subatmospheric pressure by the ECCS area/fuel handling building (FHB) filtered exhaust system following a design basis accident.

The SBVS consists of two independent 100% capacity trains; each train is actuated by a separate channel of containment isolation actuation signal (CIAS) and all redundant active components are powered from separate ESF bases. Each train includes one full capacity exhaust fan, a filter train (including a demister, electric heating coil, prefilters, HEPA filter, charcoal absorbers, and after-HEPA filter), ductwork, valves, and instrumentation and controls. A failure modes and effects analysis was performed by the applicant to show the system meets the single failure criterion. All components and ductwork are designed to meet seismic Category I requirements.

The applicant has analyzed the performance of the SBVS using the WATEMPT computer code. The results of the analysis indicate a negative pressure relative to the outside atmosphere can be maintained in the annulus throughout the transient following a LOCA, thus ensuring no primary containment out-leakage escapes unfiltered directly through the shield building. However, Appendix 6.2A to the WNP-3 FSAR does not have sufficient information concerning the WATEMPT codes to permit a complete evaluation. We will conclude on the applicant's method of analysis of the post-LOCA SBVS performance after we have had an opportunity to review the additional information we have requested from the applicant.

Preoperational testing of the SBVS will verify the system performance capability to achieve and maintain a negative annulus pressure. Periodic testing and inspection of the SBVS will be included in the plant Technical Specifications.

The applicant has also identified systems for which through-line leakage following a LOCA could result in containment bypass leakage. The applicant has committed to perform local leak rate tests on the potential bypass leak paths in accordance with the requirements of Appendix J to 10 CFR Part 50. The total potential bypass leakage rate will be limited to 22 percent of the design leak rate of the containment (0.2 weight percent of the internal net free volume per day at a pressure of 39.4 psig), or 0.044 w/o per day.

With the exception of the need for additional information concerning the post-LOCA shield building annulus pressure transient analysis the staff concludes that the secondary containment systems meet the requirements of GDC 41, 42, and 43 of Appendix A, 10 CFR Part 50, and therefore, are acceptable.

6.2.4 Containment Isolation System

The function of the containment isolation system (CIS) is to allow the normal or emergency passage of fluids through the containment boundary while preserving the ability of the boundary to minimize the release of fission products that may result from a postulated accident. This section, therefore, is concerned with the isolation of fluid systems which penetrate the containment boundary, including the design and testing requirements for isolation barriers and actuators. The isolation barriers include valves, closed piping systems, and blind flanges. In general, for each penetration at least two barriers are required between the containment atmosphere or the reactor coolant system and the outside atmosphere so that failure of a single barrier does not prevent isolation.

All non-essential systems are those systems either automatically isolated by one of the actuation signals (CIAS, SIAS, MSIS) discussed below or normally locked closed. The containment isolation actuation signal (CIAS) and safety injection actuation signal (SIAS) are initiated by high containment pressure or low pressurizer pressure signals. The main steam isolation signal (MSIS) is initiated by high containment pressure low steam generator pressure, or high steam generator water level. These signals can also be initiated manually from the control room. For the containment purge and vent system, the isolation valves are also isolated by the containment high radiation signal in addition to the CIAS. We, there, conclude that the containment isolation signals provide acceptable diversity.

We have reviewed the applicant's designation of essential systems. The essential systems do not require automatic isolation, and if the automatic isolation exists, the systems are equipped with override features for remote manual operation from the control room. Those systems or portions of systems classified by the applicant as essential include the high pressure safety injection system, containment spray system, auxiliary feedwater system, main steam and feedwater isolation system, chemical volume and control system (CVCS) charging and letdown lines, CVCS reactor coolant pump (RCP) seal injection lines, RCP component cooling water lines, instrument and control air system, hydrogen purge system, and RCP seal bleed off lines. These systems are considered important to

post-accident safe shutdown and valuable in accident mitigation, particularly in the event of a small break LOCA or a secondary system rupture, and, therefore, their classification as essential is acceptable. Provisions are made to allow the operator in the control room to detect leakage from remote manually controlled systems. These provisions include instrumentation to measure radiation levels, flow rates, pressure and sump water levels in the safety equipment area and the penetration area.

We have reviewed the closure times for the containment isolation valves. Most valves close in 10 seconds or less. In particular, the containment purge and vent systems are designed to close in 5 seconds (except for isolation valve 2PV-B019 on penetration 80, which will be discussed below). We conclude that the containment isolation valve closure times, with the exception of valve 2PV-B019, are acceptable.

We have reviewed the containment purge and vent systems against the provisions of Branch Technical Position (BTP) CSB 6-4, "Containment Purging During Normal Plant Operations." The 48-inch containment purge valves will be sealed closed during normal operation and be verified to remain closed at least every 31 days; requirements for this will be included in the plant Technical Specifications. Furthermore, as a result of our study of valve leakage due to seal deterioration, leakage integrity tests of the purge and vent system isolation valves must be conducted periodically; i.e., over and above the leak testing requirements of Appendix J. This requirement, together with the test frequency will be included in the plant Technical Specifications. We conclude that the 48-inch containment purge system design satisfies the provisions of BTP CSB 6-4 and that operation of the system as proposed; i.e., only during shutdown and refueling, is acceptable.

The containment vent system, consisting of two 8-inch containment penetrations (P-80 and P-81), is designed to close following receipt of a CIAS or high radiation signal. The containment vent exhaust penetration, P-81, has two isolation valves which are designed to close in less than 5 seconds and fail closed on loss of operating power; we have found this penetration design acceptable. However, the containment vent make-up penetration (P-80) is equipped with a motor operated isolation valve (2PV-B019) and a check valve (2PV-V021). The

motor-operated valve has a closure time of 10 seconds and is designed to fail "as is" (FAI). The FAI design is not consistent with the statement in FSAR Section 6.2.4.2 that the isolation valves on both the containment vent and purge systems are designed to fail close. Also, the 10-second valve closure time differs from the 5-second assumption used in the radiological dose analysis (FSAR Section 9.4.6.6.6). The check valve (2PV-V021) in the containment vent make up line is not an appropriate type of containment isolation valve for use in a line which directly connects the containment atmosphere to the outside environment. It is the staff's position that the containment isolation valves in line P-80 should be automatic, power operated valves having less than 5 second closure times, and should fail closed on loss of operating power. This matter will remain an open item pending receipt of additional information regarding the applicant's plans for complying with the staff position.

We note that the applicant has not provided an analysis of the reduction in the containment pressure resulting from the partial loss of containment atmosphere through purge and vent system isolation valves, which may be open at the onset of a LOCA, and the consequent impact on the minimum ECCS backpressure determination. This analysis is called for in BTP CSB 6-4. Therefore, this matter will remain an open item pending receipt of an appropriate analysis from the applicant.

Our review has confirmed that the WNP-3 containment isolation system meets the explicit requirements of GDC 54, 55, 56, and 57, except as discussed below.

The isolation provisions for the chemical and volume control system (CVCS) charging line (penetration 41) and the CVCS reactor coolant pump (RCP) seal injection line (penetration 93) conform to GDC 55 except that the power-operated isolation valves outside the containment are normally open and are closed by remote-manual control from the control room. As discussed above, the CVCS charging line and the CVCS reactor coolant pump seal injection line are essential lines that have an impact on plant safety. The isolation valves will also be subject to Type C leak testing. Based on the guidance provided in SRP 6.2.4, we find the use of remote-manual instead of automatic isolation valves acceptable. We, require, however, that Class 1E emergency power be provided to valves 2CH-VQ040 and 2CH-VQ005 in these lines; because of the above this is an open issue.

The safety injection system (SIS) recirculation sump discharge lines (penetrations 23 and 24) have only one isolation valve in each line, outside containment. If isolation valves were provided inside containment, they would be submerged following an accident. Since these lines have an important safety function, system reliability is greater with only one isolation valve in each line. Also, the SIS is a closed engineered safety feature system outside containment whose integrity is appropriately maintained throughout plant life. Based on the provisions of SRP 6.2.4, we find the single isolation valve design in the SIS recirculation sump discharge lines to be acceptable. We will, however, require the applicant to discuss the adequacy of the criteria used in the design of the piping between the containment and the isolation valve, and the valve itself, and the leakage control provisions on the penetration or in the penetration area.

The reactor building vacuum relief isolation valves (penetrations 65 and 75) are normally closed and would only open in response to a high vacuum signal in the reactor building. The inboard check valves ensure that flow would always be into the containment. Based on the system design and operating requirements we find the containment isolation provisions for the reactor building vacuum relief lines acceptable.

We have reviewed information provided by the applicant to demonstrate compliance with the provisions of NUREG-0737 Item II.E.4.2, "Containment Isolation Dependability." As previously described, the applicant has complied with the provisions regarding diversity in parameters sensed for initiation of containment isolation, identification of essential and non-essential systems, automatic isolation of nonessential systems, and closure of containment purge and vent isolation valves on a high radiation signal. In addition, the FSAR states that all power-operated isolation valves will remain in their accident position after an accident signal clears until deliberate operator action is taken to reopen the valves. The containment setpoint pressure that initiates containment isolation should be reduced to the minimum value compatible with normal operating conditions. The containment setpoint pressure and the justification for it should be provided by the applicant; this information will be reviewed by the staff in conjunction with the development of the plant technical specifications. Finally, the applicant has committed to keep the 48-inch containment purge valves

closed during the operational conditions of power operation, startup, hot standby, and hot shutdown and verify that the valves are closed at least every 31 days. This is acceptable, except that the purge valves should be sealed closed (either electrically or mechanically) in accordance with SRP 6.2.4. Except for the two issues identified above, namely, justification for the containment isolation setpoint pressure and a commitment to seal close the purge valves, we conclude that the applicant has complied with the provisions of NUREG-0737, Item II.E.4.2.

The containment isolation system meets the provisions of Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," 1.29, "Seismic Design Classification," and 1.141, "Containment Isolation Provisions for Fluid Systems." The containment isolation valves are designed in accordance wtih ASME Section III Class II requirements and Quality Class I. We conclude that the WNP-3 containment isolation system meets the requirements of GDC 54, 55, and 57, and NUREG-0737 Item II.E.4.2, and conforms to SRP 6.4.2 and CSB BTP 6-4, with the exception of the six issues summarized below:

1. The applicant should upgrade the isolation provisions for the containment vent make up line (P-80); it is the NRC position that redundant power-operated, automatic isolation valves should be provided, which close in less than 5 seconds and fail closed upon loss of power to the valve operator;

2. The applicant should provide an analysis of the effect purge system operation at the time of a LOCA on the minimum containment pressure analysis for ECCS evaluation.

3. It is the NRC position that the applicant provide Class IE emergency power to valves 2CH-VQ040 and 2CH-VQ005 in the reactor coolant pump seal injection and chemical and volume control paths;

4. The applicant should confirm the design adequacy of the piping between the containment and the isolation valve, and the valve, in the SIS recirculation sump discharge lines (penetrations 23 and 24), and the affiliated leakage control provisions.

5. The applicant should justify that the containment isolation setpoint pressure is the minimum value compatible with normal operating conditions.

6. The applicant should commit to seal close the 48-inch purge valves during operating modes requiring containment integrity.

6.2.5 Combustible Gas Control System

Following a LOCA, hydrogen may accumulate within the containment as a result of: 1) hydrogen dissolved in reactor coolant system; 2) metal-water reaction between the zirconium fuel cladding and the reactor coolant; 3) corrosion of metals by emergency core coolant and containment spray solutions; and 4) radiolytic decomposition of the post-accident emergency cooling water. The applicant has provided a combustible gas control system (CGCS) to monitor and control the hydrogen concentration in containment following a LOCA. The CGCS includes the containment hydrogen analyzers, the containment hydrogen recombiners, and the containment hydrogen purge system.

The hydrogen analyzer system consists of two redundant subsystems, each of which can take samples from six locations within containment and one location in the shield building annulus. The hydrogen recombiner system consists of two stationary 100% capacity thermal (electrical) recombiners located within the containment. Both the hydrogen analyzer system and the hydrogen recombiner system are designed to Safety Class 2 and Seismic Category I standards, and are powered from Class IE power sources. The recombiner will be started manually from the control room by the operator upon indication of a hydrogen concentration of greater than 3.0 volume percent.

Each of the two Westinghouse electric hydrogen recombiners is capable of processing 100 scfm of containment atmosphere for post-accident hydrogen control. The staff has reviewed tests that were conducted for a full-scale prototype and a production recombiner. The tests consisted of proof-of-principle testing, testing on a prototype recombiner, environmental qualification testing, and functional tests for a production recombiner. (These tests are described in WCAP-7820 and its supplements.) The results of these tests demonstrate that the recombiner is capable of controlling the hydrogen in a post-LOCA containment environment.

A purge system has been provided, in addition to the hydrogen recombiner system, in accordance with Section 50.44 of 10 CFR Part 50. The purge system consists of two 100 percent capacity exhaust trains and a single makeup train.

The applicant has analyzed the production and accumulation of hydrogen within the containment from the sources discussed above. SRP 6.2.5 recommends that the analysis of hydrogen production should be based on the parameters listed in Table 1 of Regulatory Guide 1.7 for the purpose of establishing the design basis for combustible gas control systems. The applicant has been requested to confirm that their analysis is in conformance with RG 1.7. The applicant's analysis shows that one electric hydrogen recombiner actuated at a containment hydrogen concentration of 3.0 volume percent is capable of limiting the hydrogen concentration in containment to below the R.G. 1.7 lower flammability limit of 4.0 volume percent. The applicant should discuss the emergency procedures that will be in effect to guide the operator in actuating the hydrogen analyzer.

The applicant has evaluated the possibility for pocketing of hydrogen in the containment following a LOCA and concluded that pocketing of hydrogen is not very likely. This finding is based on the open, internal design of the containment, the low hydrogen generation rates from the various potential sources, and the effectiveness of hydrogen mass transport by convection. Based on our review of the applicant's rationale, we agree with the applicant's conclusion that pockets of flammable hydrogen are not likely to form.

We conclude that the CGCS satisfies the design and performance requirements of Section 50.44 of 10 CFR Part 50, "Standards for Combustible Gas Control Systems in Light Water Cooled Power Reactors," the guidelines of Regulatory Guide 1.7 and the requirements of GDC 41, 42, 43, and 50, and is acceptable, provided the applicant justifies the hydrogen production analysis, and adequate emergency procedures are in place to guide the operator in actuating the hydrogen analyzers and hydrogen recombiners.

6.2.6 Containment Leakage Testing Program

The 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 penetration and isolation valves permits periodic leakage rate testing at the pressure specified in Appendix J to 10 CFR Part 50. Included are those penetrations that have resilient seals and expansion bellows; i.e., air locks, emergency hatches, and electrical penetrations.

The 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 service lifetime and that the leakage rates will be periodically checked during service on a timely basis to maintain such leakages within the specified limits of the Technical Specifications. The plant's Technical Specifications will contain appropriate surveillance requirements for containment leak testing, including test frequencies.

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 dose will be within 10 CFR Part 100 guidelines in the event of a design basis LOCA.

We conclude that the applicant's program complies with the requirements of Appendix J and with the requirements of GDC 52, 53, and 54, and therefore, is acceptable.

<u>Outstanding Issues, Containment Systems Branch</u>

(1) The forces and moments acting on the reactor vessel following the design basis reactor coolant system break within the reactor cavity is required. This information may be either plant (WNP-3) specific or shown to be applicable to WNP-3. The guidelines of SRP 6.2.1.2 and Section 3.2 of NUREG-0609 should be followed.

(2) A minimum containment pressure analysis to support emergency core cooling system capability studies that is applicable to WNP-3 should be provided.

(3) The applicant's response to IE Bulletin 80-04, Main Steam Line Break with Continued Feedwater Addition, is required.

(4) Additional information is required concerning the applicant's shield buiding pressure response analysis. In particular, a more detailed description of the WATEMPT code is needed, including the assumptions made in the analysis, to determine the conservatism in the results.

(5) The applicant should upgrade the isolation provisions for the containment vent make up line (P-80). It is the staff's position that the containment isolation valves in line P-80 should be automatic, power operated valves having less than 5-second closure times, and should fail close upon loss of power to the valve operators.

(6) The applicant should commit to seal close the 48-inch purge valves either electrically or mechanically.

(7) The applicant should justify that the setpoint pressure for containment isolation is the minimum value compatible with normal operation conditions.

(8) The applicant should provide an analysis of the effect of purge system operation at the time of a LOCA on the minimum containment pressure analysis for the ECCS performance evaluation.

(9) It is the staff position that the applicant provide Class IE emergency power to valves 2CH-VQ040 and 2CH-VQ005 in the reactor coolant pump seal injection and chemical and volume control paths.

(10) With regard to the SIS recirculation sump discharge lines (penetration 23 and 24), the applicant should discuss the adequacy of the criteria used in the design of the piping between the containment and the isolation valve, and the valve itself, and the leakage control provisions on the penetration or serving the penetration area.

(11) The applicant should discuss the basis for the hydrogen production analysis and justify any deviations from RG 1.7.

(12) The applicant should discuss the emergency procedures that will be in effect to guide the operator in actuating the hydrogen analyzers and hydrogen recombiners.

6.2.7 Fracture Prevention of Containment Pressure Boundary

Our safety evaluation review assesses the ferritic materials in the Washington Public Power Supply System Nuclear Project No. 3 (WNP-3) containment system that constitute the containment pressure boundary to determine if the material fracture toughness is in compliance with the requirements of General Design Criterion 51, "Fracture Prevention of Containment Pressure Boundary."

GDC 51 requires that under operating, maintenance, testing and postulated accident conditions, (1) the ferritic materials of the containment pressure boundary behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized.

The WNP-3 containment system includes a ferritic steel containment vessel enclosed by a reinforced concrete shield building. The ferritic materials of the containment pressure boundary which are considered in our assessment are those which have been applied in the fabrication of the containment vessel, equipment hatch, personnel locks, penetrations and fluid system components including the valves required to isolate the system. These components are the parts of the containment system which are not backed by concrete and must sustain loads during the performance of the containment function under the conditions cited by GDC 51.

We have determined that the fracture toughness requirements contained in ASME Code editions and addenda typical of those used in the design of the WNP-3 containment may not ensure compliance with GDC 51 for all areas of the containment pressure boundary. We have elected to apply in our licensing reviews of ferritic containment pressure boundary materials the criteria for Class 2 components identified in the Summer 1977 Addenda of Section III of the ASME Code. Because the fracture toughness criteria that have been applied in construction typically differ in Code classification and Code edition and addenda, we have chosen the criteria in the Summer 1977 Addenda of Section III of the Code to provide a uniform review, consistent with the safety function of the containment pressure boundary materials. Therefore, we will review the materials of the components of the WNP-3 containment pressure boundary according to the fracture toughness requirements of the Summer 1977 Addenda of Section III for Class 2 components.

Considered in our review will be components of the containment system which are load bearing and provide a pressure boundary in the performance of the containment function under operating, maintenance, testing and postulated accident conditions as addressed in GDC 51. These components are the containment vessel, equipment hatch, personnel airlocks, penetrations and elements of specific containment penetrating systems.

In some cases, materials will not have been fracture toughness tested or will have been inappropriately tested. Generally, those materials will not have been fracture toughness tested because the ASME Code edition and addenda in effect at the time the components were ordered did not require that they be tested. Our assessment of the fracture toughness of materials that were not fracture toughness tested or were inappropriately tested is based on the metallurgical characterization of these materials and fracture toughness data presented in NUREG-0577, "Potential for Low Fracture Toughness and Lamellar Tearing on PWR Steam Generator and Reactor Coolant Pump Supports," USNRC, October 1979 and ASME Code Section III, Summer 1977 Addenda, Subsection NC.

The metallurgical characterization of these materials, with respect to their fracture toughness, will be developed from a review of how these materials were fabricated and what thermal history they experienced during fabrication. The metallurgical characterization of these materials, when correlated with the data presented in NUREG-0577 and the Summer 1977 Addenda of the ASME Code Section III, provides the technical basis for our evaluation of compliance with the Code requirements for materials that were not fracture toughness tested.

Based on our review of the available fracture toughness data and materials fabrication histories, and the use of correlations between metallurgical characteristics and material fracture toughness, we then will conclude that the ferritic components in the WNP-3 containment pressure boundary meet the fracture toughness requirements that are specified for Class 2 components by the 1977 Addenda of Section III of the ASME Code. Compliance with these Code requirements provides reasonable assurance that the WNP-3 reactor containment pressure boundary will behave in a nonbrittle manner, that the probability of rapidly propagating fracture will be minimized, and that the requirements of GDC 51 are satisfied.

6.4 Control Room Habitability

The requirements for the protection of the control room personnel under accident conditions are specified in General Design Criterion (GDC) 19. The applicant proposes to meet these requirements by incorporating shielding and emergency ventilation systems in the control room design and by having an adequate supply of self-contained breathing apparatus available in the control room for the emergency team. The applicant has stated in the FSAR that there is redundancy in the emergency systems and that the control room emergency ventilation systems are in general conformance with Regulatory Guide 1.52. This conformance is evaluated in Section 6.5.1 of this report.

The control room heating, ventilating and air conditioning (HVAC) system is designed to automatically transfer to the emergency isolation mode of operation upon receipt of a high radiation or chlorine signal from the outside air intake duct detectors or a containment isolation actuation signal. In the emergency isolation mode, the control room air is to be recirculated at the rate of 6000 cubic feet per minute (cfm) through 99% efficient ESF-grade, charcoal filtration units. In the event of a radiation release, the operator would override the isolation mode and manually initiate the emergency pressurization mode of operation. Each emergency filter train can supply a maximum of approximately 1000 cfm of filtered outside air for pressurization. The transfer to the emergency modes of operation may also be manually initiated from the control room.

The staff has evaluated the habitability of the control room in accordance with SRP Section 6.4 and Regulatory Guides 1.78 and 1.95. The applicant's estimated concentration levels of SO_2 in the control room following an accidental release are much higher than the guideline values of Regulatory Guide 1.78. The staff will, therefore, require additional analysis, including consideration of the basis for the 13% SO_2 release fraction, outside air intake rate, isolation time, concentration of SO_2 inside and outside the control room as a function of time, basis for meteorological assumptions, emergency procedures, locations of all onsite toxic gas release points relative to normal and emergency control room air intakes and other data specified in Table C.3 of Regulatory Guide 1.78. In addition, the applicant will be required to comply with Regulatory Guide 1.78 (Regulatory Position C.3) and SRP Section 6.4 (page 12) which call for

instrumentation readout and alarm in the control room and quick acting automatic isolation, or provide adequate justification for any alternatives.

The staff also determined that the following areas relating to the control room dose need to be addressed by the applicant:

(1) justification for the location of the normal and emergency air intakes relative to major radiation release points; and
(2) measures the applicant will take (emergency procedures) to assure that the control room operator will manually switch the emergency ventilation system to the pressurization mode in the event of a radiation release.

Based upon the foregoing, the applicant has not demonstrated that the control room habitability system will adequately protect the control room operators in accordance with the requirements of NUREG-0737, Item III.D.3.4, and 10 CFR Part 50, Appendix A, GDC 19.

6.5 Engineered-Safety-Feature Atmosphere Cleanup System

6.5.2 Containment Spray as a Fission Product Removal System

FSAR Question 450.1 requested additional information concerning the design and operation of the containment spray system. Because the applicant has not supplied the requested information, the review of the containment spray system cannot be completed. Therefore, the containment spray system design will be considered as an open item.

6.6 Inservice Inspection of Class 2 and 3 Components

This section was prepared with the technical assistance of DOE contractors from the Idaho National Engineering Laboratory.

6.6.1 Compliance with the SRP

The July 1981 Edition of the "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (NUREG-0800) includes Section 6.6,

"Inservice Inspection of Class 2 and 3 Components." The review is continuing because the applicant has not submitted a Preservice Inspection (PSI) Program and has not completed the PSI examinations. The FSAR Table 1.8-3 states that compliance with Section 6.6 of NUREG-0800 is under review and a compliance statement will be provided in a subsequent amendment. The staff review to date was conducted in accordance with Standard Review Plan Section 6.6 except as discussed below.

Paragraph II.3, "Acceptance Criteria, Examination Categories and Methods," will be reviewed when the completed PSI Program has been received.

Paragraph II.4, "Acceptance Criteria, Inspection Intervals," has not been reviewed because this area applies only to inservice inspection (ISI) not to PSI. This subject will be addressed during review of the ISI program after licensing.

Paragraph II.5, "Acceptance Criteria, Evaluation of Examination Results," has been reviewed. The Applicant committed in the FSAR to incorporate ASME Code Section XI, Articles IWC-3000 and IWD-3000, "Standards for Examination Evaluation," into the PSI program.

However, ongoing NRC generic activities and research projects indicate that the presently specified ASME Code procedures may not always be capable of detecting the acceptable size flaws specified in these standards. For example, ASME Code procedures specified for volumetric examination of vessels, bolts and studs, and piping have not proven to be capable of detecting acceptable size flaws in all cases. The staff will continue to evaluate the development of new or improved procedures and will require that these improved procedures be made a part of the inservice examination requirements. The Applicant's repair procedures based on ASME Code Section XI, Articles IWC-4000 and IWD-4000, "Repair Procedures," have not been reviewed. Repairs are not generally necessary in the PSI program. This subject will be addressed during review of the ISI program.

Paragraph II.7, "Acceptance Criteria, Augmented ISI to Protect Against Postulated Piping Failures," has not been completed because this subject has not yet been addressed in the applicant's PSI program. The applicant's augmented ISI program will be reviewed after it is submitted.

Paragraph II.8, "Acceptance Criteria, Code Exemptions," will be reviewed for compliance to IWC-1220 when the Applicant's PSI Program has been received. Paragraph II.9, "Acceptance Criteria, Relief Requests," has not been completed because the applicant has not identified the limitations to examination. Specific areas where ASME Code examination requirements cannot be met will be identified as the PSI progresses. The complete evaluation of the PSI program wil be presented in a supplement to the Safety Evaluation Report (SER) after the Applicant submits the required examination information and identifies all plant-specific areas where ASME Code Section XI requirements cannot be met and provides a supporting technical justification.

6.6.2 Examination Requirements

General Design Criteria 36, 39, 42, and 45, Appendix A of 10 CFR Part 50 requires, in part, that the Class 2 and 3 components be designed to permit appropriate periodic inspection of important components to ensure system integrity and capability. Section 50.55a(g) of 10 CFR Part 50 defines the detailed requirements for the preservice and inservice inspection programs.

Based upon the construction permit date of April 11, 1978, this section of the regulations requires that a preservice inspection program for Class 2 and 3 components be developed and implemented using at least the Edition and Addenda of Section XI of the ASME Code applied to the construction of the particular components. The components (including supports) may meet the requirements set forth in subsequent editions of this Code and Addenda which are incorporated by reference in 10 CFR 50.55a(b) subject to the limitations and modifications listed therein.

The initial ISI program must comply with the requirements of the latest Edition and Addenda of Section XI of the ASME Code in effect twelve months prior to the date of issuance of the operating license, subject to the limitations and modifications listed in Section 50.55a(b) of 10 CFR Part 50.

6.6.3 Evaluation of Compliance with 10 CFR 50.55a(g)

Review has been completed on the information presented in the FSAR through Amendment 3 dated April 1983. The Class 2 and Class 3 piping and components

will receive preservice examinations in accordance with the requirements of the 1977 Edition of ASME Code Section XI with Addenda through Summer 1978. The secondary side of the steam generators will be examined in accordance with the requirements of the 1974 Edition of ASME Code Section XI with Addenda through Summer 1975.

The Preservice Inspection (PSI) Program for the Class 2 and 3 components has not been received. However, the applicant has stated in the FSAR that these components will be examined per the applicable Code requirements. Based on the review of the FSAR, the staff has established technical positions that should be included in the PSI Program. The applicant has committed to identify all plant-specific areas where the Code requirements cannot be met after the examinations are performed and provide a supporting technical justification for requesting relief. The SER will be completed after the Applicant:

(1) Dockets a complete and acceptable PSI Program,

(2) Submits the requested additional information regarding the PSI/ISI program, and

(3) Submits all relief requests with a supporting technical justification.

The staff considers the review of the PSI Program an open issue subject to the Applicant providing an acceptable response to the above requirements.

The initial Inservice Inspection Program has not been submitted by the applicant. This program will be evaluated after the applicable ASME Code Edition and Addenda can be determined based on Section 50.55a(b) of 10 CFR Part 50, but before inservice inspection commences during the first refueling outage.

6.6.4 Conclusions

Compliance with the preservice and inservice inspections required by the ASME Code and 10 CFR Part 50 constitutes an acceptable basis for satisfying applicable requirements of General Design Criteria 36, 39, 42, and 45.

6.6.5 References

1. NUREG-0800, Standard Review Plan, Section 6.6, "Inservice Inspection of Class 2 and 3 Components," July 1981.

2. Code of Federal Regulations, Volume 10, Part 50.

3. American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI, Division 1.

   1974 Edition through Summer 1975 Addenda
   1977 Edition through Summer 1978 Addenda

<u>Review of the FSAR and Technical Positions Regarding the Preservice(PSI)/Inservice(ISI) Inspection Programs</u>

Question 250.1

For completion of SER Sections 5.2.4 and 6.6, the staff requires that the PSI Program Plan be submitted for review. The PSI Program should include reference to the ASME Code Section XI Edition and Addenda that will be used for the selection of components for examinations, lists of the components subject to examination, a description of the components exempt from examination by the applicable Code, and the examination isometric drawings.

Paragraph 50.55a(b)(2)(iv) requires that ASME Code Class 2 piping welds in the Residual Heat Removal (RHR) Systems, Emergency Core Cooling (ECCS) Systems, and Containment Heat Removal (CHR) Systems shall be examined. These systems should not be completely exempted from preservice volumetric examination based on Section XI exclusion criteria contained in IWC-1220. To satisfy the inspection requirements of General Design Criteria 36, 39, 42, and 45, the Preservice Inspection Program must include volumetric examination of a representative sample of welds in the RHR, ECCS and CHR Systems.

Question 250.2

Plans for preservice examination of the reactor pressure vessel welds should address the degree of compliance with Regulatory Guide 1.150. In FSAR Section 1.8, Table 1.8-1, the Applicant indicates exceptions to Regulatory Guide 1.150. List the exceptions being taken and discuss the degree of compliance and the qualification of procedures to be used to assure finding service-induced flaws on the inside surface.

Question 250.3

Describe the measures taken to ensure that austenitic stainless steel piping welds are examined using effective techniques and the methods of assuring adequate examination sensitivity over the required examination volume. Discuss the preservice examination criteria used to record, report, and plot geometric or metallurgical ultrasonic indications in the piping systems to assure correlation of baseline data with inservice inspection results.

The ASME Code, Section XI, 1977 Edition with Addenda through Summer 1978 and 1980 Edition specifies the use of Appendix III of Section XI for ferritic piping welds. If this requirement is not applicable (for example, for austenitic piping welds), ultrasonic examination is required by Section XI to be conducted in accordance with the applicable requirements of Article 5 of Section V, as amended by IWA-2232. A technical justification is required if any alternatives are used. If Section XI, Appendix III, Supplement 7, will be used for the examination of austenitic piping welds, discuss the following:

(1) All modifications permitted by Supplement 7

(2) Methods of qualifying the procedure for examination through the weld (if complete examination is to be considered for examination conducted with only one side access).

When using either Article 5 of Section V or Appendix III of Section XI for examination of either ferritic or austenitic piping welds, the following should be incorporated.

(1) Any crack-like indication, regardless of ultrasonic amplitude, discovered during examination of piping welds or adjacent base metal materials should be recorded and investigated by a Level II or Level III examiner to the extent necessary to determine the shape, identity, and location of the reflector.

(2) The Owner should evaluate and take corrective action for the disposition of any indication investigated and found to be other than geometrical or metallurgical in nature.

Question 250.4

All preservice examination requirements defined in Section XI of the ASME Code that have been determined to be impractical must be identified and a supporting technical justification for requests for relief must be provided. The relief request submittal should include at least the following information:

(1) For ASME Code Class 1 and 2 components, provide a table similar to IWB-2500 and IWC-2500 confirming that either the Section XI preservice examination was performed on the component or relief is requested.

(2) Where relief is requested for pressure retaining welds in the reactor vessel, identify the specific welds that did not receive a 100% preservice ultrasonic examination, and indicate the extent of the examination that was performed.

(3) Where relief is requested for piping system welds (Examination Category B-J, C-F, and C-G), provide a list of the specific welds that did not receive a complete Section XI preservice examination including drawing or isometric identification number, system, weld number, and physical configuration (e.g., pipe-to-nozzle weld, etc.). Indicate the extent of the preservice examination that was performed. When the volumetric examination was performed from one side of the weld, discuss whether the entire weld volume and the heat affected zone (HAZ) and base metal on the far side of the weld were examined. State the primary reason that a specific examination is impractical (e.g., support of component restricts access,

fitting prevents adequate ultrasonic coupling on one side, component-to-component welds prevent ultrasonic examination, etc.). Indicate any alternative or supplemental examinations performed and methods of fabrication examination.

the guidelines of RG 1.29, Position C.1 and RG 1.27, Positions C.1, C.2, and C.3 and BTP ASB 9-2 with respect to design capability, seismic classification, and capability to remove sufficient decay heat to maintain plant safety and is, therefore, acceptable. The UHS meets the acceptance criteria of SRP Section 9.2.5.

9.3 Process Auxiliaries

9.3.1 Compressed Air System

9.3.2 Process and Post-Accident Sampling System

A. Process Sampling System

Introduction

The process sampling system is designed to provide representative liquid and gaseous samples drawn from the primary and secondary coolant systems, the associated auxiliary system process streams, and the spent fuel pool cleanup system. Provisions are made to assure that representative samples are obtained from well mixed streams or volumes of effluent by the selection of proper sampling procedures. In the event of an accident, all sample lines which pass through the containment are automatically isolated by two fail-closed solenoid operated valves.

Evaluation

The information provided by the applicant has been reviewed in accordance with Section 9.3.2 of the Standard Review Plan (NUREG-0800, July 1981)

The process sampling system includes piping and other components associated with the system from the point of sample withdrawal from a fluid system up to the analyzing station, sampling station, or local sampling point. Our review included the provisions proposed to sample all principal fluid process streams associated with plant operation and the applicant's proposed design of these

systems, including the location of sampling points, as shown on piping and instrumentation diagrams.

We determined that the proposed process sampling system meets (1) the requirements of General Design Criterion 13 in Appendix A to 10 CFR Part 50 to monitor variables that can affect the fission process for normal operation, anticipated operational occurrences, and accident conditions, by sampling the reactor coolant, the ECCS core flooding tank, the refueling water storage tank, the boric acid mix tank, and the boron injection tank for boron concentration; (2) the requirements of General Design Criterion 14 in Appendix A to 10 CFR Part 50, to assure a low probability of abnormal leakage, rapidly propagating failure, and gross rupture, by sampling the reactor coolant and the secondary coolant for chemical impurities that can affect the reactor coolant pressure boundary material integrity; (3) the requirements of General Design Criterion 26 in Appendix A to 10 CFR Part 50 to control the rate of reactivity changes, by sampling the reactor coolant, the refueling water storage tank, and the boric acid mix tank for boron concentration; and (4) the requirements of General Design Criteria 63 and 64 in Appendix A to 10 CFR Part 50 to monitor for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents, by sampling the reactor coolant, the pressurizer tank, the steam generator blowdown, the secondary coolant condensate treatment waste, the sump inside containment, the containment atmosphere, the spent fuel pool, the gaseous radwaste storage tank for radioactivity, and the CESSAR interface requirements discussed in the CESSAR SER.

We further determined that the proposed process sampling system meets (a) the standards of ANSI N13.11969 for obtaining airborne radioactive samples; (b) the requirements of 10 CFR Part 20, 20.1(c) and regulatory positions 2.d(2), 2.f(3), 2.f(8), and 2.i(6) of Regulatory Guide 8.8, Revision 3, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable," to maintain radiation exposures to as low as is reasonably achievable, by providing (1) ventilation systems and gaseous radwaste treatment system to contain airborne radioactive materials; (2) liquid radwaste treatment system to contain radioactive material in fluids; (3) spent fuel Pool cleanup system to remove radioactive contaminants in the

spent fuel pool water; and (4) remotely operated containment isolation valves to limit reactor coolant loss in the event of rupture of a sampling line; (c) the requirements of General Design Criterion 50 in Appendix A to 10 CFR Part 50 to control the release of radioactive materials to the environment by providing isolation valves that will fail in the closed position; and (d) regulatory positions C.1, C.2, and C.3 of Regulatory Guide 1.26, Revision 3, "Quality Group Classifications and Standards for Water-Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants," and C.1, C.2, C.3, and C.4 of Regulatory Guide 1.29, Revision 3, "Seismic Design Classification," by designing the sampling lines and components of the process sampling system to conform to the classification of the system to which each sampling line and component is connected, and thus meets the quality standards requirements of General Design Criterion 1 and the seismic requirements of General Design Criterion 2.

Conclusion

On the basis of the above evaluation, we conclude that the proposed process sampling system meets the relevant requirements of 10 CFR Part 20, § 20.1(c), General Design Criteria 1, 2, 13, 14, 26, 60, 63, and 64 in Appendix A to 10 CFR Part 50, and the appropriate sections in Regulatory Guides 8.8, 1.26 and 1.29 and therefore is acceptable.

B. Post-Accident Sampling System (NUREG-0737, II.B.3)

Introduction

Subsequent to the TMI-2 incident, the need was recognized for an improved post-accident sampling system (PASS) to determine the extent of core degradation following a severe reactor accident. Criteria for an acceptable sampling and analysis system are specified in NUREG-0737, Item II.B.3. The system should have the capability to obtain and quantitatively analyze reactor coolant and containment atmosphere samples without radiation exposure to any individual exceeding 5 rem to the whole body or 75 rem to the extremities (GDC-19) during and following an accident in which there is core degradation. Materials to be analyzed and quantified include certain radionuclides that are indicators of

severity of core damage (e.g., noble gases, isotopes of iodine and cesium, and nonvolatile isotopes), hydrogen in the containment atmosphere and total dissolved gases or hydrogen, boron, and chloride in reactor coolant samples.

To comply with NUREG-0737, Item II.B.3, the applicant should (1) review and modify his sampling, chemical analysis, and radionuclide determination capabilities as necessary and (2) provide the staff with information pertaining to system design, analytical capabilities and procedures in sufficient detail to demonstrate that the criteria are met.

Evaluation

By letter dated April 1, 1983, the applicant provided information on the PASS.

Criterion (1):

> The applicant shall have the capability to promptly obtain reactor coolant samples and containment atmosphere samples. The combined time allotted for sampling and analysis should be three hours or less from the time a decision is made to take a sample.

The applicant has designated that the static uninterruptable power supply be available for the post accident sampling system in the event of a loss of offsite power. Information was not provided on the capability to obtain and analyze reactor coolant and containment atmosphere samples within three hours from the time a decision is made to take a sample. We find the applicant partially meets Criterion (1).

Criterion (2):

> The applicant shall establish an onsite radiological and chemical analysis capability to provide, within the three hour time frame established above, quantification of the following:

(a) Certain radionuclides in the reactor coolant and containment atmosphere that may be indicators of the degree of core damage (e.g., noble gases, iodines and cesiums and non-volatile isotopes);

(b) hydrogen levels in the containment atmosphere;

(c) dissolved gases (e.g., H_2), chloride (time allotted for analysis subject to discussion below), and boron concentration of liquids;

(d) alternatively, have inline monitoring capabilities to perform all or part of the above analyses.

The PASS provides diluted and undiluted liquid and gaseous samples for grab sample analysis, and inline monitoring of hydrogen in the containment atmosphere.

We find that these provisions partially meet Criterion (2). The applicant should provide a procedure to estimate the extent of core damage based on radionuclide concentrations and taking into consideration other physical parameters such as core temperature data and sample location. Also, information on an onsite radiological and chemical analysis capability must be provided.

Criterion (3):

Reactor coolant and containment atmosphere sampling during post-accident conditions shall not require an isolated auxiliary system (e.g., the letdown system, reactor water cleanup system) to be placed in operation in order to use the sampling system.

Reactor coolant and containment atmosphere sampling during postaccident conditions does not require an isolated auxiliary system to be placed in operation in order to perform the sampling function. The applicant's proposal to meet Criterion (3) is acceptable since PASS sampling is performed without requiring

operation of an isolated auxiliary system and all PASS valves which are not accessible after an accident are environmentally qualified.

Criterion (4):

> Pressurized reactor coolant samples are not required if the applicant can quantify the amount of dissolved gases with unpressurized reactor coolant samples. The measurement of either total dissolved gases or H_2 gas in reactor coolant samples is considered adequate. Measuring the O_2 concentration is recommended but is not mandatory.

Pressurized reactor coolant samples are cooled and degassed to obtain representative dissolved hydrogen and oxygen samples at the PASS sampling station. We have determined that these provisions meet Criterion (4) of Item II.B.3 in NUREG-0737 and are, therefore. acceptable.

Criterion (5):

> The time for a chloride analysis to be performed is dependent upon two factors: (a) if the plant's coolant water is seawater or brackish water, and (b) if there is only a single barrier between primary containment systems and the cooling water. Under both of the above conditions the applicant shall provide for a chloride analysis within 24 hours of the sample being taken. For all other cases, the applicant shall provide for the analysis to be completed within 4 days. The chloride analysis does not have to be done onsite

The applicant proposes that chloride analysis will be done at an offsite laboratory to meet the four day requirement. However, to comply with Criterion (5), specific arrangements need to be made with the offsite laboratory and a licensed shipping container needs to be available for transporting the sample.

Criterion (6):

> The design basis for plant equipment for reactor coolant and containment atmosphere sampling and analysis must assume that it is possible to obtain and analyze a sample without radiation exposures to any individual exceeding the criteria of GDC-19 (Appendix A, 10 CFR Part 50) (i.e., 5 rem whole body, 75 rem extremities). (Note that the design and operational review criterion was changed from the operational limits of 10 CFR Part 20 (NUREG-0578) to the GDC-19 criterion (October 30, 1979 letter from H. R. Denton to all licensees.))

The applicant has performed a shielding analysis on operator exposure while obtaining and transporting a PASS sample. This evaluation does not meet Criterion (16) which requires a manrem exposure estimate based on person motion study for sampling, transport, and analysis of all required parameters.

Criterion (7):

> The analysis of primary coolant samples for boron is required for PWRs. (Note that Rev. 2 of Regulatory Guide 1.97 specifies the need for primary coolant boron analysis capability at BWR plants.)

The applicant will have the capability to analyze reactor coolant samples for boron. Boron accuracy is discussed in Criterion (10). This provision meets the recommendations of Regulatory Guide 1.97, Rev. 2 and Criterion (7) and is, therefore, acceptable.

Criterion (8):

> If inline monitoring is used for any sampling and analytical capability specified herein, the applicant shall provide backup sampling through grab samples, and shall demonstrate the capability of analyzing the samples. Established planning for

analysis at offsite facilities is acceptable. Equipment provided for backup sampling shall be capable of providing at least one sample per day for 7 days following onset of the accident and at least one sample per week until the accident condition no longer exists.

Reactor coolant dissolved oxygen and containment hydrogen concentration measurements will be performed inline. The PASS has the capability of providing backup samples of diluted gas, diluted liquid and undiluted liquid. We find these provisions meet Criterion (8) and are, therefore, acceptable.

Criterion (9):

The applicant's radiological and chemical sample analysis capability shall include provisions to:

(a) Identify and quantify the isotopes of the nuclide categories discussed above to levels corresponding to the source term given in Regulatory Guides 1.3 or 1.4 and 1.7. Where necessary and practicable, the ability to dilute samples to provide capability for measurement and reduction of Personnel exposure should be provided. Sensitivity of onsite liquid sample analysis capability should be such as to permit measurement of nuclide concentration in the range from approximately 1 μCi/g to 10 Ci/g.

(b) Restrict background levels of radiation in the radiological and chemical analysis facility from sources such that the sample analysis will provide results with an acceptably small error (approximately a factor of 2). This can be accomplished through the use of sufficient shielding around samples and outside sources, and by the use of a ventilation system design which will control the presence of airborne radioactivity.

The radionuclides in both the primary coolant and the containment atmosphere will be identified and quantified. Provisions are available for diluted reactor

coolant samples to minimize personnel exposure. The PASS can perform radioisotope analyses at the levels corresponding to the source term given in Regulatory Guides 1.4, Rev. 2, and 1.7. Radiation background levels will be restricted by shielding. We find these provisions partially meet Criterion (9). The applicant should determine whether radiochemical analysis results can be obtained within an acceptably small error (approximately a factor of 2). Also, information on the ventilation system design provisions to control airborne radioactivity should be provided.

Criterion (10):

> Accuracy, ranage, and sensitivity shall be adequate to provide pertinent data to the operator in order to describe radiological and chemical status of the reactor coolant systems.

The applicant has not provided sufficient information for our review to determine compliance with the requirements of Criterion (10).

Criterion (11):

> In the design of the postaccident sampling and analysis capability, consideration should be given to the following items:
>
> (a) Provisions for purging sample lines, for reducing plateout in sample line, for minimizing sample loss or distortion, for preventing blockage of sample lines by loose material in the RCS or containment, for appropriate disposal of the samples, and for flow restrictions to limit reactor coolant loss from a rupture of the sample line. The postaccident reactor coolant and containment atmosphere samples should be representative of the reactor coolant in the core area and the containment atmosphere following a transient or accident. The sample lines should be as short as possible to minimize the volume of fluid to be taken from containment. The residues of sample collection should be returned to containment or to a closed system.

(b) The ventilation exhaust from the sampling station should be filtered with charcoal adsorbers and high-efficiency particulate air (HEPA) filters.

The applicant has addressed provisions for purging to ensure samples are representative, size of sample line to limit reactor coolant loss from a rupture of the sample line. and ventilation exhaust from PASS filtered through charcoal adsorbers and HEPA filters. To limit iodine plateout, the containment air sample line is heat traced. We determined that these provisions meet Criterion (11) of Item II.B.3 of NUREG-0737, and are, therefore, acceptable.

<u>Conclusion</u>

On the basis of the above evaluation, we conclude that the post-accident sampling system meets five of the eleven criteria of NUREG-0737, Item II.B.3. Additional information is needed to complete our review of the six remaining criteria:

Criterion (1):

Provide information on the capability to promptly obtain and analyze reactor coolant and containment atmosphere samples within three hours from the time a decision is made to take a sample.

Criterion (2):

Provide information on onsite radiological and chemical analysis capability. Provide a procedure to estimate the extent of core damage.

Criterion (5):

Make specific arrangements for chloride analysis at an offsite laboratory and also for a licensed shipping container.

Criterion (6)

Perform a personmotion study for man-rem exposures accumulated during sampling, transport, and analysis of all required parameters.

Criterion (9):

Discuss whether radiochemical analysis results can be obtained within an acceptably small error (approximately a factor of 2). Discuss the ventilation system design relative to airborne radioactivity control.

Criterion (10):

Discuss the accuracy, range, and sensitivity of the PASS parameters to ensure an adequate description of the radiological and chemical status of the reactor coolant system.

9.3.3 Equipment and Floor Drainage System

9.3.4 Chemical and Volume Control System

Our evaluation of the Chemical and Volume Control System (CVCS) proposed for use in WNP-3 is presented in Section 9.3.4 of the CESSAR SER. In that evaluation, we conclude that the CESSAR CVCS is acceptable, provided the CVCS interface requirements for balance of plant (BOP) are adequate.

CESSAR identifies interface requirements for the CVCS with the BOP in Section 9.3.4, which include normal and emergency power; protection from natural phenomena such as floods, winds, tornadoes, and earthquakes; protection from pipe failure and missiles; separation of components; thermal limitations; inspection and testing; materials compatibility; system/component arrangements; radwaste management; overpressure protection; refueling water tank design parameters; alternate source of borated water from the spent fuel pool; fire protection; operating temperature ranges; environmental control; and mechanical interaction between components.

9.5 Other Auxiliary Systems

9.5.1 Fire Protection Program

We have reviewed the fire protection program for conformance with SRP Section 9.5.1, Fire Protection. The SRP contains, in BTP CMEB 9.5.1, the technical requirements of Appendix A to BTP 9.5.1 and Appendix R to 10 CFR 50.

The applicant's Fire Hazards Analysis, transmitted by letter dated October 22, 1982 and amended in Revision 2 of the FSAR, was in response to the staff's request that he evaluate his fire protection program against the guidelines of Appendix R to BTP ASB 9.5.1, "Guidelines for Fire Protection for Nuclear Power Plants."

The applicant has provided an evaluation of the plant fire protection program against the guidelines of BTP CMED 9.5.1 (NUREG-0800, July 1981 which includes Appendix R to 10 CFR 50. We will require that such an evaluation be performed.

We have reviewed the automatic and manually operated water fire suppression systems, the fire detection systems, fire barriers, fire doors and dampers, fire protection administrative controls, and the fire brigade size and training. The objective of this review is to ensure that in the event of a fire, personnel and plant equipment would be adequate to safely shutdown the reactor, to maintain the plant in a safe shutdown condition, and to minimize the release of radioactive material in the environment. As part of its reviews, we will visit the plant site to examine the relationship of safety-related components, systems, and structures in specific plant areas to both combustible materials and to associated fire detection and suppression systems.

Our consultant, Gage Babcock & Associates, Inc., participated in the review of the fire protection program.

9.5.1.2 Fire Protection Program Requirements

Fire Protection Program

The fire protection program establishes policy for the protection of structures, systems and components important to safety. The applicant has not committed to conform to the technical requirements for fire protection programs in BTP CMEB 9.5.1, Section C.1. We will require that the applicant commit to conform to the above guidelines.

Fire Hazards Analysis

The applicant provided a fire hazard analysis with the FSAR. The analyses identified the fire areas of the plant, and for each fire area specified the combustible materials present, identified safety related systems, determined the consequences of a fire on safe shutdown capability, and summarized available fire protection. Our evaluation of the identified fire hazards is contained in the balance of this report.

GDC 3, Appendix A to 10 CFR Part 50 requires that "Fire fighting systems shall be designed to assure that rupture or inadvertent operation does not significantly impair the safety capability of those structures, systems and components."

To satisfy this requirement the applicant has designed components required for hot shutdown so that rupture or inadvertent operation of fire suppression systems will not adversely affect the operability of these components. Where necessary, appropriate protection is provided to prevent impingement of water spray on components required for hot shutdown. Redundant trains of components that are susceptible to damage from water spray are physically separated so that manual fire suppression activities will not adversely affect the operability of components not involved in the postulated fire. However, we are concerned that the mechanism by which fire fighting hose streams systems may cause the simultaneous failure of redundant or diverse trains have not been considered in the design. We will require that the applicant identify such mechanisms that were considered in his fire hazards analysis and the measures taken to preclude

the fire or fire suppressant induced failure of redundant or diverse safety trains.

Implementation of Fire Protection Program

The licensee has not indicated when the fire protection program will be operational. We will require that the fire protection program be operational at or before initial fuel loading.

9 5.1.3 Administrative Controls

The administrative controls for fire protection consist of the fire protection organization, the controls over combustibles and ignition sources, and the pre-fire plans and procedures for fighting fires. In the fire hazards analysis, the applicant did not provide information for us to independently verify compliance with our guidelines. We will require that the applicant provide details showing compliance with the guidelines in BTP CMEB 9.51, Item C.2 regarding administrative controls.

Fire Brigade and Fire Brigade Training

The applicant, has not provided a description of the plant fire brigade, including equipment, to verify the guidelines contained in BTP CMEB 9.51, Item C.3. We will require that the applicant provide details showing compliance with the guidelines in BTP CMEB 9.51, Item C.3 in the establishment and training of the fire brigade.

9.5.1.4 General Plant Guidelines

Building Design

Fire areas are defined by walls and floor/ceiling assemblies that have a fire resistance rating of 3 hours. Some walls and floor/ceilings are not fire rated. They are delineated in Amendment 2 of the applicant's fire hazard analysis. We are concerned that all fire barriers which separate redundant shutdown-related

divisions are not 3-hour fire rated as determined by the test method identified in ASTM E119. We are also concerned that such barriers are not complete, in that they may contain unprotected openings which may act as an avenue for vertical and horizontal fire spread. We will require that all walls and floor/ceiling assemblies which define fire areas and which separate redundant shutdown systems be tested by an independent fire authority and meet the acceptance criteria of ASTM E119 in accordance with Section C.5.a(1) of BTP CMEB 9.51.

Piping, conduit, cable tray, and bus duct penetrations of fire rated barriers are provided with penetration seals that have been successfully tested in accordance with ASTM E119, by an independent laboratory in the configurations which are typical of what is found throughout the plant. Openings inside conduit are provided with penetration seals either at the barrier or at both ends of the conduit on either side of the barrier. This configuration conforms with Section C.5.a of BTP CMEB 9.51 and is, therefore, acceptable.

Door openings in fire rated barriers are, for the most part, equipped with labeled fire doors. In the fire hazards analysis, the applicant indicated that certain multifunction doors, such as bullet proof, watertight and pressure resistive doors are not fire rated but are of "certified fire resistive construction". We are concerned that these doors may not be able to withstand anticipated fire exposures in the plant and maintain their integrity. We will require that the applicant provide compensatory fire protection for these doors to meet Section C.5.a(5) of BTP CMEB 9.51.

Ventilation ducts that penetrate all fire barriers are provided with fire dampers commensurate with the fire rating of the barrier. The fire dampers are U.L. labeled and installed according to the manufacturer's directions and NFPA Standards Nos. 80 and 90A.

We conclude that the fire dampers will be provided in accordance with guidelines of BTP CMEB 9.51, Section C.5.a(5) and are, therefore, acceptable.

Walls and structural materials throughout the plant are noncombustible. Materials that are used as interior finish, including thermal insulation, radiation

shielding and sound proofing have all been tested in accordance with the method in ASTM E84 in the configuration in which they are found in the plant. They have been demonstrated to exhibit flame spread, smoke and fuel contribution ratings of 50 or less by an independent laboratory. We find this acceptable.

All oil filled transformers are located outside in the transformer yard. They are separated from safety-related structures by more than 75 feet. The transformers are installed over gravel filled pits, designed to collect all oil from a potential spill. We find this acceptable. The applicant has not provided information in the fire hazards analysis regarding transformers inside fire areas containing safety-related systems. We will require that such transformers be of the dry type or insulated and cooled with noncombustible liquid in accordance with Section C.5.a(12) of BTP CMEB 9.51.

Safe Shutdown Capability

The information provided by the applicant is insufficient to verify compliance with our guidelines. We will require the applicant to provide a safe shutdown analysis in accordance with the guidelines of Section C.5.b of BTP CMEB 9 51.

Alternate Shutdown Capability

The applicant has not completed a fire hazards analysis of the consequences of a fire in areas, such as the control room, which contain redundant shutdown systems and which require the provision of an alternate shutdown capability. We will require that an alternate shutdown capability comply with the guidelines contained in Section C.5.c of BTP CMEB 9.51.

Control of Combustibles

The applicant has stated in the fire hazard analysis that safety-related systems have been isolated or separated from combustible materials to the extent possible, that the use of plastic materials has been minimized, and that storage of flammable materials complies with the requirements of NFPA 30. The staff finds this acceptable.

The applicant has not provided information in the fire hazards analysis, regarding the routing of hydrogen lines, to enable us to verify compliance with our guidelines. We will require that hydrogen lines in safety related areas be either designed to seismic Class I requirements, or sleeved and vented to the outside, or equipped with excess flow valves in accordance with Section C.5.d(5) of BTP CMEB 9.51.

Electrical Cable Construction, Cable Trays, and Penetrations

Cable trays are constructed of galvanized steel. Fire breaks, consisting of fire resistive material are provided in horizontal and vertical cable trays to prevent the propagation of fire. We find this acceptable.

The power, control and instrumentation cables used in the plant has been tested and has passed the acceptance criteria of IEEE-38374. We find this acceptable.

Safety related cable trays outside the cable spreading room are not provided with continuous line-type heat detectors. The applicant has not justified this deviation.

In addition, the applicant has not provided sufficient information for us to verify compliance with Section C.5.e.(2) of BTP CMEB 9.51. We are concerned that safety related cable trays which are not accessible for manual fire fighting may be prone to fire damage. We will require that safety related cable trays which are not accessible be protected in accordance with Section C.5.e of BTP CMEB 9.51.

Ventilation

There are no ventilation systems in the plant designed specifically to exhaust smoke or other products of combustion. Normal plant ventilation systems will be utilized for that purpose. Portable smoke ejectors are provided to facilitate the removal of the products of combustion should normal ventilation systems be unavailable because of damper closures or other failures. Because the normal ventilation system is capable of being realigned so as to shut down the supply

air system which maintaining the return air system in an exhaust mode, we find this acceptable. The power supply and controls for the ventilation systems are located outside the fire areas served by the systems.

Stairwells are designed to reduce smoke infiltration during a fire. Charcoal filters have been protected in accordance with Regulatory Guide 1.52. This includes the provision of manual fire suppression system for nonsafety filters. We find this acceptable.

Lighting and Communication

Redundant AC emergency lighting is provided in areas where safety-related functions are performed, in access routes to these areas, and for emergency evacuation. In addition, emergency DC lighting from the 125-volt station batteries provided emergency lighting for the control room and remote shutdown rooms. We are concerned that under this arrangement, a fire in one area may result in damaged circuits for emergency lighting in other fire areas. We will require that fixed, self-contained emergency lighting units, with individual, 8-hour capacity power supplies, be installed in accordance with Section C.5.g of BTP CMEB 9.51.

Emergency plant communications is provided via voice powered head sets located at preselected stations. However, the applicant has not provided sufficient information for us to verify compliance with our guidelines. We will require that a communications capability, independent of normal plant systems, be provided for use by the fire brigade and other operations personnel as delineated in Section C.5.g of BTP CMEB 9.51.

Fire Detection and Suppression

Fire Detection

A fire detection system is provided for all areas containing safety and safe shutdown-related systems and in areas which present a potential fire exposure to such systems. The detection systems and all signaling circuits, such as

water flow alarms, are Class A supervised as defined in NFPA Standard 72D. The systems provide alarm and trouble indication to the control room even under single break or ground fault conditions.

All fire detection devices and associated equipment are either UL listed or FM approved. The fire detection systems are designed and installed in accordance with NFPA Standards Nos. 72A and 72D. We find this acceptable.

The applicant has not provided sufficient information for us to verify that the primary and secondary power supplies for fire detection and signaling systems complies with our guidelines. We will require that such systems comply with Section C.6 a(6) of BTP CMEB 9.51.

Fire Protection Water Supply System

The water supply system for fire fighting consists of two fire pumps: one is electric motor driven and the other is diesel engine driven. We are concerned that the fire pumps and their controllers are not listed by an independent testing laboratory for the intended use. We will require that the fire pumps and controllers be UL listed or FM approved in accordance with Section C.6.b. of BTP CMEB 9.51.

The two fire pumps are located in separate cubicles of the fire pumphouse and are completely separated by 3-hour fire rated walls. Each pump has a separate suction, and discharges through independent underground connections into the main fire water piping distribution system. Each fire pump has a rated capacity of 3500 gpm at 125 psi.

Fire protection water for the plant is taken from wells at the makeup pumphouse and pumped by the makeup booster pump into two 300,000-gallon fire water storage tanks which are reserved exclusively for fire fighting. The fire pumps can draw suction from either or both tanks. The interconnecting piping is arranged such that a leak in one tank will not cause drainage of both the water storage tanks. We find this acceptable.

The greatest water demand for the fixed fire suppression systems is 2000 gpm, and coupled with 1000 gpm for hose streams, creates a total water demand of 3000 gpm. We find that the water supply system can deliver the required water demand with one fire pump out of service.

An 18,000-gallon auxiliary fire water storage tank is provided as a water supply for standpipe and hose systems protecting equipment required for safe shutdown in the event of a Safe Shutdown Earthquake (SSE). The tank is located on the roof of the Reactor Auxiliary Building. The tank is designed in accordance with ASME Section VIII, is seismically designed and supported, and is sized to supply two 75 gpm hose streams for two hours. This conforms with Section C.6.c(4) of BTP CMEB 9.51 and is, therefore, acceptable.

The fire protection water distribution system consists of an underground 12-inch pipe loop around the main plant building. The loop provides two separate flow paths to the internal distribution system in the Fuel Handling, Reactor Auxiliary and Turbine Buildings. The water pressure in the distribution system is maintained at approximately 120 psi by an electrically driven jockey pump. We find this acceptable.

Yard hydrants are provided at intervals of approximately 250 feet along the fire main loop. The lateral to each hydrant is provided with a curb box valve to facilitate hydrant maintenance and repairs without shutting down any part of the water distribution system. Hose houses are installed adjacent to each hydrant and are equipped with a compliment of fire fighting equipment in accordance with NFPA Standard No. 24. We find this acceptable.

All sectional and isolation valves in the fire water distribution system are either post indicator valves (PIV) for underground piping or outside screw and yoke (OS&Y) valves for interior building piping. Supervision has not been provided for all valves in the fire protection water supply system in accordance with NFPA Standards Nos. 26 and 13. To meet out guidelines in Section C.6.c of BTP CMEB 9.51, we will require the applicant to provide locks or electronic tamper switches for all PIV and OS&Y type control and sectional water supply valves.

Sprinkler and Standpipe Systems

The sprinkler systems and standpipe hose systems are independently connected to the looped yard main or from the internal cross connections through buildings so that no single failure in the water supply system will impair both the primary and backup fire protection system.

The wet pipe sprinkler systems, preaction sprinkler system, multicycle sprinkler systems and water spray systems are designed and installed in accordance with the appropriate provisions of NFPA Standards Nos. 13 and 15. The areas equipped with water suppression systems are identified in Appendix 9.5A1 of the applicant's fire hazards analysis.

Interior manual hose stations are provided and equipped so as to reach any location inside a building with at least one effective hose stream. Individual standpipes are 2 1/2, 3 or 4 inches in diameter for multiple hose connections and 2 1/2 inches for single hose connections. Hydraulic calculations have verified that the 2 1/2 and 3 inch diameter piping is capable of supplying at least 100 gpm at 65 psi at the hydraulically most remote outlet. This conforms to the requirements of NFPA Standard No. 14 and is, therefore, an acceptable deviation from Section C.6.c(4) of BTP CMEB 9.51.

The standpipe and hose systems installed in the Reactor Building, Reactor Auxiliary Building and Fuel Handling Building are designed to supply water for manual hose use in areas within hose reach of equipment required to be operational in the event of an SSE. The piping system serving such hose stations have been analyzed for SSE loading and have been provided with supports to assure system integrity. The seismically analyzed standpipe system is supplied from the dedicated 18,000-gallon capacity Auxiliary Fire Water Tank. This conforms with the guidelines of Section C.6.c(4) of BTP CMEB 9.51 and is, therefore, acceptable.

Foam Suppression Systems

Foam fire suppression systems utilizing a 3 percent solution of Agueous Film Forming Foam (AFFF) are provided for the protection of the Diesel Fuel Oil Storage Tank areas. The systems are designed and installed in accordance with NFPA Standard No. 11B. We find this acceptable.

Portable Fire Extinguishers

Portable fire extinguishers are provided throughout the plant in conformance with NFPA Standard No. 10. The extinguisher types used are: dry chemical; carbon dioxide; halon; and water. All portable fire extinguishers are either UL listed or FM approved. This complies with Section C.6.f of BTP CMEB 9.51 and is, therefore, acceptable.

9.5.1.5 Fire Protection of Specific Plant Areas

A. Containment

The applicant has not identified the location and relative position of redundant shutdown systems within containment. Therefore, sufficient information has not been provided to independently verify compliance with our guidelines. We will require that fire protection for redundant shutdown systems in containment comply with Section C.7.a of BTP CMEB 9.51.

The reactor coolant pumps are protected by automatic multicycle sprinkler systems in lieu of an oil collection system. This system does not provide an equivalent level of protection. We will require that the reactor coolant pumps be equipped with an oil collection system in accordance with Section C.7.a of BTP CMEB 9.51.

B. Control Room

The control room is separated from adjacent plant areas by 3-hour fire rated walls, floor and ceiling. A computer tape storage room and supervisor's office,

which is not separated from the control room by 1-hour fire walls, are not equipped with an automatic fire suppression system, and therefore, are not consistent with Section C.7.b of BTP CMEB 9.51. We will require that peripheral rooms in the control room are protected in accordance with Section C.7.b of BTP CMEB 9.51.

Ionization-type smoke detectors are located throughout the control room at the ceiling. Fire detectors have not been provided in control room cabinets and consoles, and therefore, are not consistent with Section C.7.b of BTP CMEB 9.51. We will require that the applicant provide cabinet mounted fire detectors in accordance with Section C.7.b of BTP CMEB 9.51.

The applicant has not completed a fire hazards analysis of the consequences of a fire in the control room on redundant shutdown systems. We will report on this issue in a subsequent SER.

Cable Spreading Rooms

Cable Vault "A" contains control and instrument cables associated with Safety Trains A and C. Cable Vault "B" contains control and instrument cables associated with Safety Trains B and D. The vaults are separated from each other and from adjoining plant areas by walls, floors and ceilings of 3-hour fire resistive construction. All openings in the fire barriers are protected by either 3-hour rated fire doors, dampers or penetration seals. The cable vaults are equipped with preaction type sprinkler systems and areawide smoke detection systems. We find this acceptable. The applicant has not provide sufficient information for us to verify compliance with our guidelines concerning the establishment of aisleways between cable tray stacks. We will require that aisle separation between tray stacks be at least 3 feet wide and 8 feet high in accordance with Section C.7.c of BTP CMEB 9.51.

9.5.1 8 Summary of Deviations from CMEB 9.51

The technical requirements of Appendix R to 10 CFR 50 and Appendix A to BTP ASB 9.51 have been included in BTP CMEB 9.51. Listed below are the deviations from the guidelines of BTP CMEB 9.51 that have been identified and approved:

1. Diameter of standpipe supply piping, Section 9.5.1.4

Conclusion

The following are the open fire protection items:

The applicant has been informed of the necessity to resolve all open items so that all fire protection features can be implemented prior to fuel loading. We will report our review of these unresolved items in a subsequent safety evaluation report.

10 STEAM AND POWER CONVERSION SYSTEM

10.3 Main Steam Supply System

10.3.5 Secondary Water Chemistry

Introduction

In late 1975, we incorporated provisions into the Standard Technical Specifications that required limiting conditions for operation and surveillance requirements for secondary water chemistry parameters. The Technical Specifications for all pressurized water reactor plants that have been issued an operating license from 1974 until 1979 contain either these provisions or a requirements to establish these provisions after baseline chemistry conditions have been determined. The intent of the provisions was to provide added assurance that the operators of newly licensed plants would properly monitor and control secondary water chemistry to limit corrosion of steam generator components such as tubes and tube support plates.

In a number of instances, the Technical Specifications have significantly restricted the operational flexibility of some plants with little or no benefit with regard to limiting degradation of steam generator tube and the tube support plates. Based on this experience and the knowledge gained in recent years, we have concluded that Technical Specification limits are not the most effective way of assuring that steam generator degradation will be minimized.

Due to the complexity of the corrosion phenomena involved and the state-of-the-art as it exists today, we are of the opinion that, in lieu of specifying limiting conditions in the Technical Specification, a more effective approach would be to specify a Technical Specification that required the implementation of a secondary water chemistry monitoring and control program containing appropriate procedures and administrative controls. This has been the approach for control of secondary water programs since 1979.

The required program and procedures are to be developed by applicants with input from their reactor vendor or other consultants, to account for site and plant specific factors that affect water chemistry conditions in the steam generators.

In our view, plant operation following such procedures would provide assurance that licensees would devote proper attention to controlling secondary water chemistry, while also providing the needed flexibility to allow them to deal effectively with an offnormal condition that might arise.

Evaluation

In the FSAR the applicant provided details of a secondary water chemistry monitoring and control program. The information provided in the FSAR was not sufficient for us to complete our evaluation. The applicant provided additional information by letters dated July 15 and September 2, 1983. The information provided by the applicant is inadequate to complete our evaluation. To complete our evaluation, we need the following information;

A summary of operative procedures to be used for the steam generator secondary water chemistry control and monitoring program, addressing the following:

1. Identify the sampling schedule for the critical chemical and other parameters and the control points or limits for these parameters for each operating mode of the plant, i.e., dry layup, cold shutdown, hot standby/shutdown, and power operation.

2. Identify the procedures used to measure the values of the critical parameters, i.e., standard identifiable procedures and/or instruments.

3. Identify the sampling points, considering as a minimum the steam generator blowdown, the hot well discharge, the feedwater, and the demineralizer effluent. We recommend a process flow chart similar to that in EPRI NP-2704-SR "PWR Secondary Water Chemistry Guidelines."

4. State the procedures for recording and management of data, defining corrective actions for various out-of-specification parameters.

5. Identify (a) the authority responsible for interpreting the data and initiating action and (b) the sequence and timing of administrative events required to initiate corrective action.

10.4 Other Features

10.4.6 Condensate Cleanup System

Introduction

The purpose of the Condensate Cleanup System is to remove dissolved and suspended solids from the condensate in order to maintain a high quality of the feedwater being supplied to the steam generators under all normal plant conditions (startup, shutdown, hot standby, power operation). This is accomplished by directing the full flow of condensate to a set of mixed bed demineralizer units. Since the demineralizers need periodic resin regeneration, spare units are provided in the system to replace units taken out of service. The system provides final polishing of the secondary cycle condensate water.

Evaluation

The condensate cleanup system is designed to assist in the control of the secondary side water chemistry and is part of the total control system.

The condensate cleanup system includes all components and equipment necessary for the removal of dissolved and suspended impurities which may be present in the condensate.

We have reviewed the CCS equipment design, materials and system operation in accordance with Section 10.4.6 of Standard Review Plan, NUREG-0800.

The system meets the requirements for condensate cleanup capacity, provides effluent of the required purity, and contains adequate instrumentation to monitor the effectiveness of the system. The system is connected to radioactive waste disposal systems to allow disposal of spent resin or regenerant solutions when required. We have reviewed the sampling equipment, sampling locations,

and instrumentation to monitor and control the CCS process parameters. On the basis of this review, we find that the instrumentation and sampling equipment provided is adequate to monitor and control process parameters.

Based on our review of the applicant's proposed design criteria and design bases for the condensate cleanup system and the requirements for operation of the system, we conclude that the design of the condensate cleanup system and supporting systems meets our guidelines and is, therefore, acceptable. The secondary water chemistry monitoring and control program is evaluated in Section 10.3.5.

Conclusion

Based on the foregoing evaluation, we conclude that the condensate cleanup system meets our guidelines, and is therefore, acceptable.

10.4.7 Condensate and Feedwater System

10.4.8 Steam Generator Blowdown System

Introduction

The Steam Generator Blowdown System (BDS) is designed to maintain the specified water chemistry in the steam generators during all operational modes. The system continuously removes particulate impurities from the blowdown flow by directing the flow through electromagnetic filters and demineralizers before returning to the condenser.

Evaluation

The steam generator blowdown system (SGBS) controls the concentration of chemical impurities and radioactive materials in the secondary coolant. The scope of review of the SGBS included piping and instrumentation diagrams, seismic and quality group classifications, design process parameters, and instrumentation and process controls. The review has included the applicant's evaluation of the proposed system operation and the applicant's estimate of the controlling process parameters.

The steam generator blowdown system is monitored continuously for radiation in the secondary side of the steam generator. Radioactive blowdown is handled routinely in the demineralizer system and the electromagnetic filters. Backwash fluids are handled in Secondary Particulate Waste System and the Radwaste System.

The steam generator blowdown system from steam generator nozzles to the BEX area is designed to seismic Category 1 and ASME III Class 2 requirements up to last isolation valve and downstream up to BEX area. The steam generator blowdown system from BEX area to Seismic Interface Restraint System is designed to seismic Category 1 and ANSI B31.1 requirements. Thus, the SGBS meets the quality standards requirements of General Design Criterion 1 and the seismic requirements of General Design Criterion 2.

The secondary water chemistry monitoring and control program is evaluated in Section 10.3.5.

We have reviewed the SGBDS in accordance with Section 10.4.8 of Standard Review Plan, NUREG-0800.

Instrumentation and automatic controls are provided to monitor and control the operation of the blowdown system, with provision for sampling of the blowdown, in conformance with the guidelines of Branch Technical Position MTEB 53.

Conclusion

Based on the foregoing evaluation, we conclude that the proposed steam generator blowdown system meets our guidelines and is, therefore, acceptable.

12 RADIATION PROTECTION

The staff has evaluated the proposed radiation protection program presented in Chapter 12 of the FSAR against the review guidelines, and criteria set forth in the Standard Review Plan (SRP), NUREG-0800, Section 12. The radiation protection measures at WNP-3 are intended to ensure that internal and external radiation dose to plant personnel and contractors, due to plant conditions, including anticipated operational occurrences, will be within applicable limits of 10 CFR 20, and will be as low as is reasonably achievable (ALARA).

The basis of the staff's acceptance of the WNP-3 radiation protection program is that doses to personnel will be maintained within the limits of 10 CFR 20, "Standards for Protection Against Radiation." The applicant's radiation protection design and program features are consistent with the guidelines of Regulatory Guide 8.8, "Information Relevant To Ensuring That Occupational Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable" (Rev. 3). On the basis of this review, the staff concludes that the radiation protection measures incorporated in the design and the proposed radiation protection program will provide a reasonable assurance that occupational doses will be maintained ALARA and below the limits of 10 CFR 20 both during plant operation and during decommissioning.

12.1 Ensuring that Occupational Radiation Doses are ALARA

The staff has audited the policy considerations, design considerations, and operational considerations contained in the WNP-3 FSAR against the criteria set forth in NUREG-0800, Section 12.1. The staff review consisted of ensuring that the applicant had either committed to following the criteria of the regulatory guides and staff positions referenced in NUREG-0800 (SRP) Section 12.1 or provided acceptable alternatives. In addition, the staff selectively reviewed the applicant's FSAR against the acceptance criteria of the SRP using the review procedures in the SRP. This selective review found the plant acceptable in these areas. Details of the review follow.

12.1.1 Policy Considerations

The applicant provides a management commitment to ensure that WNP-3 will be designed, constructed, and operated in a manner consistent with Regulatory Guides 8.8 and 1.8 "Personnel Selection and Training" (Rev. 1). The applicant has committed to implement a radiation protection program in accordance with Regulatory Guide 8.10 "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable." The overall ALARA responsibility, upper management direction and support lies with the Director of Support Services. The Plant Manager is responsible for the radiological safety of all in-plant personnel and the implementation of ALARA Policy by his staff. The Plant Health Physics/Chemistry Manager is responsible for development of good procedures and radiation protection practices, including preplanning, use of equipment and work techniques.

In addition, line supervisors are also responsible for maintaining plant doses ALARA. The ALARA philosophy was applied during the initial design of the plant. Since then, the applicant has continued to review, update, and modify the plant design and construction phases. The plant's staff - Health Physics and Chemistry, periodically review, update, and modify plant design features and maintenance features as appropriate, using dose data and experience gained from operating nuclear power plants. This is done to ensure that occupational doses will be kept ALARA in accordance with Regulatory Guide 8.8 and NUREG-0800 criteria.

12.1.2 Design Considerations

The objective of plant radiation protection design is to maintain individual personnel doses ALARA as well as the collective doses of all personnel and within the limits of 10 CFR Part 20.

The applicant, using feedback information from operating plants and following guidelines of Regulatory Guide 8.8, has incorporated facility and equipment design improvements at St. Lucie 1 and 2, and Waterford 3, which plants are similar in design to WNP-3, to satisfy the plant's radiation protection design objectives.

Examples of these design features include:

(1) Packaging of units, skid mounted for free access and quick removal to a low radiation area for maintenance or repair.

(2) Most pumps are flanged to facilitate ease of removal to a low radiation area, pump casings are provided with drain connections and pumps are equipped with mechanical seals for greater reliability and reduction in servicing time.

(3) Ion exchangers are designed for complete drainage, spent resin is removed remotely by hydraulically flushing the resin to the solid waste management system.

(4) Absorber (Charcoal) Beds are completely drainable, with single point connections for charcoal removal and equipped with accessories to provide a safe and rapid removal of contaminated charcoal.

(5) Tanks are designed to be isolated for maintenance and are completely drainable, and flushable, with a minimum of crevices to avoid accumulation of radioactive crud.

These design considerations conform with the guidelines of Regulatory Guide 8.8 and NUREG-0800 and are acceptable.

12.1.3 Operational Considerations

The WNP-3 operational considerations included the development of a radiological training program using the guidelines of Regulatory Guide 8.27, "Radiation Protection Training for Personnel at Light-Water Cooled Nuclear Power Plants" and 8.29 "Instruction Concerning Risk From Occupational Radiation Exposure," a radiation zoning and access control system, and general guidelines for workers performing maintenance in high radiation areas. These operational considerations are to ensure that operating and maintenance personnel will follow specific plans and procedures in order to ensure that ALARA goals are achieved in the operation of the plant. High radiation exposure operations are to be preplanned

and carried out by personnel trained in radiation protection and using proper equipment. During such activities, personnel will be monitored for exposure to radiation and contamination. Upon completion of major maintenance jobs, personnel radiation exposures will be evaluated and compared with predicted person-rem exposures. The results are used to make changes in future job procedures and techniques. The plant's health physics management will periodically review radiation dose trends to determine major problem areas and to determine which worker groups are accumulating the highest dose. Plant personnel will use these findings to recommend design modifications or changes in plant procedures. The operational considerations conform to Regulatory Guides 8.8 (Rev. 3), and to NUREG-0800 and are acceptable.

The staff concludes that the policy, design and operational considerations at WNP-3 are adequate to ensure that occupational radiation exposures will be ALARA in accordance with Regulatory Guides 8.8 and 8.10 and meet the criteria of NUREG-0800 and are acceptable.

12.2 Radiation Sources

The staff has audited the contained sources and airborne radioactive material source terms provided in Section 12.2 and Chapter 11 of the WNP-3 FSAR against the criteria set forth in Section 12.2 of NUREG-0800. These source terms are used as inputs for dose assessment and for the design of the shielding and ventilation systems. The staff review consisted of ensuring that the applicant had either committed to following the criteria of the regulatory guides and staff positions referenced in Section 12.2 of NUREG-0800 or provided acceptable alternatives. In addition, the staff selectively compared source terms for specific systems used by the applicant against those used for plants of similar design. This selective review found the plant's source terms equivalent to those used at other plants. Details of the review follow.

12.2.1 Contained and Airborne Sources

Inside the containment during power operation, the greatest potential for personnel dose during operation is due to nitrogen-16, noble gases, and neutrons. Outside the containment and after shutdown inside the containment

the primary sources of personnel exposures are fission products from fuel cladding defects and activation products, including activated corrosion products. Almost all of the airborne radioactivity within the plant is due to equipment leakage. The fission product source terms are based on 1% fuel cladding defect at full power operation. The coolant and corrosion activation product source terms are based on operating experience and reactors of similar design; allowances are included for the buildup of activated corrosion products. Neutron and prompt gamma source terms are based on reactor core physics calculations and operating experience from reactors of similar design. The source terms presented are comparable to estimates by other applicants with similar design and are acceptable.

The applicant has provided a tabulation of the maximum expected radioactive airborne concentrations in equipment cubicles, corridors, and operating areas, from equipment leakage. The bases of these leakage calculations are in accordance with Regulatory Guide 1.112, "Calculations of Releases of Radioactive Materials in Gaseous and Liquid Effluents From Light-Water-Cooled Power Reactors," and are acceptable.

- The WNP-3 FSAR shows maximum expected radioactive airborne concentrations in some plant areas in excess of maximum permissible concentration as defined in 10 CFR 20.203(d)(1)(ii). The applicant should resolve this discrepancy and, until resolution, this item will remain open (471.12).

The ventilation system will be designed to provide sufficient volume changes per hour in occupied areas which may contain significant airborne activity to maintain exposure to personnel ALARA. Air will be routed from areas of low potential airborne contamination to areas of increasing potential airborne contamination. The resulting estimated airborne radioactivity concentrations in frequently occupied areas will be a small fraction of 10 CFR 20.103 limits and are acceptable. In accordance with NUREG-0800, the source terms used to develop these airborne concentration values are comparable to estimates by other applicants with similar design and are acceptable.

12.3 Radiation Protection Design Features

The staff has audited the facility design features, shielding, ventilation, and radiation and airborne monitoring instrumentation contained in the WNP-3 FSAR against the criteria set forth in NUREG-0800, Section 12.3. The staff review consisted of ensuring that the applicant had either committed to following the criteria of the regulatory guides and staff positions referenced in Section 12.3 of NUREG-0800, or provided acceptable alternatives. In addition, the staff selectively reviewed the applicant's FSAR against the specific areas of review and review procedures identified in NUREG-0800. This review found the plant acceptable in these areas. Details of the review follow.

12.3.1 Facility Design Features

The applicant has provided evidence that the dose accumulating functions performed by workers have been considered in the plant design. Features have been included in the design to help maintain doses ALARA in the performance of those functions. These features will facilitate access to work areas, reduce or allow the reduction of source intensity, reduce the time required in the radiation fields, and provide for portable shielding and remote-handling tools. The applicant's facility design features are consistent with the guidance of Regulatory Guide 8.8 (Rev. 3) and NUREG-0800. Therefore, the staff concludes that the facility design features are acceptable.

The applicant has provided five radiation zones as a basis for classifying occupancy and access restrictions for various areas within the plant. On this basis, maximum design dose rates are established for each zone and used as input for shielding of the respective zones. The areas that will have to be occupied on a predictable basis during normal operations and anticipated occurrences are zones so that exposures are below the limits of 10 CFR 20, and will be ALARA. The zoning system and access control features also meet the posting entry requirements of 10 CFR 20.203 or standard NRC Technical Specifications, and are consistent with Regulatory Guide 8.8.

Several features are included in the plant design and operational program to minimize the buildup of activated corrosion products, a major contribution to occupational doses. Examples include:

(1) Most pumps are provided with drain connections to facilitate decontamination.

(2) Ion exchangers are designed for complete drainage and are designed with a minimum of crevices to reduce accumulation of radioactive crud.

(3) A steam generator drain pump is provided to accomplish a more complete and rapid drainage.

(4) An electromagnetic filter is used to remove radioactive corrosion products from coolant.

The applicant's corrosion product control features are consistent with the guidance of Regulatory Guide 8.8 (Rev. 3) and NUREG-0800 and are acceptable.

The design features incorporated by the applicant for maintaining occupational radiation doses ALARA during plant operation and maintenance will also serve to maintain radiation doses ALARA during decommissioning operations and are, therefore, acceptable.

12.3.2 Shielding

The objective of the plant's radiation shielding is to provide protection against radiation for personnel, both inside and outside the plant, during normal operation, including anticipated operational occurrences and during reactor accidents. The shielding was designed to meet the requirements of the radiation dose rate zone system discussed above. The following are several of the shielding design features incorporated into WNP-3.

(1) Reduction of neutron activation of equipment, piping, supports and other materials by providing suitable shielding around the reactor vessel, and to minimize radiation streaming into the reactor cavity and general containment spaces.

(2) Shielding is provided for all equipment anticipated to contain radiation sources.

(3) Shielding discontinuities such as shield plugs, hatch covers, shield doors to high radiation areas are provided with offsets to reduce radiation streaming.

(4) Access labyrinths are provided for areas containing high level radiation sources to preclude a direct radiation path from the equipment to accessible areas.

These shielding techniques are designed to maintain personnel radiation exposures ALARA. Therefore, the staff concludes that the shielding design objectives are acceptable.

The applicant's shielding-design methods included the use of standard computer codes. The applicant also used shielding information from operating nuclear plants as input data for the shield design calculations. The staff concludes that the shielding-design methodology presented is acceptable.

The fuel transfer tube shield structure is a combination of concrete, steel and lead. The design objective of the shield is to completely enclose the fuel transfer tube by shielding materials to prevent inadvertent exposure of personnel to this high radiation source. The access opening is covered by a one foot thick shield of lead shot. The expected dose rate limits are 5 mrem/hr in the most likely occupied areas and 25 mrem/hr in areas of infrequent occupancy and narrow gap areas.

- The applicant stated that portable continuous radiation monitoring equipment with local audible and visual alarms will be provided if conditions warrant. It is our position that all accessible portions of the spent fuel system must be clearly posted with signs stating that potentially lethal radiation fields are possible during fuel transfer. If other than permanent shielding is used, local audible and visible alarming radiation monitors must be installed as required by NUREG-0800. Use of portable radiation alarm monitors to be installed "if conditions warrant" is not

acceptable. The applicant should state in Subsection 12.3.2.3.5 of the FSAR that the access to fuel transfer tube will be in compliance with NUREG-0800. Until then this remains an open item (471.26) (new item).

In accordance with the criteria of Item II.B.2, NUREG-0737, "Clarification of TMI Action Plan Requirements" the applicant has performed a design review of station shielding to allow access to plant areas after an accident.

The systems designed to function after an accident include: Safety Injection, Shutdown Cooling, CVCS, Containment Spray and Recirculation, Sampling, Gaseous Radwaste, Shield Building Ventilation and Control Room Air Air-conditioning systems.

The dose rate calculations were performed for the areas of the above systems by using well known computer codes in superimposing the effects of all sources to obtain the maximum expected dose rate throughout the plant. The radiation environment was evaluated for 1, 2, 4, 8, 12 hours, 1 day, 1 week, 1, 3, 6 months and 1 year following the reactor shutdown following a LOCA with significant core damage. Dose rate zone maps were provided for each relevant area.

Vital areas requiring accessibility following an accident are identified with respect to location, occupancy requirements, and maximum dose levels. Vital areas include: Control Room, TSC, Sampling room, Hot Lab, Health Physics Office, and Counting room.

The shielding design review by the applicant showed the less-than-15 mrems criterion is met by WNP-3 for vital areas requiring extended or continuous occupancy. Additionally, GDC 19 limits are met for those vital areas requiring only infrequent access.

On the basis of its review, the staff has concluded that the applicant has performed a radiation and shielding design review for vital areas access in accordance with Item II.B.2 of NUREG-0737.

12.3.3 Ventilation

The ventilation systems at WNP-3 are designed to protect personnel and equipment from extreme thermal environmental conditions and ensure that plant personnel are not inadvertently exposed to airborne contaminants exceeding those given in 10 CFR 20.103. The applicant intends to maintain personnel exposures ALARA by:

(1) Maintaining airflow from areas of potentially low airborne contamination to areas of higher potential concentrations;

(2) Ensuring negative or positive pressures to prevent exfiltration or infiltration of potential contaminants, respectively; and

(3) Locating ventilation systems intakes so that intake of potentially contaminated air from other building exhaust points is minimized.

The design criteria are in accordance with the guidelines of Regulatory Guide 8.8 (Rev. 3). Some examples of exposure reduction features in the ventilation system are listed below.

(1) Adequate space is provided around the ventilation fans and filter units to allow rapid servicing and replacement of sections and filters.

(2) Pipe equipment vents directly to the appropriate radwaste subsystem for treatment thus preventing spread of contamination.

(3) Welded seams are used throughout the duct work on contaminated systems to the extent possible to reduce system leakage.

(4) Use of filters that can be easily maintained for containing radioactivity so they will not create additional radiation hazards to personnel in normally occupied area.

12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

12.3.4.1 Area Radiation Monitoring Instrumentation

The applicant's area radiation monitoring system is designed to:

(1) Monitor the radiation levels in areas where radiation levels could become significant and where personnel may be present;

(2) Alarm when the radiation levels exceed preset levels to warn of excessive radiation levels; and

(3) Provide a continuous record of radiation levels at key locations throughout the plant.

In order to meet these objectives, the applicant plans to use 73 area monitors located in areas where personnel may be present and where radiation levels could become significant. The area radiation monitoring system is equipped with local and remote audio and visual alarms and a facility for central recording.

- The applicant should inform the NRC whether or not post-accident radiation monitors for containment and sampling area will be provided in compliance with Regulatory Guide 1.97 or should provide a basis for an acceptable deviation (471.1).

  The applicant has provided area radiation monitors around the fuel storage area to meet the requirements of 10 CFR 70.24.

- As per SRP, NUREG-0800 the applicant should commit to the implementation of Regulatory Guide 8.12, "Criticality Accident Alarm Systems" and N16.2-1969 or provide a description of their alternative approach. Until then, this remains an open item (471.23).

  To meet the criteria of the TMI Action Plan Item II.F.1.3, applicant has commited to installing two high-range gamma monitors at WNP-3.

- The range of the monitors does not comply with Table II.F.1-3 of NUREG-0737, thus Table 12.3.4-1 of the FSAR should be revised to show a range of 1 R/hr to 10^7 R/hr (gamma only). Until such revision this remains open item (471.27), (new item).

  In addition, the applicant has provided plant layout drawings showing the location of the high-range monitors.

12.3.4.2 Airborne Radioactivity Monitoring Instrumentation

The design objectives of the applicant's airborne radioactivity monitoring system are:

(1) To assist in maintaining occupational radiation exposure to airborne contaminants ALARA;

(2) To check on the integrity of systems containing radioactivity which are being monitored; and

(3) To warn of inadvertent release of airborne radioactivity to prevent over-exposure of personnel.

In order to meet these objectives, the applicant plans to use ventilation duct monitors in key locations throughout the plant and portable continuous air monitors. The ventilation system monitors will be used to provide representative air concentrations and rapid indication of abnormal conditions at fixed locations such as exhaust ducts from areas in which the airborne radioactivity could increase and in which personnel normally have access, consolidated ventilation exhausted from the plant, and air intake ducts to the control room for post-accident habitability monitoring purposes. Portable or mobile air monitors can be relocated to virtually any location of the plant, and can be connected to the plant radiation monitoring communication's system through the spare junction boxes located throughout the plant.

All radiation monitors will be periodically calibrated with standard sources traceable to the NBS.

- The applicant should describe how the radioactive airborne monitoring system will detect the MPC-hours of radioactivity (particulates, iodine, and noble gases) from any compartment which could contain airborne radioactivity and which could be occupied by personnel as per SRP, NUREG-0800, Section 12.3-12.4, 4.b.1. (471.21).

12.4 Dose Assessment

The staff has audited the applicant's dose assessment for the WNP-3 provided in Section 12.4 of the FSAR, against the criteria set forth in NUREG-0800, Section 12.3-12.4. This review consisted of ensuring that the applicant had either committed to following the criteria of the regulatory guides and staff positions referenced in Section 12.3-12.4 of NUREG-0800, or provided acceptable alternatives. In addition, the staff selectively compared the dose assessment made by the applicant for specific functions against those made for other plants of similar design. This selective review found the plant's dose assessment equivalent to those of other plants. Details of the review follow.

The applicant has performed an assessment of the doses that will be received by plant and contractor personnel. This dose assessment is based on occupancy, factors, expected dose rates, expected airborne radioactivity concentrations, and historical information from operation BWR power plants. The dose assessment includes a breakdown of the annual person-rem doses associated with major functions such as routine operations, routine maintenance, inservice inspections, special maintenance, radwaste processing, refueling, and health physics. The applicant estimated the total annual collective dose to plant personnel and contractors to be 440 person-rems. This estimate is consistent with the acceptance criteria in NUREG-0800, that is, using the methods outlined in Regulatory Guide 8.19.

Currently, operating BWRs average 740 person-rems per unit annually, with particular plants experiencing an average lifetime annual dose as high as 1850 person-rems. These dose average are based on widely varying yearly doses for BWRs. The staff finds the bases for the WNP-3 exposure estimate acceptable.

- The dose breakdown for refueling work in Table 12.4-8 does not agree with dose for refueling work given in Table 12.4-2. The applicant should resolve the discrepancy (471.14).

The applicant has provided a tabulation of the maximum expected radioactive airborne concentrations, as well as estimates of the inhalation dose equivalent rates to plant personnel. The dose equivalent rates are derived from the airborne radioactivity source terms given in Chapter 11 of the FSAR. The applicant's assumptions and models on which his internal and submersion dose estimates are based for occupational exposures are consistent with those of the staff and are acceptable.

The staff concludes that the applicant's dose assessments for contained sources and airborne radioactive material are comparable to estimates by other applicants with similar design and are acceptable.

12.5 Operational Radiation Protection Program

The staff has audited the organization, equipment instrumentation, facilities, and procedures for radiation protection contained in the WNP-3 FSAR against the criteria of NUREG-0800, Section 12.5. The plant's health physics program objectives are to provide reasonable assurance that the limits of 10 CFR 20 are not exceeded, to further reduce unavoidable exposures, and to ensure that individual and total person-rem occupational radiation doses are maintained ALARA. The staff review consisted of ensuring that the applicant had either committed to following the criteria of the regulatory guides and staff positions referenced in Section 12.5 of NUREG-0800 or provided acceptable alternatives and selectively compared the applicant's FSAR against the acceptance criteria of the SRP using review procedures in NUREG-0800. This selective review found the plant acceptable in these areas. Details of the review follow.

12.5.1 Organization

The Health Physics/Chemistry manager at WNP-3, in conjunction with line supervisors, is responsible for implementing and enforcing the plant's health physics program. However, the ultimate responsibility of the health physics program lies with the Plant Manager.

The WNP-3 station radiation protection organization has been evaluated in accordance with the position of NUREG-0731, "Criteria for Utility Management and Technical Organization," Regulatory Guide 8.8 (Section C.1.b(21,31)) and NUREG-0800.

The paragraphs below present an evaluation of how the health physics organization for WNP-3 compares with the various staff positions concerning plant organization and management criteria.

(1) The organization description for WNP-3 shows that the Health Physics/Chemistry Manager reports to the Plant Manager. This satisfies the requirements of Regulatory Guide 8.8 and is acceptable.

(2) The health physics and chemistry function at WNP-3 are separated into a Health Physics Section and Chemistry Section that are supervised by a Health Physics Supervisor and a Chemistry Supervisor. This satisfies the recommendation of NUREG-0731 and is acceptable.

(3) The applicant has shown that the qualifications of Health Physics/Chemistry Manager meets the requirements of Regulatory Guide 1.8, "Personnel Selection and Training," and is acceptable as the station's RPM.

(4) The applicant has committed to using the criteria of ANSI 3.1, December 1979 draft, in selecting the individual temporarily filling the Health Physics/Chemistry Manager's position. This satisfies the requirement of NUREG-0731 and is acceptable.

(5) The applicant has proposed to train health physics technicians in accordance with the criteria of ANSI/ANS 3.1-1978, which requires one year of related technical training and two years of experience. This is equivalent to the criteria of ANSI 18.1 which also requires such training and experience and therefore is acceptable.

(6) The applicant has committed to having at least one health physics technician on site at all times. This satisfies the criteria of NUREG-0731 and is acceptable.

WNP-3 has, shown that the current health physics organization meets staff criteria as stated in NUREG-0731, Regulatory Guide 8.8 and NUREG-0800 for an acceptable radiation protection organization.

12.5.2 Equipment, Instrumentation, and Facilities

The radiation protection features at WNP-3 station include a counting room, a radiochemistry laboratory, a conventional chemistry laboratory, health physics offices, calibration facility, laundry facility, personnel decontamination, and dry cleaning area. These facilities are sufficient to maintain occupational radiation exposure ALARA and are consistent with the provisions of Regulatory Guide 8.8. Equipment to be used for radiation protection purposes includes portable radiation survey instruments, personnel monitoring equipment, fixed and portable area and airborne radioactivity monitors, laboratory equipment, air samplers, respiratory protection equipment, and protective clothing.

In order to meet the criteria of III.D.3.3 of NUREG-0737, the applicant has committed to having the post-accident-capability to sample and determine airborne radioiodine concentrations by using portable air samplers and using silver zeolite as a sample medium. If entrapped noble gases interfere with the radioiodine analysis, clean air or nitrogen flushing will be performed.

- The Regulatory Guide 1.70 and the SRP (NUREG-0800) state that the description of health physics instrumentation should include the instruments sensitivity. The applicant provided (in Table 12.5-1) the type of radiation the instruments detect and not the instrument's sensitivity. The applicant should provide the instrument's sensitivity (471.6).

Low background counting facilities for postaccident analysis are available. The use of sampling equipment and analysis systems for the determination of radioiodine during an accident situation has been incorporated into the WNP-3 station's training program.

The postaccident radioiodine sampling and analysis provisions described for the WNP-3 station satisfactorily meet the staff's position as outlined in NUREG-0737 and are acceptable.

12.5.3 Procedures

All station personnel entering controlled radiation areas will be assigned thermoluminescent dosimeter (TLD) badges and pocket dosimeters. Special neutron surveys will be provided when plant personnel enter neutron areas when required by 10 CFR 20. Whole-body counts of all plant personnel will be conducted on a scheduled basis and other bioassays will be provided when deemed necessary by the plant's health physics staff, using the guidance of Regulatory Guide 8.9 and 8.26. All radiation exposure information will be processed and recorded in accordance with 10 CFR 20.

Maintenance, repair, surveillance, and refueling procedures and methods used by the applicant are reviewed to ensure that all plant radiation protection procedures, and practices, and criteria have been considered, to ensure that occupational radiation exposures will be ALARA and in accordance with Regulatory Guide 8.8. Procedures are also developed to meet the requirements of Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)."

Based on the information presented in the FSAR and the applicant's responses to the staff's questions, the staff concludes that the applicant intends to implement a radiation protection program that will maintain inplant radiation exposures within the applicable limits of 10 CFR Part 20 and will maintain exposures ALARA in accordance with Regulatory Guide 8.8.

13 CONDUCT OF OPERATIONS

13.1 Organizational Structure of Applicant

13.2 Training

The applicant's training programs for licensed reactor operators and nonlicensed plant staff were reviewed according to SRP 13.2 (NUREG-0800). The staff acceptance criteria included applicable portions of 10 CFR Parts 19, 50 and 55, and Regulatory Guide 1.8, as well as the TMI Action Plan (NUREG-0737) and H. R. Denton's letter of March 28, 1980 to all power reactor applicants and licensees.

13.2.1 Licensed Operator Training Program

A training program for WNP-3 licensed reactor operators has been implemented to develop and maintain an organization fully qualified to operate the plant and maintain the plant safety. The initial and requalification programs, which are designed to meet the requirements of 10 CFR Parts 50 and 55, and TMI Action Plan related requirements, are based on the individual employee's level of education, experience and skills as well as on the level of assigned responsibility and intended position.

13.2.1.1 Initial Training Program

The initial training program for personnel who will be licensed consists of the following discrete segments:

(1) Academic and Nuclear Plant Fundamental

This training course will be approximately twenty weeks in length and is designed to provide individuals with basic knowledge in science and technology of power plant operations. The major areas to be covered are mathematics, physics, basic nuclear physics, reactor theory, radiation protection, chemistry,

instrumentation and control, health physics, electrical theory, transient analysis, fluid flow, thermodynamics and heat transfer. The extent of participation will be determined by the individual's experience and education level. An integral part of the fundamental training program is reactor startup experience. This is a one-week training course conducted at a research reactor for cold license candidates only.

With respect to instruction in the topics of fluid flow, thermodynamics and heat transfer, we require the applicant to provide a program in accordance with the guidelines as outlined in Enclosure 2 of H. R. Denton's March 28, 1980 letter. We will review the program when it is received and report our findings in the final SER.

(2) Plant Systems - Classroom

This training course is designed to provide cold license candidates with an in-depth study of the WNP-3 systems and equipment. The course consists of approximately nine weeks of classroom lectures on nuclear steam supply system and balance of the plant system design, components and operation; instrumentation, control and electrical system design and operation; safety analysis and technical specifications; and operating and emergency experience. Effectiveness of this training will be monitored through examinations.

In addition to the above topics, we require (as specified in Enclosure 1 of H. R. Denton's March 28, 1980 letter) the applicant to modify the program to provide training in the use of installed plant systems to control or mitigate an accident in which the core is severely damaged. We will review this modification, when it is received, and report our findings in the final SER.

(3) Plant Systems - Observation

This segment of the training program consists of four weeks of plant observation. The major objective of this program is to familiarize each cold license candidate with the daily routine involved in the operation of a PWR of similar design.

(4) Simulator Training

This training program will be approximately ten weeks in length on the WNP-3 simulator, and will consist of classroom lectures, simulator control room lectures and demonstration, and simulator control room exercises. The training will include, but not be limited to the following:

- Classroom sessions consisting of lectures, seminars, and examinations.

- Demonstrations of how to control individual systems and the integrated plant.

- Practice of normal and emergency plant operations, including recognition of emergency conditions and response to malfunctions by the license candidate.

- Exercises during which the license candidates operate the simulator without instructor assistance and receive evaluation of ability to safely and efficiently operate the plant.

Simulator sessions will also include all the control manipulations as listed in Section 13.2.1.2.1(2)(b) of this report. At the conclusions of the simulator training phase, each candidate will be given examinations to determine his ability to control the operations of the plant in a safe and competent manner.

(5) Onsite Experience

Training in the form of practical work assignments at the WNP-3 will be provided for approximately 26 weeks. Work assignments may include: plant operating procedure preparation and verification, preoperational testing of plant systems, participation in hot functional testing program, low power physics and escalation to power test programs, and preparing and providing instruction on plant systems. Emphasis shall be on the license candidate gaining thorough knowledge of WNP-3.

(6) Senior Operator and Shift Manager Duties

This training phase, designed to train senior reactor operators and shift managers, is approximately one to three weeks in duration. The program consists of: leadership, communication, motivation of personnel, problem analysis, decision analysis, command responsibility and limits, and administrative requirements for the particular SRO position.

(7) License Review Training

This is an approximately four week course designed to improve the weak areas brought out from a comprehensive examination and to bring the license candidates to a peak knowledge level for the NRC examinations.

(8) Training Program Evaluation

The performance of employees participating in the cold license training program are monitored and evaluated throughout the program. Frequent examinations are given to license candidates in order to determine the effectiveness of the training and knowledge of the trainees.

Based on our review, we find that the applicant's initial training program conforms to the requrements of the applicable portions of 10 CFR Parts 50 and 55, and follows the guidance given in Regulatory Guide 1.8. However, as noted in the above Sections 13.2.1.1(1) and (2) of this report, the applicant's initial training program does not satisfy the requirements as outlined in H. R. Denton's March 28, 1980 letter to all power reactor applicants and licensees. Thus, we have not been able to conclude that the applicant's initial training program for reactor operators and senior reactor operators is acceptable.

13.2.1.2 Licensed Operator Requalification and Replacement Training Programs

Following the initial licensing of cold license candidates, requalification and replacement training programs will be implemented to maintain and demonstrate the continued competence and the level of proficiency of all licensed personnel.

13.2.1.2.1 Requalification Training Program

A requalification training program conducted by the applicant for all licensed reactor operators and senior reactor operators will be implemented shortly after receipt of cold licenses. This program will be conducted on a repetitive two-year cycle and will consist of the following:

(1) Lecture Series

The applicant has indicated that at least six pre-planned requalification training lectures will be scheduled throughout the year. Lecture subjects and content will be based on the results of the annual examination administered to licensed reactor operators and senior reactor operators. However, the content of the examination described in the FSAR by the applicant does not cover all the subjects as listed in Appendix A of 10 CFR Part 55. We will require the content of the annual examinations to be modified to include the following subjects as listed in Appendix A of 10 CFR Part 55 as well as in Enclosure 1 of H. R. Denton's March 28, 1980 letter:

- Theory and principles of operation
- General and specific plant operating characteristics
- Plant instrumentation and control systems
- Plant protection systems
- Engineered safety systems
- Normal, abnormal, and emergency operating procedures
- Radiation control and safety
- Technical specifications
- Applicable portions of Title 10, Charter I, Code of Federal Regulations

- Heat transfer, fluid flow, thermodynamics and mitigation of accidents involving a degraded core.

The annual written examination results will indicate the scope and depth of training needed by each individual in the above areas.

We will review the applicant's modification to include instruction of the above subjects in the Lecture Series and report our findings in the final SER.

(2) On-the-Job Training

The on-the-job training portion of the requalification program will consist of the following:

(a) Simulator Training

Each licensed reactor operator will spend 40 hours annually in a simulator training program. The program will contain instruction in facility changes, recognition of emergency conditions, and operating experience at WNP-3 and similar plants.

(b) Control Manipulations

Licensed reactor operators will manipulate and direct or evaluate the activities of those manipulating the station controls through the following reactivity changes during the term of their licenses. The asterisked items will be performed annually and all other items will be performed on a two-year cycle.

* • Plant or reactor start-ups to include a range that reactivity feedback from nuclear heat addition is noticeable and heat up rate is established

• Plant shutdown

* • Manual control of steam generators and/or feedwater during start-up and shutdown

- Boration and/or dilution during power operation

* • Any significant (≥10%) power changes due to manual changes in control rod position or boron concentration

* • Loss of coolant including:

  1. Significant PWR steam generator tube leaks
  2. Inside primary containment
  3. Large and small, including leak-rate determination
  4. Saturated Reactor Coolant response (PWR)

- Loss of instrument air

- Loss of electrical power (and/or degraded power sources)

* • Loss of core coolant flow/natural circulation

- Loss of condenser vacuum

- Loss of service water if required for safety

- Loss of shutdown cooling

- Loss of component cooling system or cooling to an individual component

- Loss of normal feedwater or normal feedwater system failure

* • Loss of all feedwater (normal and emergency)

- Loss of protective systems channel

- Mispositioned control rod or rods (or rod drops)

- Inability to drive control rods

- Conditions requiring use of emergency boration of standby liquid control system

- Fuel cladding failure or high activity in reactor coolant or offgas

- Turbine or generator trip

- Malfunction of automatic control system(s) which affect reactivity

- Malfunction of reactor coolant pressure/volume control system

- Reactor trip

- Main steam line break (inside or outside containment)

- Nuclear Instrumentation failure(s)

The above control manipulations will be performed on the simulator and/or the plant.

(c) Knowledge of Facility Design, Procedure, and License Changes

The applicant has not addressed the instructions of procedure changes and facility license changes. As described in Appendix A of 10 CFR Part 55, we require the applicant to provide a training program to ensure that each licensed reactor operator and senior reactor operator will be cognizant of facility design changes, procedure changes, and facility license changes. We will review the applicant's modification of the program to incude instructions of these subjects and report our findings in the final SER.

(d) Knowledge of Abnormal and Emergency Procedures

In order to ensure a continuing awareness of the actions and responses necessary during abnormal and emergency situations, as described in the Appendix A of 10 CFR Part 55, we require the applicant to provide a program to ensure that each licensed reactor operator and senior reactor operator will review the

contents of all abnormal and emergency procedures on a regularly scheduled basis. We will review this program when it is received and report our findings in the final SER.

(3) Evaluation

The evaluation program for licensed personnel includes the following:

(a) Annual Written Examination

An annual written examination will be given to each licensed reactor operator and senior reactor operator. The examination will contain the categories as described under Lecture Series. The applicant has indicated that a grade of less than 70% in any category shall require accelerated requalification in that category. A grade of less than 75% overall requires accelerated requalification in all categories graded less than 75%.

As specified in H. R. Denton's March 28, 1980 letter, we require the above criteria for accelerated requalification to be modified to be consistent with the new passing grade for issuance of a license; 80% overall and 70% each category. We will review the applicant's modification to the above criteria and report our findings in the final SER.

(b) Annual Oral Examination and Performance Observation

An annual oral requalification examination will be given. In addition, each licensed reactor operator and senior reactor operator will be evaluated annually on his performance. The evaluation will include observation of performance during actual or simulated plant conditions. Any individual given an unsatisfactory overall evaluation will require accelerated requalification.

(4) Accelerated Requalification

Individuals requiring accelerated requalification as a result of annual examination will not perform licensed-duties until successfully completing the program. Accelerated requalification will be given in the categories required or areas

identified in the written or oral examination. Successful completion of the program will be measured by a reexamination of the individual categories, repeating an entire written examination or repeating the oral examination. Successful completion of an accelerated requalification program will be by grade criteria identified in the written and oral sections above.

13.2.1.2.2 Replacement Training Program

Replacement training will be conducted to fill vacancies and will prepare individuals for increased responsibility in the supervisory, technical or operating staff. Replacement personnel will receive training comparable to that received by the initial staff. This will ensure that the required level of proficiency is maintained.

As noted in the above Sections 13.2.1.2(1), 13.2.1.2.1(2)(c) and (d), and 13.2.1.2.1(3)(c) of this report, we find that the applicant's requalification and replacement training programs do not satisfy the requirements specified in Appendix A of 10 CFR Part 55 and in the letter from H. R. Denton to all power reactor reactor applicants and licensees dated March 28, 1980. Therefore, we have not been able to conclude that the applicant's requalification and replacement training programs for reactor operators and senior reactor operators are acceptable.

13.2.1.2.3 TMI Related Requirements for New Operating License

I.A.2.1 Immediate Upgrading Reactor Operator and Senior Reactor Operator Training and Qualifications

The applicant has established a program intended to assure that all reactor operator and senior reactor operator license candidates have the prescribed experience, qualification and training.

The applicant has indicated that certifications completed pursuant to Section 55.10(a)(6) and 55.33a(4) and (5) of 10 CFR Part 55 shall be made by the Plant Manager. However, as specified in Enclosure 1 of H. R. Denton's March 28, 1980 letter, we require that all licensed operator candidates will be certified

competent to take the NRC license examinations by the highest level of corporate management for plant operation (for example, Vice President for Operations) prior to application for the examinations.

As an operating license applicant, WNP-3 is not subject to the one year of experience requirements for cold license SRO candidates. However, after one year of station operation, individuals applying for an SRO license will be required to comply with the one year experience requirement for hot license SRO applicants, unless previously experienced in an equivalent position at another nuclear plant or at a military propulsion reactor. The experience of license applicants in the latter category will be documented by the applicant on a case-by-case basis in sufficient detail so that the staff can make a finding regarding equivalency. SRO license applicants who possess a degree in engineering or applicable sciences are considered to meet the one year experience requirements as an RO provided they: (1) satisfy the requirements set forth in Sections A.1.a and A.2 of Enclosure 1 to the letter from H. R. Denton to all power reactor applicants and licensees, dated March 28, 1980, and (2) have participated in a training program equivalent to that of a cold senior reactor operator applicant. The applicant has not committed to comply with the above requirements.

Also, the requirement for three months on-shift experience for control room operators an SRO candidates as an extra person on shift is not required for cold license candidates and, hence, is not applicable to WNP-3. However, WNP-3 will comply with this requirement for hot license candidats after three months of station operation.

The applicant's training program includes topics in heat transfer, fluid flow, and thermodynamics. However, the applicant has not provided a program for the instruction of these topics in accordance with Enclosure 2 of H. R. Denton's March 28, 1980 letter. We require the applicant to provide this program for review, and we will report our findings in the final SER.

Reactor and plant transient training is primarily performed by each license applicant at a simulator facility and includes classroom discussions of typical transients as well as demonstration of casualty and transient response on the

simulator. This knowledge is tested in-depth during the certification examination given by the training facility. Based on our review, we have not been able to conclude that the applicant of WNP-3 has satisfied the requirements of this item of the TMI Action Plan.

I.A.2.3 Administration of Training Program

As specified in Enclosure 1 of H. R. Denton's March 28, 1980 letter, we require that all instructors who teach systems, integrated responses, transient and simulator courses shall be SRO certified and will continue to participate in appropriate requalification programs. Vendor-supplied instructors who teach the above subjects shall also be similarly certified. Other members of the permanent or nonpermanent training staff who are responsible for teaching technical subjects, such as reactor theory, heat transfer, fluid mechanics, thermodynamics, health physics, chemistry, and instrumentation are not expected to have an RO or SRO license. Guest lecturers considered to be used on a limited bases shall be monitored by a qualified instructor. These guest lecturers are exempt from the SRO criterion.

Based on our review, we find that the applicant of the WNP-3 has not committed to comply with the above requirements of this item of the TMI Action Plan.

II.B.4 Training for Mitigating Core Damage

As specified in Enclosure 3 of H. R. Denton's March 28, 1980 letter, we require that shift technical advisors and personnel in the operating chain up to and including the plant manager will receive training for mitigating core damage. Managers and technicians in the instrumentation and control, health physics and chemistry departments will receive mitigating core damage training commensurate with their responsibilities.

Based on our review, we find that the applicant has not complied with the above requirements. In addition, the applicant has not provided, for NRC review, a training program for mitigating core damage in accordance with the guidance as specified in Enclosure 3 of H. R. Denton's March 28, 1980 letter. We will review this training program when it is received and will report our findings in the final SER.

13.2.2 Training for Nonlicensed Plant Staff

The applicant has described in the FSAR the details of the training given to nonlicensed plant personnel. The training program for nonlicensed personnel will provide training for maintenance personnel, equipment operators, health physics and chemistry technicans, management and supervisory personnel, technical personnel and training instructors.

All permanently employed plant personnel will participate in a general employee training program consisting of, but not limited to radiological health and safety, quality assurance, industrial safety, plant security, station emergency plan, fire protection and other appropriate plant plans and procedures.

The applicant has not provided a training program for the Shift Technical Advisors (STA). We require the applicant to provide for our review a training program for the STA in accordance with the guidance as specified in NUREG-0737, Appendix C. We will report the results of our review in the final SER.

The fire protection training program includes classroom instructions and training in fire fighting equipment use, strategies, techniques and periodic drills. We conclude that the applicant's fire protection training program conforms to the guidance given in the Standard Review Plan, Section 13.2.2.II.C.A and is acceptable.

Based on our review, we find that the training given for nonlicensed plant staff personnel meets the requirements of 10 CFR Part 19 and Part 50 and follows the guidance given in Regulatory Guide 1.8. Therefore, we conclude that the applicant's training program for nonlicensed plant staff, with the exception of the STA training program, is acceptable.

13.3 Emergency Planning

The applicant has submitted emergency plans required by the upgraded regulations on emergency planning that were published in the Federal Register on August 19, 1980, and became effective on November 3, 1980. The regulations contain a

revised Appendix E to 10 CFR 50, "Emergency Planning and Preparedness for Production and Utilization Facilities," which establishes minimum requirements for an acceptable state of onsite emergency preparedness, and a new 10 CFR 50.47, "Emergency Plans," which specifies standards that must be met for both onsite and offsite emergency response. This latter section incorporates the joint NRC/Federal Emergency Management Agency (FEMA) standards for use in evaluating State and local radiological emergency plans and preparedness.

The applicant's Emergency Plan, Revision 0 dated April 12, 1982, is in the process of review by the NRC. Following this review, requests for additional information may be generated. When this information is submitted, the NRC staff will review the information and make a finding on its adequacy.

NRC and FEMA have agreed that FEMA will make a finding and determination as to the adequacy of State and local government emergency response plans. NRC will determine the adequacy of the applicant's emergency response plans with respect to the standards listed in 10 CFR 50.47(b), the requirements of Appendix E to 10 CFR 50, and the guidance in NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," dated November 1980. After the above determinations are made by NRC and FEMA, the NRC staff will make a finding in the licensing process as to the overall and integrated state of preparedness. In accordance with 10 CFR 50.47(a), a full power operating license will not be issued unless the NRC staff's overall finding is such that the state of onsite and offsite emergency preparedness provides reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. The NRC staff will provide its evaluation in a supplement to this report.

13.6 Industrial Security

13.6.1 Introduction

The Washington Public Power Supply System has filed with the Nuclear Regulatory Commission for the WNP-3 site a Physical Security Plan, Safeguards Contingency Plan, and a Security Training and Qualification Plan.

This Safety Evaluation Report (SER) summarizes how the applicant has provided for meeting the requirements of 10 CFR Part 73. The SER is composed of a basic analysis that is available for public review, and a protected Appendix.

13.6.2 Physical Security Organization

To satisfy the requirements of 10 CFR 73.55(b) the Washington Public Power Supply System has provided a physical security organization that includes a Shift Supervisor who is onsite at all times with the authority to direct the physical protection activities. To implement the commitments made in the physical security plan, training and qualification plan, and the safeguards contingency plan, written security procedures specifying the duties of the security organization members have been developed and are available for inspection. The training program and critical security tasks and duties for the security organization personnel are defined in the "WNP-3 Security Personnel Training and Qualification Plan" which meets the requirements of 10 CFR Part 73, Appendix B for the training, equipping and qualification of the security organization members. The physical security plan and the training program provide commitments that preclude the assignment of any individual to a security related duty or task prior to the individual being trained, equipped and qualified to perform the assigned duty in accordance with the approved guard training and qualification plan.

13.6.3 Physical Barriers

In meeting the requirements of 10 CFR 73.55(c) the applicant has provided a protected area barrier which meets the definition in 10 CFR 73.2(f)(1). An isolation zone, to permit observation of activities along the barrier, of at least 20 feet is provided on both sides of the barrier with the exception of the locations listed in the Appendix. The staff has reviewed those locations and determined that the security measures in place are satisfactory and continue to meet the requirements of 10 CFR 73.55(c).

Illumination of 0.2 foot-candles is maintained for the isolation zones, protected area barrier, and external portions of the protected area. In areas

where illumination of 0.2 foot-candles cannot be maintained, special procedures are applied as described in the Appendix.

13.6.4 Identification of Vital Areas

The Appendix contains a discussion of the applicant's program and identifies those areas and equipments determined to be vital.

Vital equipment is located within vital areas which are located within the protected area and which requires passage through at least two barriers, as defined in 10 CFR 73.2(f)(1) and (2), to gain access to the vital equipment. Vital area barriers are separated from the protected area barrier.

The control room and central alarm station are provided with bullet-resistant walls, doors, ceilings, floors, and windows. Based on these findings and the analysis set forth in paragraph D of the Appendix, the staff has concluded that the applicant's program for identification and protection of vital equipment satisfies the regulatory intent. However, this program is subject to onsite validation by the staff in the future, and to subsequent changes if found to be necessary.

13.6.5 Access Requirements

In accordance with 10 CFR 73.55(d) all points of personnel and vehicle access to the protected area are controlled. The individual responsible for controlling the final point of access into the protected area is located in a bullet-resistant structure. As part of the access control program, vehicles (except under emergency conditions), personnel, packages, and materials entering the protected area are searched for explosives, firearms and incendiary devices by electronic search equipment and/or physical search.

Vehicles admitted to the protected area, except licensee designated vehicles, are controlled by escorts. Licensee designated vehicles are limited to on-site station functions and remain in the protected area except for operational maintenance, repair, security and emergency purposes. Positive control over the vehicles is maintained by personnel authorized to use the vehicles or by the

escort personnel. A picture badge/key card system, utilizing encoded information, identifies individuals that are authorized unescorted access to protected and vital areas, and is used to control access to these areas. Individuals not authorized unescorted access are issued non-picture badges that indicate an escort is required. Access authorizations are limited to those individuals who have a need for access to perform their duties.

Unoccupied vital areas are locked and alarmed. During periods of refueling or major maintenance, access to the reactor containment is positively controlled by a member of the security organization to assure that only authorized individuals and materials are permitted to enter. In addition, all doors and personnel/equipment hatches into the reactor containment are locked and alarmed. Keys, locks combinations and related equipment are changed on an annual basis. In addition, when an individual's access authorization has been terminated due to the lack of reliability or trustworthiness, or for poor work performance, the keys, locks, combinations and related equipment to which that person had access are changed.

13.6.6 Detection Aids

In satisfying the requirements of 10 CFR 73.55(e) the applicant has installed intrusion detection systems at the protected area barrier, at entrances to vital areas, and at all emergency exits. The applicant has exceeded the regulation by providing two separate and dissimilar perimeter intrusion detection systems at the protected area barrier. Alarms from the intrusion detection system annunciate within the continuously manned central alarm station and a secondary alarm station located within protected area. The central alarm station is located such that the interio f the station is not visible from outside the perimeter of the protected area. In addition, the central station is constructed so that walls, floors, ceilings, and doors are bullet-resistant. The alarm stations are located and designed in such a manner so a single act cannot interdict the capability of calling for assistance or responding to alarms. The central alarm station contains no other functions or duties that would interfere with its alarm response function. The intrusion detection system transmission lines and associated alarm annunciation hardware are self-checking and tamper-indicating. Alarm annunciators indicate the type of alarm and its location when

activated. An automatic indication of when the alarm system is on standby power is provided in the central alarm station.

13.6.7 Communications

As required in 10 CFR 73.55(f) the applicant has provided for the capability of continuous communications between the central and secondary alarm station operators, guards, watchmen, and armed response personnel through the use of a conventional telephone system, and a security radio system. In addition, direct communication with the local law enforcement authorities is maintained through the use of a conventional telephone system and two-way FM radio links.

All non-portable communication links, except the conventional telephone system, are provided with an uninterruptible emergency power source.

13.6.8 Test and Maintenance Requirements

In meeting the requirements of 10 CFR 73.55(g) the applicant has established a program for the testing and maintenance of all intrusion alarms, emergency alarms, communication equipment, physical barriers and other security related devices and equipment. Equipment or devices that do not meet the design performance criteria or have failed to otherwise operate will be compensated for by appropriate compensatory measures as defined in the "WNP-3 Physical Security Plan" and in site procedures. The compensatory measures defined in these plans will assure that the effectiveness of the security system is not reduced by failures or other contingencies affecting the operation of the security related equipment or structures. Intrusion detection systems are tested for proper performance at the beginning and end of any period they are used for security. Such testing will be conducted at least once every seven days.

Communication systems for onsite communications are tested at the beginning of each security shift. Offsite communications are tested at least once each day.

Audits of the security program are conducted once every 12 months by personnel independent of site security management and supervision. The audits, focusing

on the effectiveness of the physical protection provided by the onsite security organization implementing the approved security program plans, include, but are not limited to: a review of the security procedures and practices; system testing and maintenance programs; and local law enforcement assistance agreements. A report is prepared documenting audit findings and recommendations and is submitted to the Plant Management.

13.6.9 Response Requirements

In meeting the requirements of 10 CFR 73.55(h) the applicant has provided for armed responders immediately available for response duties on all shifts consistent with the requirements of the regulations. Considerations used in support of this number are attached (see Appendix). In addition, liaison with local enforcement authorities to provide additional response support in the event of security events has been established and documented.

The applicant's safeguards contingency plan for dealing with thefts, threats, and radiological sabotage events satisfies the requirements of 10 CFR Part 73, Appendix C. The Plan identifies appropriate security events which could initiate a radiological sabotage event and identifies the applicant's preplanning, response resources, safeguards contingency participants and coordination activities for each identified event. Through this plan, upon the detection of abnormal presence or activities within the protected or vital areas, response activities using the available resources would be initiated. The response activities and objectives include the neutralization of the existing threat by requiring the response force members to interpose themselves between the adversary and their objective, instructions to use force commensurate with that used by the adversary, and authority to request sufficient assistance from the local law enforcement authorities to maintain control over the situation.

To assist in the assessment/response activities a closed circuit television system, providing the capability to observe the entire protected area perimeter, isolation zones and a majority of the protected area, is provided to the security organization.

13.6.10 Employee Screening Program

In meeting the requirements of 10 CFR 73.55(a) to protect against the design basis threat as stated in 10 CFR 73.1(a)(1)(ii), the Washington Public Power Supply System has provided an employee screening program. Personnel who successfully complete the employee screening program or its equivalent may be granted unescorted access to protected and vital areas at the WNP-3 site. All other personnel requiring access to the site are escorted by persons authorized and trained for escort duties and who have successfully completed the employee screening program.

The employee screening program is based upon accepted industry standards and includes a background investigation, a psychological evaluation, and a continuing observation program. In addition, the applicant may recognize the screening program of other nuclear utilities or contractors based upon a comparability review. The plan also provides for a "grandfather clause" exclusion which allows recognition of a certain period of trustworthy service with the utility or contractor, as being equivalent to the overall employee screening program. The staff has reviewed the applicant's screening program against the accepted industry standards (ANSI N18.17 1973) and has determined that the Washington Public Power Supply System's program is acceptable.

14. INITIAL TEST PROGRAM

The initial test program encompasses the scope of events that commences with completion of system construction and terminates with the completion of power ascension testing. The initial test program is conducted in three separate and sequential subprograms: system lineup testing, the preoperational test program and the startup test program. At the conclusion of these subprograms, a unit is ready for normal power operation. The system lineup testing, preoperational, and startup test subprograms are accomplished in the following five distinct and sequential major phases:

- Phase I - System Lineup Testing
- Phase II - Preoperational Testing
- Phase III - Fuel Loading and Post Core Hot Functional Testing
- Phase IV - Initial Criticality and Low Power Physics Testing
- Phase V - Power Ascension Testing

System lineup testing includes cleaning and flushing of piping systems and equipment; electrical equipment checks such as insulation resistance measurements, phase verification, continuity checks, voltage measurements, grounding checks, circuit breaker operation and relay operation; initial operation of motors and valves; calibration of instruments; and adjustments of relief and safety valves.

The preoperational test program commences with system/component turnover from the construction activity to the operations activity, and terminates with the beginning of unit fuel loading. These tests demonstrate, to the extent practicable, the capability of structures, systems, and components to meet performance requirements, and to satisfy design requirements. To the extent practicable, the objectives of the preoperational test program are to:

(1) Document the performance and operability of equipment and systems

(2) Provide base-line test and operating data on equipment and systems for future reference

(3) Operate new equipment for a sufficient time period so that design, manufacturing, and installation defects can be detected and corrected

(4) Ensure that plant systems operate together on an integrated basis

(5) Familiarize the plant operating, technical, and maintenance personnel with the plant operation

(6) Confirm the adequacy of plant operating and emergency operating procedures.

The startup test program commences with fuel loading, and terminates with the completion of power ascension testing. These tests confirm the design bases and demonstrate, to the extent practicable, that the plant operates and responds to transients as designed. Startup testing is sequenced to ensure that the safety of the plant is not dependent upon the performance of untested structures, systems, or components.

The objectives of the startup test program are to:

(1) Accomplish a controlled, orderly, and safe initial core loading

(2) Accomplish a controlled, orderly, and safe initial criticality

(3) Conduct low power testing sufficient to ensure that design parameters are satisfied, and that safety analysis assumptions are correct or conservative

(4) Perform a controlled, orderly, and safe power ascension with requisite testing, terminating at plant rated conditions.

The applicant has made extensive reference to the Combustion Engineering Standard Safety Analysis Report (CESSAR). Preoperational and startup test abstracts pertinent to Combustion Engineering's "System 80" Nuclear Steam Supply System

(NSSS) are contained in CESSAR, as are other NSSS-related topics (e.g., system descriptions, accident analysis, and Standard Technical Specifications). CESSAR was evaluated in NUREG-0852.

Our review of Chapter 14 of the applicant's FSAR concentrated on the administration of the test program and the completeness of the preoperational and startup tests. The Safety Evaluation Report which was issued at the completion of the Construction Permit review was reexamined to determine the principal design criteria for the plant and to identify any specific concerns or unique design features that would warrant special test consideration. Chapters 1 through 12 of the FSAR were reviewed for familiarization with the facility design and nomenclature. Chapter 15 was reviewed to identify assumptions pertaining to performance characteristics that should be verified by testing and to identify all structures, systems, components and design features that were assumed to function (either explicitly or implicitly) in the accident analyses. Licensee Event Report Summaries for operating reactors of similar design were reviewed to identify potentially serious events and chronic or generic problems that might warrant special test consideration. Standard Technical Specifications were reviewed to identify all structures, systems, and components that would be relied upon for establishing conformance with safety limits or limiting conditions for operation. Post-TMI related testing requirements were reviewed in conformance with: NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident;" NUREG-0694, "TMI-Related Requirements for New Operating Licenses;" and NUREG-0737, "Clarification of TMI Action Plan Requirements." And finally, Startup Test Reports for other Combustion Engineering reactor plants were reviewed to identify problem areas that should be emphasized in the initial test program.

In determining the acceptability of the applicant's test program, the criteria of the Standard Review Plan, NUREG-0800, Section 14.2 were used. Our review included verification of the following features of the initial test program:

(1) The applicant plans to develop test procedures using input from the NSSS vendor, the architect-engineer, the applicant's engineering staff, and other equipment suppliers and contractors. Operating experiences at similar plants are being factored into the development of the test procedures.

(2) The applicant plans to conduct tests using approved test procedures. Administrative controls cover (a) the completion of test prerequisites, (b) the completion of necessary data sheets and other documentation, and (c) the review and approval of modifications to test procedures. The applicant has stated that administrative procedures also cover implementation of modifications or repair requirements identified as being required by the tests and any necessary retesting.

(3) The applicant plans to review the results of each test for technical adequacy and completeness by review groups that include the NSSS vendor and architect-engineer as appropriate. Preoperational test results will be reviewed prior to fuel loading and the startup from each test condition or power level will be reviewed prior to proceeding to the next test condition or power level.

(4) The applicant plans to use normal plant operating and emergency operating procedures in performing the initial test program, thereby verifying the correctness of the procedures to the extent practicable.

(5) The schedule for conducting the initial test program allows adequate time to conduct all preoperational and startup tests. Preoperational test procedures will be available for NRC Regional Administrator review at least 60 days prior to scheduled implementation. Startup test procedures will be available for review not less than 60 days prior to the scheduled fuel loading date.

(6) A description of each test is presented in Chapter 14 of the FSAR. We verified that there are test descriptions for those structures, systems, components, and design features that: (a) will be used for shutdown and cooldown of the reactor under normal, transient, and accident conditions and for maintaining the reactor in a safe condition for an extended shutdown period; (b) will be used for establishing conformance with safety limits or limiting conditions for operation that will be included in the facility technical specifications; (c) are classified as engineered safety features or will be relied on to support or ensure the operations of engineered safety features within design limits; (d) are assumed to function

or for which credit is taken in the accident analysis of the facility, as described in the FSAR; or (e) will be used to process, store, control, or limit the release of radioactive materials.

(7) The test objectives, prerequisites, test methods, and acceptance criteria for each test description are in sufficient detail to establish that the functional adequacy of the structures, systems, components, and design features will be demonstrated.

(8) The test program conforms with Regulatory Guide 1.68, "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors," Revision 2. Exceptions have been identified, and technical justification provided.

The applicant made a number of changes to the initial test program because of the staff's comments. Examples of these changes are:

(1) Testing was added to more accurately determine the Reactor Protection and Engieered Safety Feature System trip response times.

(2) Testing was added to verify steam generator safety relief valve operability.

(3) Testing was modified to verify that no blockages exist in the containment spray nozzle flow path.

(4) Natural circulation tests were expanded and will be repeated for training purposes to comply with TMI-2 Action Plan Item I.G.1 for low power training and testing.

(5) Testing was added to provide improved assurance of proper auxiliary feed-water system performance.

(6) Testing was added to verify the heat removal capacity of ECCS coolers during post-accident conditions.

(7) Testing was added to verify the capabilities of the emergency lighting system.

(8) Testing was added for the loose parts monitoring system.

(9) Testing was added to demonstrate operability and perform leak tests of sectionalizing devices in the spent fuel storage pool.

(10) Testing was added to demonstrate that, for hot containment penetrations where coolers are not used, concrete temperatures do not exceed design limits.

The following items remain unresolved:

Question Number	Request for Additional Information (RAI) Question
640.01	Tests which may be waived or rescheduled, conditional on the results of earlier tests, should be identified.
640.03	FSAR Section 14.2.11 should state that fuel loading and startup test procedures should be available at least 60 days prior to fuel loading.
640.04	Individual test descriptions should be expanded to indicate the sources of acceptance criteria.
640.08	FSAR Subsection 14.2.12.2.23 (ECCS Area Ventilation) should state that data will be extrapolated to verify design heat removal capability as stated in response to this item.
640.09	(1) FSAR Subsection 14.2.7 should state the level of conformance with the testing requirements of Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release." (2) FSAR Subsection 14.2.12.2.48 (Control Room Leak Rate Test) should state conformance with Regulatory Guide 1.95 acceptance criteria, or the testing described in this test abstract should be referenced by, or

incorporated into, FSAR Subsection 14.2.12.16 (Control Room HVAC).

640.11 (1) FSAR Subsection 14.2.12.2.8 (Diesel Generators and Auxiliary Systems) should be modified to demonstrate that testing is conducted in accordance with Regulatory Guide 1.108, Positions C.2.a (1-9) and C.2.b.

640.12 (1) The Essential 125V DC Preoperational Test description should be clarified to ensure that testing will verify the ability of equipment to start and operate with batteries at minimum voltage.

(2) The Standby Transformers Preoperational Test description should be revised to demonstrate proper operation of transformer cooling under rated load, or how part-load test data will be used to verify full-load capability.

640.14 (1) Provide the following test abstracts, modify existing test abstracts, or provide justification for exception with the following positions of Regulatory Guide 1.68, Appendix A:

1.b (1)	Service Water System
1.n (16)	Heating systems for the refueling water storage tank
5.n	Obtain baseline data for the loose parts monitoring system

14A FSAR Subsection 14.2.7 (Conformance of Test Programs with Regulatory Guides) states that the WNP-3 test program "generally conforms to the requirements of"

the listed regulatory guides. Either state that the WNP-3 test program conforms to the applicable guides, or provide justification for any exceptions taken.

14B FSAR Subsections 14.2.12.5 and 14.2.12.8 state that additional tests will be provided at a later date. Either provide these additional tests, or modify these sections accordingly.

Based on our review of the FSAR Section 14.2 as amended through Amendment 5, we have concluded that (with the exception of the items described above) the initial plant test program is acceptable and meets the requirements of 10 CFR Part 50, §50.34(b)(6)(iii) that requires inclusion of plans for preoperational testing and initial operations in the FSAR and 10 CFR Part 50, Appendix B, Section XI that requires a test program to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents. We have further concluded that if acceptable responses to the above items are made, then the initial test program described in the application will meet the acceptance criteria of Section 14.2 of the Standard Review Plan, NUREG-0800, and the successful completion of the test program will demonstrate the functional adequacy of plant structures, systems, and components.

This review and evaluation was performed with the assistance of Battelle Pacific Northwest Laboratories' personnel.

Future changes to this approved test program should be submitted to the staff with justification for the changes for review by the staff. If a change relates to individual test descriptions, the justification should consider the safety-related categories enumerated above pertaining to the criteria of Standard Review Plan 14.2.

15 ACCIDENT ANALYSES

15.4 Reactivity and Power Distribution Anomalies

The input for the following items are identical to the corresponding items in the CESSAR FSAR:

15.4.1 CEA Withdrawal from Low Power
15.4.2 CEA Withdrawal from Full Power
15.4.3 CEA Misoperation Events
15.4.7 Fuel Misloading Event
15.4.8 CEA Ejection Event

15.x Radiological Consequences of Design Basis Accident

The WNP-3 FSAR references the CESSAR FSAR in areas pertaining to the NSSS. Consequently, the staff has not analyzed certain accidents but has determined whether or not the interface requirements for CESSAR as specified below have been satisfied for WNP-3. The accidents which have not been specifically analyzed include

- Steam Line Break Accidents
- Reactor Coolant Pump Locked Rotor
- Steam Generator Tube Rupture
- Rod Ejection Accident and
- Small Line Break Accident

For those accidents stated above, the CESSAR SER has established the following site-related interface requirements for CESSAR reference plants on the basis of the analyses described in the CESSAR SER and its supplements:

(1) Coolant activity: 0.1 uCi/gm dose equivalent I-131 (DEI-131) for secondary coolant maximum fission product concentration; 1.0 uCi/gm DEI-131 for

primary coolant maximum equilibrium fission product concentration and a spike limit of 60 uCi/gm DEI-131.

(2) Steam generator tube leakage: 1.0 gallon per minute (gpm) primary to secondary.

(3) Maximum condensate storage tank iodine concentration: 0.1 uCi/lb_m DEI-131.

(4) Atmospheric dispersion factors (X/Q, sec/m^3) equal to or less than: 2.5E-3* (2.0E-3 for radiological dose evaluation of Reactor Coolant Pump Locked Rotor Accident) for 0-2 hour post accident X/Q at the Exclusion Area Boundary; 1.0E-4 for the 0-8 hour X/Q at the outer boundary of the Low Population Zone; 2.8E-5 for the 1-4 day X/Q at the outer boundary of the Low Population Zone; and 8.3E-6 for 4-30 day X/Q at the outer boundary of the Low Population Zone.

The atmospheric dispersion parameters are given in Section 2.3 of this report. Based on these values, the staff concludes that the above meteorological parameters envelope those of the WNP-3 site and therefore, the accident interface requirements on the meteorological parameters are met.

The staff will also ensure that the technical specification interface requirements related to coolant activity, steam generator tube leakage and condensate storage tank iodine concentration levels identified above will be included in the WNP-3 plant technical specifications.

15.x.1 Loss-of-Coolant Accident

The applicant has selected and analyzed a hypothetical design basis LOCA and has shown that the distances to the Exclusion Area and to the Low Population Zone Boundaries are sufficient to provide reasonable * 2.5E-3 = 2.5×10^{-3} assurance that the radiological consequences of such an accident are within the guidelines in 10 CFR 100.1 (a)(1) and (2). The analysis has included the following sources and radioactivity transport paths to the atmosphere:

(1) contribution from contaiment leakage;

(2) contribution from post-LOCA leakage from ESF systems outside containment; and

(3) contribution from containment purge.

The staff review has confirmed the applicant's findings based on the following:

(1) The applicant's provisions for a design of the containment system and the shield building ventilation system (SBVS) are acceptable as identified in Section 6 of this report.

(2) The staff's independent analysis of the radiological consequences of a hypothetical design-basis LOCA as described below.

(a) Containment Leakage Contribution

The WNP-3 plant includes a double containment design to collect and filter the leakage of fission products from a postulated design-basis LOCA. The double containment consists of a free-standing steel primary containment vessel surrounded by a reinforced concrete shield building. The reinforced concrete auxiliary building is also a part of the secondary containment barrier. Leakage from the primary containment which enters the secondary containment is treated by either the SBVS or the EA/FHBFES before its release to the atmosphere. Both of these systems are engineered safety features (see SER Sections 6.5.1 and 9.4 for a description of these systems). Another ESF in the primary containment is the containment spray system with a sodium hydroxide additive to enhance the removal of iodine in the containment following a LOCA (see SER Section 6.5.2 for a system description).

The principal assumptions employed in the staff analysis are listed in Table 15.2. The dose model and dose conversion paramaters are consistent with those given in Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors."

In the analysis of the design basis LOCA, the primary containment was assumed to leak at the design leak rate of 0.5 percent per day for the first 24 hours following the accident and at 0.25 percent per day thereafter. The applicant established to the satisfaction of the staff that the shield building annulus

and auxiliary building would not experience a period of post accident positive pressure and, hence, no exfiltration. Therefore, no leakage into the secondary containments was assumed to bypass an ESF gas treatment system throughout the course of the accident.

Forty percent of the leakage from the primary containment was assumed to enter the shield building annulus were it was postulated to go directly to the intake of the SBVS. Thirty-eight percent of the leakage from the primary containment was assumed to enter the auxiliary building where the staff also assumed that it went directly to the intakes of the EA/FHBFES. Because neither of these systems contain any recirculation capabilities, no credit for holdup or mixing in either the shield building annulus or the auxiliary building was assumed. The remaining twenty-two percent was assumed to bypass the secondary containment and go directly to the environment. In response to FSAR question 450.7, the applicant has committed to incorporate these leakage pathway fractions into the plant technical specification for containment leakage. Incorporation of these fractions into the technical specifications will assure that the containment is tested in a manner consistent with this LOCA evaluation.

(b) Post-LOCA Leakage from ESF Systems Outside Containment

As part of the LOCA, the staff also evaluated the consequences of leakage of containment sump water that is circulated by the ECCS after the postulated accident. During the recirculation mode, the sump water is circulated outside containment to the auxiliary building. If a significant leak should develop, a fraction of the iodine in the water could become airborne in the auxiliary building and exit to the atmosphere. Because the ECCS areas in the auxiliary building of WNP-3 are served by an ESF atmospheric cleanup system (the EA/FHBFES), doses from passive failures were not considered as specified in SRP Section 15.6.5, Appendix B. The staff evaluated the potential radiological consequences from normal ECCS component leakage by assuming a total

ECCS component leak rate of 1 gpm. The resulting radiological consequences were only 6 Rem to the thyroid at the Exclusion Area Boundary and 10 Rem to the thyroid at the Low Population Zone.

Staff Conclusions

The staff's calculated thyroid and whole body doses from the hypothetical LOCA are listed in Table 15.1. The staff concludes that the distances to the Exclusion Area and to the Low Population Zone Boundaries of the WNP-3 site, in conjunction with the ESF's of the WNP-3 design, are sufficient to provide reasonable assurance that the total radiological consequences of a postulated LOCA will be within the guidelines set forth in 10 CFR 100. The conclusion is based upon the staff review of the applicant's analyses, and on an independent analysis performed by the staff to verify that the total calculated doses are within the guidelines.

15.7 Radioactive Releases from a Subsystem or Component

15.7.4 Fuel Handling Accident

For the analysis of the fuel-handling accident in the fuel pool, the staff assumed that a fuel assembly was dropped in the fuel pool during refueling operations and that the equivalent of all of the fuel rods in the dropped assembly were damaged, thereby releasing the volative fission gases from the fuel rod gaps into the pool. The radiation monitors located above the pool would rapidly detect the release of activity from the pool and initiate the ESF-grade EA/FHBFES. The radioactive material that would escape from the fuel pool was assumed to be released to the environment as a "puff" release, with the iodine activity reduced by filtration through the EA/FHBFES. The radiological consequences following the postulated accident are shown in Table 15.1 and the assumptions and parameters used in the analysis are shown in Table 15.3. 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."

The staff has also evaluated the consequences of a fuel-handling accident inside primary containment. The applicant states that at all times during refueling operations the containment will be ventilated to the atmosphere through the reactor building purge system.

In response to FSAR question 450.10, the applicant has demonstrated that a release from a fuel handling accident can be rapidly detected and the containment isolated prior to the release of any material from the containment.

The staff finds that the applicant has provided an adequate system to mitigate the radiological consequences of a postulated fuel-handling accident inside the containment and in the spent fuel pool area. The staff concludes that the fuel-handling area ventilation system meets the relevant requirements of GDC 61. The staff further concludes that the distances to the Exclusion Area and the Low Population Zone Boundaries, in conjunction with the operation of the dose mitigating ESF and implementation of plant procedures, are sufficient to provide reasonable assurance that the calculated offsite radiological consequences of a postulated fuel-handling accident are well within the 10 CFR 100 exposure guidelines.

The staff's conclusion is based on (1) the staff's determination that the design features and plant procedures at WNP-3 meet the requirements of GDC 61 with respect to radioactivity control; (2) the staff review of the applicant's assumptions and analyses of the radiological consequences from the fuel-handling accident; (3) the staff's independent analyses using the assumptions in Regulatory Guide 1.25, Sections C.1.a through C.1.k, and (4) the WNP-3 Technical Specifications relating to fuel-handling and ventilation system operation.

Table 15.1 Radiological consequences of design-basis accidents

Postulated Accident	Exclusion Area Boundary (Rem)		Low Population Zone Boundary (Rem)	
	Thyroid	Whole Body	Thyroid	Whole Body
Loss of coolant accident (LOCA)				
Containment leakage				
0-2 hr	194	16	-	-
0-8 hr	-	-	76	4.9
8-24 hr	-	-	33	1.2
24-96 hr	-	-	27	0.3
96-720 hr	-	-	19	0.1
Total Containment leakage	194	16	155	6.4
ECCS component leakage	5.7	0.01	9.6	0.01
Total (LOCA)	200	16	160	6.4
Fuel-handling accident in fuel-handling area	1.4	0.5	0.2	0.1

Table 15.2 Assumptions used to calculate loss-of-coolant accident doses

Power level, Mwt	4100
Operating time, years	3
Fraction of core inventory available for leakage, %	
Iodines	25
Noble Gases	100
Initial iodine composition in containment, %	
Elemental	91.0
Organic	4.0
Particulate	5.0
Primary containment volumes, cu ft	
Sprayed	2.8E6*
Unsprayed	4.2E5
Primary containment leakage rate, %/day	
0-24 hours after accident	0.5
after 24 hours	0.25
Containment leakage pathways, %	
Reactor building	40
Auxiliary building	38
Bypass	22
Containment spray system effectiveness, inverse hr	
Elemental iodine removal coefficient	10
Organic iodine removal coefficient	0
Particulate iodine removal coefficient	0.45
Atmospheric diffusion factors (second per cubic meter)	
0-2 hour Exclusion Area Boundary	4.1E-4
0-8 hour Low Population Zone Boundary	6.0E-5
8-24 hour Low Population Zone Boundary	4.0E-5
1-4 day Low Population Zone Boundary	1.6E-5
4-30 day Low Population Zone Boundary	4.3E-6

* 2.8E6 = 2.8×10^6 = 2,800,000

Table 15.3 Assumptions used for estimating the radiological consequences following a postulated fuel handling accident

Parameter and unit of measure	Quantity
Power level, Mwt	4100
Number of fuel assemblies damaged	1
Total number of fuel assemblies in core	217
Peaking factor of damaged rod	1.65
Shutdown time, hr	72.0
Inventory released from damaged rods (iodines and noble gases), %	10
Pool decontamination factors	
Iodine	100
Noble gases	1
Iodine fractions released from pool, %	
Elemental	75
Organic	25
Iodine removal efficiencies, %	
Elemental	99
Organic	99
X/Q values, sec/cubic meter	
0-2 hour EAB	4.1E-4
0-8 hour LPZ	6.0E-5

18 HUMAN FACTORS ENGINEERING

Position

Licensees and applicants for operating licenses shall conduct a Detailed Control Room Design Review (DCRDR). The objective is to "improve the ability of nuclear power plant control room operators to prevent accidents or cope with accidents if they occur by improving the information provided to them" (NUREG-0660, Item I.D). The need to conduct a DCRDR was confirmed in NUREG-0737 and Supplement 1 to NUREG-0737. DCRDR requirements in Supplement 1 to NUREG-0737 replaced those in earlier documents. Supplement 1 to NUREG-0737 requires each applicant or licensee to conduct a DCRDR on a schedule negotiated with the Nuclear Regulatory Commission (NRC).

NUREG-0700 describes four phases of the DCRDR and provides applicants and licensees with guidelines for its conduct.

The phases are

(1) planning
(2) review
(3) assessment and implementation
(4) reporting

Criteria for evaluating each phase are contained in Appendix A to the Standard Review Plan (SRP), Section 18.1 of NUREG-0800.

A Program Plan is to be submitted within two months of the start of the DCRDR. Consistent with the requirements of Supplement 1 to NUREG-0737, the Program Plan shall describe how the following elements of the DCRDR will be accomplished:

(1) Establishment of a qualified multidisciplinary review team.

(2) Function and task analyses to identify control room operator tasks and information and control requirements during emergency operations.

(3) A comparison of display and control requirements with a control room inventory.

(4) A control room survey to identify deviations from accepted human factors principles.

(5) Assessment of human engineering discrepancies (HEDs) to determine which HEDs are significant and should be corrected.

(6) Selection of design improvements.

(7) Verification that selected design improvements will provide the necessary correction.

(8) Verification that improvements will not introduce new HEDs.

(9) Coordination of control room improvements with changes from other programs such as SPDS, operator training, Reg. Guide 1.97 instrumentation and upgraded emergency operating procedures.

A Summary Report is to be submitted at the end of the DCRDR. As a minimum it shall

(1) outline proposed control room changes

(2) outline proposed schedules for implementation

(3) provide summary justification for HEDs with safety significance to be left uncorrected or partially corrected

The NRC will evaluate the organization, process, and results of the DCRDR. Evaluation will include review of required documentation (Program Plan and Summary Report) and may also include reviews of additional documentation,

briefings, discussions, and on-site audits. In-progress audits may be conducted after submission of the Program Plan but prior to submission of the Summary Report. Pre-implementation audits may be conducted after submission of the Summary Report. Evaluation will be in accordance with the requirements of Supplement 1 to NUREG-0737. Additional guidance for the evaluation is provided by NUREG-0700 and NUREG-0800, Appendix A to SRP Section 18.1. Results of the NRC evaluation of a DCRDR will be documented in a Safety Evaluation Report (SER) or SER Supplement.

NUREG-0737 Supplement 1 also requires that each operating reactor be provided with a safety parameter display system (SPDS) that is located convenient to the control room operators. This system will continuously display information from with the plant safety status can be readily and reliably assessed. The principal purpose and function of the SPDS is to aid the control room personnel during abnormal and emergency conditions in determining the safety status of the plant and in assessing whether abnormal conditions warrant corrective action by operators to avoid a degraded core. A written SPDS safety analysis shall be prepared describing the basis on which the selected parameters are sufficient to assess the safety status of each identified function for a wide range of events, which include symptoms of severe accidents. The applicant's safety analysis and SPDS implementation plant will be reviewed by the NRC staff to confirm: (1) the adequacy of the parameters selected to be displayed to detect critical safety functions; (2) that means are provided to assure that the data displayed are valid; (3) the adequacy of the design and installation of the system from a human factors perspective; and, (4) the adequacy of the verification and validation (V&V) program to assure a highly reliable to SPDS.

Status

As requested by Generic Letter 82-33, in its letter of April 14, 1983, the Washington Public Power Supply System (Supply System) submitted a proposed schedule for activities required by Supplement 1 to NUREG-0737. By letter dated July 12, 1983, the Supply System submitted its "Control Room Design Review Program Plan of WPPSS Nuclear Project 3." The staff did not complete its review of the Program Plan.

The staff review will be resumed when the project is reactivated.

REFERENCES

1. NUREG-0660, Volume 1, May 1980; NRC Action Plan Developed as a Result of the TMI-2 Accident.

2. NUREG-0737, November 1980; Clarification of TMI Action Plan Requirements.

3. Supplement 1 to NUREG-0737, December 1982; Requirements for Emergency Response Capability (Generic Letter 82-33).

4. NUREG-0700, September 1981; Guidelines for Control Room Design Reviews.

5. Letter to G. W. Knighton, NRC, from G. D. Bouchey, Washington Public Power Supply System, Subject: Nuclear Project 3 Response to Generic Letter No. 82-33 Requirements for Emergency Response Capability, dated April 14, 1983.

6. Letter to G. W. Knighton, NRC, from G. D. Bouchey, Washington Public Power Supply System, Subject: Nuclear Project 3 Control Room Design Review Program Plan, Dated July 12, 1983.