

Safety Evaluation Report

NUREG-0421

U. S. Nuclear
Regulatory Commission

related to construction of
**Davis-Besse Nuclear Power
Station, Units 2 and 3**

Office of Nuclear
Reactor Regulation

Docket Nos. 50-500
and 50-501

Toledo Edison Company, et al

July 1978

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JULY 6, 1978

SAFETY EVALUATION REPORT

BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

U. S. NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF

TOLEDO EDISON COMPANY,

CLEVELAND ELECTRIC ILLUMINATING COMPANY,

DUQUESNE LIGHT COMPANY,

OHIO EDISON COMPANY,

AND

PENNSYLVANIA POWER COMPANY

DAVIS-BESSE NUCLEAR POWER STATION UNITS 2 AND 3

DOCKET NOS. 50-500 AND 50-501

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1.0 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

The Toledo Edison Company, the Cleveland Electric Illuminating Company, Duquesne Light Company, Ohio Edison Company, and Pennsylvania Power Company (hereinafter referred to collectively as the applicant) filed with the Nuclear Regulatory Commission (NRC or Commission) an application docketed on August 9, 1974 for licenses to construct and operate its proposed Davis-Besse Nuclear Power Station Units 2 and 3 (Davis-Besse Units 2 and 3 or facility). The facility will be located on a 954-acre site on the southwestern shore of Lake Erie. The proposed site is located in Ottawa County, Ohio, approximately 20 miles southeast of Toledo. Davis-Besse Unit 1, a similar nuclear power plant, is also located on this site and is operational.

A Preliminary Safety Analysis Report was submitted with the application. The information in the Preliminary Safety Analysis Report has been revised by Revisions 1 through 20. At this time we have completed our review through Revision 20. The Preliminary Safety Analysis Report and these revisions are available for public inspection at the U.S. Nuclear Regulatory Commission Public Document Room, 1717 H Street, Washington, D.C. 20555, and at the Ida Rupp Public Library, Port Clinton, Ohio 43452.

This Safety Evaluation Report summarizes the results of the technical evaluation of the proposed Davis-Besse Units 2 and 3 performed by the Commission's staff. It delineates the scope of the technical matters considered in evaluating the radiological safety aspects of the facility. Aspects of the environmental impact considered in the review of the facility were discussed in the Commission's Final Environmental Statement, NUREG-75/083, issued in September 1975.

Upon favorable resolution of the outstanding issues discussed herein and summarized in Section 1.8 of this report, we will be able to conclude that Davis-Besse Units 2 and 3 can be constructed and operated as proposed without endangering the health and safety of the public. Our summary conclusions are presented in Section 21.0 of this report.

The review and evaluation of the preliminary design of this facility, as reported herein, is only the first stage of a continuing review by the Commission's staff of the design, construction, and operating features of Davis-Besse Units 2 and 3. Construction will be accomplished under the surveillance of the Commission's staff. Prior to a decision on issuance of an operating license, we 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, and under the continued surveillance of the Commission's staff.

1.2 General Plant Description

The Davis-Besse Nuclear Power Station will consist of three nuclear generating units. Unit 1, of design similar to that of Units 2 and 3, was licensed by the Commission for operation on April 22, 1977. The nuclear reactors proposed for Units 2 and 3 will be designed to produce a rated core thermal power of 2772 megawatts, the same as Unit 1.

Units 2 and 3 will be similar to Unit 1. The principal differences between Unit 1 and Units 2 and 3 involve the number and size of fuel rods in each fuel assembly, and the ultimate heat sink. These differences, and others resulting from the application of advances in technology and of changed regulatory requirements, are discussed in the appropriate section of this report.

Each unit will be separate and independent from the other two, except that the switchyard and the forebay and intake canal will serve all three units. A site layout is shown in Figure 1.1.

The major facility structures for each unit will be a free-standing steel containment structure surrounded by a reinforced concrete shield building, the auxiliary building, the turbine building, and a cooling tower.

The containment structure of each unit will house the nuclear steam supply system and most of the auxiliary components that contain reactor coolant. The containment structure and the shield structure will be provided with filtration, purge, and spray systems that will limit the external release of radioactive material that could be released from the reactor coolant system in the unlikely event of the design basis loss-of-coolant accident.

The auxiliary building for each unit will house the engineered safety features components, radioactive waste management systems, new and spent fuel handling and storage facilities, chemical and volume control systems, the control room including its ventilation system, and the two emergency diesel generators. The turbine building for each unit will house a General Electric turbine-generator.

An intake canal, separated from Lake Erie by a beach and beach front dike, will provide a long reservoir where water is stored for facility use. Pumps located in bays in the intake structure will supply all the water used by the facility.

The reactor of each unit will produce a core thermal power of 2,772 thermal megawatts. The nuclear steam supply system, to be supplied by the Babcock & Wilcox Company, will consist of a pressurized water reactor with a two-loop reactor coolant system. The steam generators will be located at a higher elevation in relation to

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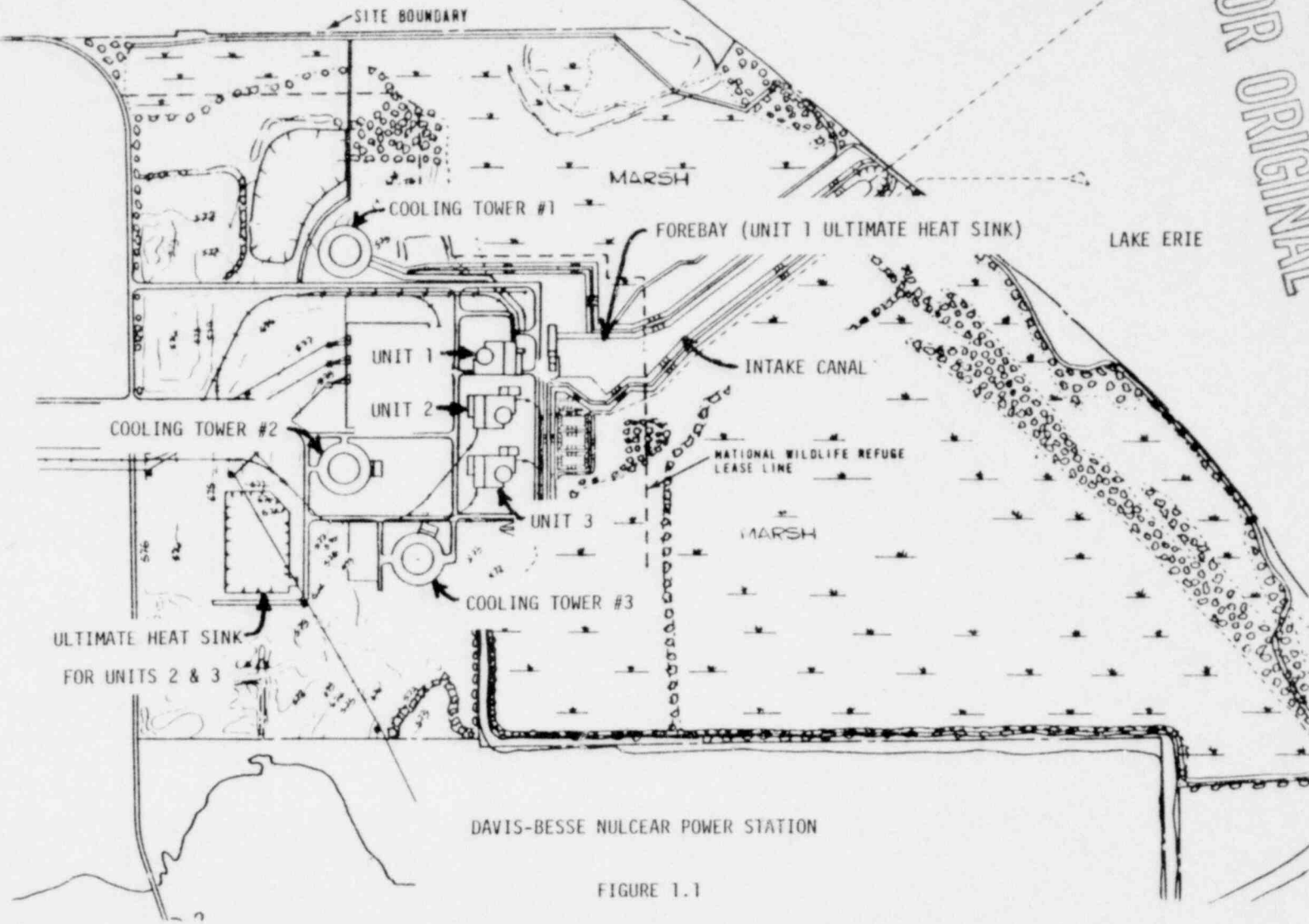


FIGURE 1.1

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the reactor vessel. The initial reactor fuel loading will use 177 fuel assemblies arranged in three regions, each containing a different enrichment of uranium-235. Water will serve as both moderator and coolant, and will be circulated through the reactor by four coolant pumps, two located in each loop. The reactor coolant water, heated by the reactor, will flow through two vertical once through steam generators where heat will be transferred to the secondary steam system. An electrically heated pressurizer will establish and maintain the reactor coolant pressure, and will provide a surge chamber and water reservoir to accommodate reactor coolant volume changes during operation.

The steam and power conversion system for the facility will be designed to remove heat energy from the nuclear steam supply system and to convert it into electrical energy by means of a steam turbine-generator. Waste heat rejected to the steam condensers will be discharged from the closed cycle circulating water system through the natural draft counter-flow cooling tower. Makeup, for water lost from the cooling tower by evaporation, drift and blowdown, will be pumped from Lake Erie by way of the intake canal structure.

The reactor will be regulated by control rod movement and by changing the boric acid concentration in the reactor coolant. A reactor protection system will be provided that automatically initiates appropriate corrective action whenever a safety-related condition monitored by the system approaches pre-established limits. The reactor protection system and an engineered safety features actuation system will act to shut down the reactor, close isolation valves, and initiate operation of the engineered safety features should any or all of these actions be required.

Each unit will include an emergency core cooling system. This system will consist of a core flooding system and both high and low pressure injection systems. It will include means for recirculating the borated water after the injection phase. Combinations of these systems will assure core cooling for the complete range of postulated reactor coolant pipe break sizes up to a double-ended break. Other engineered safety features will include the containment, containment isolation valves, containment spray system, containment cooling system, and combustible gas control system.

The station will be supplied with electrical power, from offsite power sources, by three independent transmission lines. In the event of an accident with a concurrent loss of offsite power, either of the two fast-starting diesel generators supplied for each unit and its associated bus will be capable of providing adequate power to safely shut down the unit or to operate the engineered safety features. A constant supply of direct current power to vital instruments and controls will be assured through the redundant 125 volt buses and their associated battery banks and battery chargers.

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1.3 Comparison with Similar Facility Designs

The principal features of the design of this facility are similar to those we have evaluated and approved previously for other nuclear facilities now under construction or in operation. These include Davis-Besse Unit 1 (Docket No. 50-346), the Oconee Nuclear Station, Units 1, 2 and 3 (Docket Nos. 50-269, 50-270, and 50-287), and the Rancho Seco Nuclear Station, Unit 1 (Docket No. 50-312). To the extent feasible, we have made appropriate use of our previous evaluations of these facilities in conducting our review of this facility. Where this has been done, the appropriate sections of this report identify these other facilities. Our safety evaluations for these other facilities have been published, and are available for public inspection at the Commission's Public Document Room at 1717 H Street, N.W., Washington, D. C.

1.4 Identification of Agents and Contractors

Davis-Besse Units 2 and 3 will be owned by the Toledo Edison Company, the Cleveland Electric Illuminating Company, Duquesne Light Company, Ohio Edison Company, and Pennsylvania Power Company, as tenants in common. Toledo Edison is responsible for the design, construction, and operation of the units and acts as agent in all matters.

In the design of the facility, Toledo Edison has retained Bechtel Associates Professional Corporation to provide architect-engineering services. United Engineers and Constructors has been retained, by letter of intent, to construct both units. Onsite materials testing services will be done by an independent contractor. Other consultants have been or will be retained by Toledo Edison to provide technical assistance for design, construction, and operation.

Babcock & Wilcox will furnish the nuclear steam supply systems and will be responsible for the detailed design of the systems and components.

1.5 Summary of Principal Review Matters

Our technical review and evaluation of the information submitted by the applicant considered the principal matters summarized below:

- (1) We evaluated the population density and land use characteristics of the site environs, and the physical characteristics of the site (including seismology, meteorology, geology, and hydrology), to establish that these characteristics have been determined adequately and have been given appropriate consideration in the plant design, and that the site characteristics are in accordance with the Commission's siting criteria (10 CFR Part 100), taking into consideration the design of the facility, including the engineered safety features provided.

- (2) We evaluated the design, fabrication, construction and testing criteria, and expected performance characteristics, of the plant structures, systems, and components important to safety, to determine that they are in accord with the Commission's General Design Criteria, Quality Assurance Criteria, Regulatory Guides, and other appropriate rules, codes and standards, and that any departures from these criteria, codes and standards have been identified and justified.
- (3) We evaluated the expected response of the facility to various anticipated operating transients and to a broad spectrum of postulated accidents. Based on this evaluation, we determined that the potential consequences of a few highly unlikely postulated accidents (design basis accidents) would exceed those of all other accidents considered. We performed conservative analyses of these design basis accidents to determine that the calculated potential offsite radiation doses that might result, in the very unlikely event of their occurrence, would not exceed the Commission's guidelines for site acceptability given in 10 CFR Part 100.
- (4) We evaluated the applicant's engineering and construction organizations, plans for the conduct of plant operations (including the organizational structure and the general qualifications of operating and technical support personnel), the plans for industrial security, and the planning for emergency actions to be taken in the unlikely event of an accident that might affect the general public, to determine that the applicant will be technically qualified to safely operate the facilities.
- (5) We evaluated the design of the systems to be provided for control of the radiological effluents from the facility to determine that these systems will be capable of controlling the release of radioactive wastes from the facility within the limits of the Commission's regulations (10 CFR Part 20), and that the equipment to be provided will be capable of being operated by the applicant in such a manner as to reduce radioactive releases to levels that are as low as reasonably achievable within the context of the Commission's regulations (10 CFR Part 50), and to meet the dose design objectives of Appendix I, 10 CFR Part 50.
- (6) We evaluated the applicant's quality assurance program for the design and construction of the facility, to assure that the program complies with the Commission's regulations (10 CFR Part 50) and that the applicant will have proper controls over the facility design and construction such that there will be a high degree of assurance that, when completed, the facility can be operated safely and reliably.
- (7) We are evaluating the financial data and information supplied by the applicant as required by the Commission's regulations (Section 50.33(f) of 10 CFR Part 50

and Appendix C to 10 CFR Part 50) to determine that the applicant is financially qualified to design and construct the proposed facility.

1.6 Facility Modifications as a Result of Staff Review

During the review of the application for Davis-Besse Units 2 and 3, numerous meetings were held with the applicant's representatives, its contractors, and consultants to discuss the proposed facility and the technical material submitted. A chronological listing of the meetings and other significant events is given in Appendix D to this report. During the course of the review, the applicant proposed, or we requested, a number of technical and administrative changes. These are described in various amendments to the original application, and are discussed in appropriate sections of this report.

1.7 Requirements for Further Technical Information

In Section 1.5 of the Preliminary Safety Analysis Report, the applicant identified research and development programs that are being conducted by Babcock & Wilcox. These programs are directed toward confirming the design adequacy of certain components in the nuclear steam supply system. The objectives and description of these programs are provided by reference to Babcock & Wilcox topical reports submitted for our review. The components and their respective program objectives that we have determined to be necessary to confirm design adequacy for the proposed facility, are listed in Table 1.1 of this report. The test programs listed in Table 1.1 have not all been completed by Babcock & Wilcox. These test programs are discussed further in Sections 4.2 and 4.4 of this report.

Based on our review of the verification programs, we have concluded that (1) the test program outlined in the Preliminary Safety Analysis Report and in referenced topical reports will provide the necessary information to confirm margins in the nuclear steam supply system design, and (2) in the event the programs provide unexpected results, appropriate restrictions on operation can be applied, or modifications in designs can be made, to protect the health and safety of the public.

We have also identified the need for certain design information that we believe should be reviewed before the applicant begins the construction of certain structures (in the event of a favorable decision on issuance of construction permits). Since the evaluation of this information may affect subsequent construction actions, we conclude that our review of these matters should be made during construction rather than later during the operating license stage of review. We required and the applicant agreed to provide, if construction permits are granted, a final report, for our review, on the conditions of the rock quarry slopes within 200 feet of the ultimate heat sink pump house (Section 2.6.5).

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TABLE 1.1

BABCOCK & WILCOX RESEARCH AND DEVELOPMENT PROGRAMS

<u>Test Components</u>	<u>Objectives</u>
<u>Mark C (17 x 17) Fuel Assemblies</u>	
Fuel Assembly Flow	To verify fuel rod and assembly vibration characteristics, resistance to fretting and wear, pressure drop characteristics, and hydraulic lift forces.
Assembly Mechanical	To verify vibration and damping characteristics, and static load deflections and stresses.
Components Mechanical	To verify spring characteristics of the spacer grid, ability of grids to withstand predicted seismic impact loads, ability of end fitting to withstand seismic and loss-of-coolant accident loads, mechanical adequacy of control rod guide tubes and of the assembly hold-down spring.
Critical Heat Flux	To verify predicted critical heat flux on full-length heated test bundles; to verify heat transfer characteristics during core reflood conditions; and to demonstrate that sufficient thermal margin exists in the 17 x 17 assembly to accommodate the effect of rod bow on departure from nucleate boiling.

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In addition, we have listed in Table 1.2 items discussed in this report that will require the submittal of additional technical information prior to approval of the final design. Also indicated in Table 1.2 are references to the sections in this report in which each of the items is discussed. We have determined that this information is of the type that, in accordance with the provisions of Section 50.35 of 10 CFR Part 50, can be left for later consideration and may be supplied in a Final Safety Analysis Report describing Davis-Besse Units 2 and 3.

1.8 Outstanding Items

We have identified nine outstanding items in our review, some of which will require that the applicant provide additional information to confirm that the proposed design will meet our requirements. We will require resolution of all of these items prior to a decision on issuance of a construction permits. These items are listed below, and are discussed further in the sections of this report indicated for each item.

- (1) We require that all valves in the main feedwater and main steam lines, required to function to mitigate the consequences of a main steam line break, be Quality Group B (Sections 3.2.2 and 6.2.1).
- (2) We require the applicant to commit that, should the results of the Electric Power Research Institute tests, or a generic resolution of missile penetration modeling be unavailable prior to construction of affected structures, the protection requirements against turbine missiles will be evaluated in terms of the current concrete penetration modeling involving the use of equations that are commonly referred to as the "modified NDRC formulas" and that are given as Equations 5 through 10 in "A Review of Procedures for the Analysis and Design of Concrete Structures to Resist Missile Impact Effects," by R. P. Kennedy (Section 3.5.1).
- (3) We require adequate protection of safety-related equipment against main steam line and main feedwater line breaks outside containment (Section 3.6.2).
- (4) We have not completed our review of the applicant's methods for analysis of loss-of-coolant accident loadings on reactor coolant system components and supports. We will report on this matter in a supplement to this Safety Evaluation Report (Section 3.9.2).
- (5) We have not completed our review of the capability to bring the plants to a cold shutdown condition in approximately 36 hours, using only safety-grade equipment, assuming a loss of onsite or offsite power and assuming a single failure. We will report on this matter in a supplement to this Safety Evaluation Report (Section 5.4.4).

TABLE 1

ITEMS REQUIRING ADDITIONAL TECHNICAL INFORMATION

<u>Item</u>	<u>Section(s) in this Report in Which Discussed</u>
(1) Qualification of balance-of-plant Class IE equipment	3.11
(2) Data from irradiation of two Mark C fuel assemblies in Oconee	4.2.3
(3) Data from Mark C Fuel mechanical and hydraulic tests	4.2.3
(4) Effects of rod bowing	4.2.3
(5) Response of Mark C fuel and reactor internals to seismic and loss-of-coolant accident asymmetric loads	4.2.3
(6) Fuel surveillance program on first two cores of Mark C fuel	4.2.4
(7) Confirmation of BAW-2 correlation	4.4.1
(8) Qualification of instrumentation isolation devices	7.1.2

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- (6) We have not completed our review of containment subcompartment pressurization. We will report on the resolution of this matter in a supplement to this Safety Evaluation Report (Section 6.2.1, page 6-5).
- (7) The applicant has proposed to calculate leakage through steam generator tube cracks rather than to leak test the associated penetrations of containment. We have not completed our review of this proposal, but will report on the resolution of this matter in a supplement to this Safety Evaluation Report (Section 6.2.1, page 6-6).
- (8) We require the applicant to commit to inclusion of hydrogen recombiners as the primary means for control of combustible gases following a loss-of-coolant accident (Section 6.2.4).
- (9) Documentation must be provided, for inspection by the Office of Inspection and Enforcement concerning (1) QA program implementation activities included in the constructor's (UE&C) scope of work, and (2) administrative and construction procedures for site engineering design control, site-initiated procurement control, and construction (Section 17.6).

1.9 Generic Items

The Advisory Committee on Reactor Safeguards periodically issues a report listing various generic matters applicable to light water reactors. Our discussion of these matters is provided in Appendix C to this report. Appendix C includes references to those sections of this report in which there are more specific discussions on particular generic matters related to the proposed facility.

2.0 SITE CHARACTERISTICS

2.1 Geography and Demography

2.1.1 Site Description

The site of Davis-Besse Units 2 and 3 is located adjacent to Unit 1 on a 954-acre tract of land on the southwestern shore of Lake Erie in Ottawa County, Ohio. The site is approximately 20 miles east-southeast from Toledo, Ohio (population 383,818) and 20 miles west-northwest from Sandusky, Ohio (population 32,674). The geographic location is shown in Figure 2.1. The principal features of the Davis-Besse site are shown in Figure 2.2.

The topography of the site consists of approximately 354 acres of flat shoreland and 600 acres of marshland. The latter is leased to the U.S. Government as a national wildlife refuge.

2.1.2 Exclusion Area and Control

The site boundary shown in Figure 2.2 forms the exclusion area, which has a minimum exclusion distance of 635 meters from the edge of the Unit 3 containment structure to the closest site boundary. Toledo Edison Company and The Cleveland Electric Illuminating Company, as tenants in common, own all of the exclusion area including the section leased to the U.S. Government. Toledo Edison has the authority to exclude people from the exclusion area if station conditions require such exclusion. Maintenance of the dikes surrounding the wildlife areas is the responsibility of Toledo Edison. Water level control of the marsh areas is the responsibility of the U.S. Bureau of Sport Fisheries and Wildlife. There is expected to be only limited physical activity in or public access to these areas.

The exclusion area is not traversed by any public roads, waterways or railroads. State Route 2, adjacent to the site, and a railroad spur of the Norfolk and Western Railroad, provide access to the Davis-Besse station site.

2.1.3 Population Distribution

The 1970 census population and the population projected for the years 1980 and 2020, in the area surrounding the site, are shown in Table 2.1.

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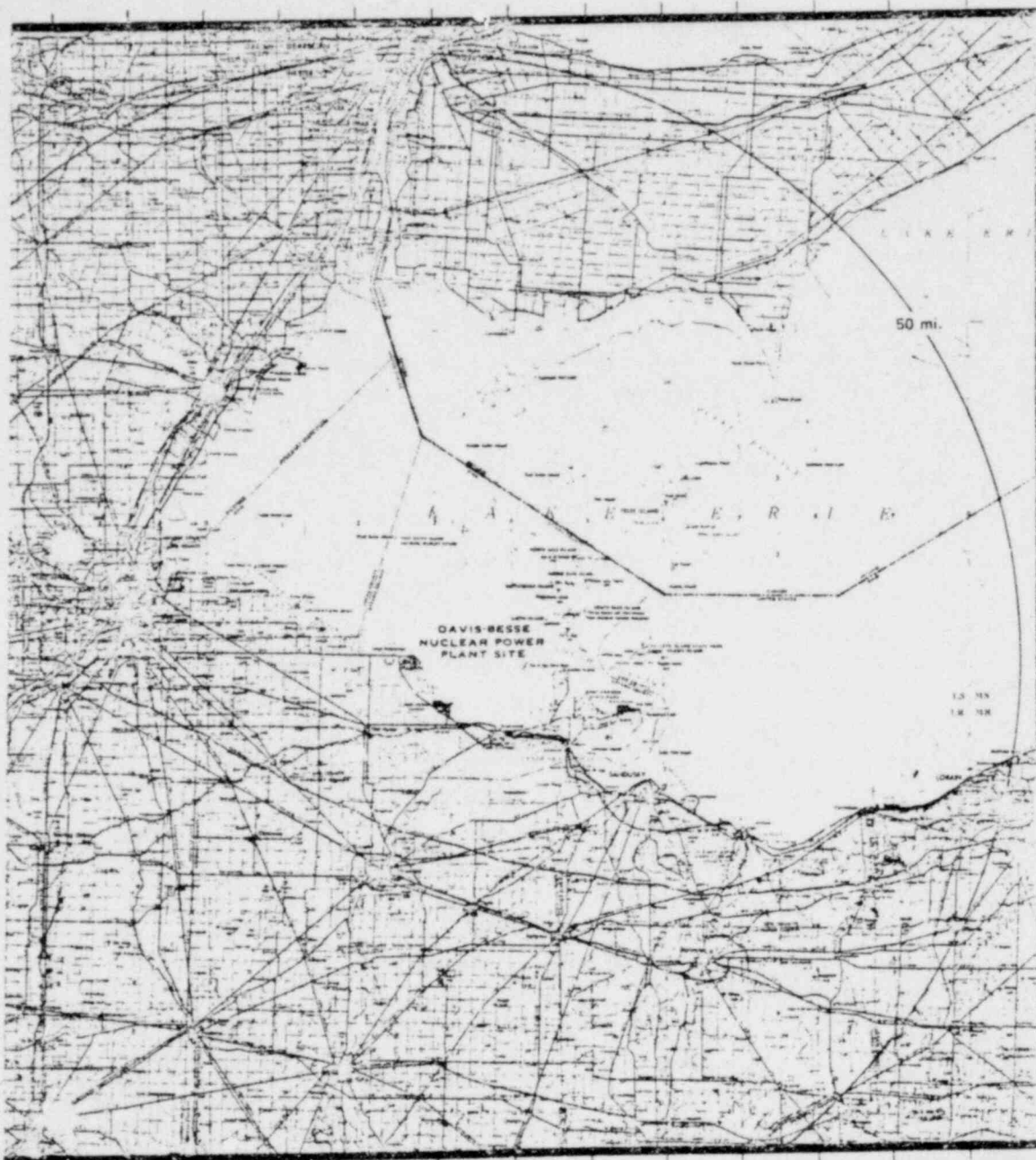


Figure 2.1 Geographic Site Location

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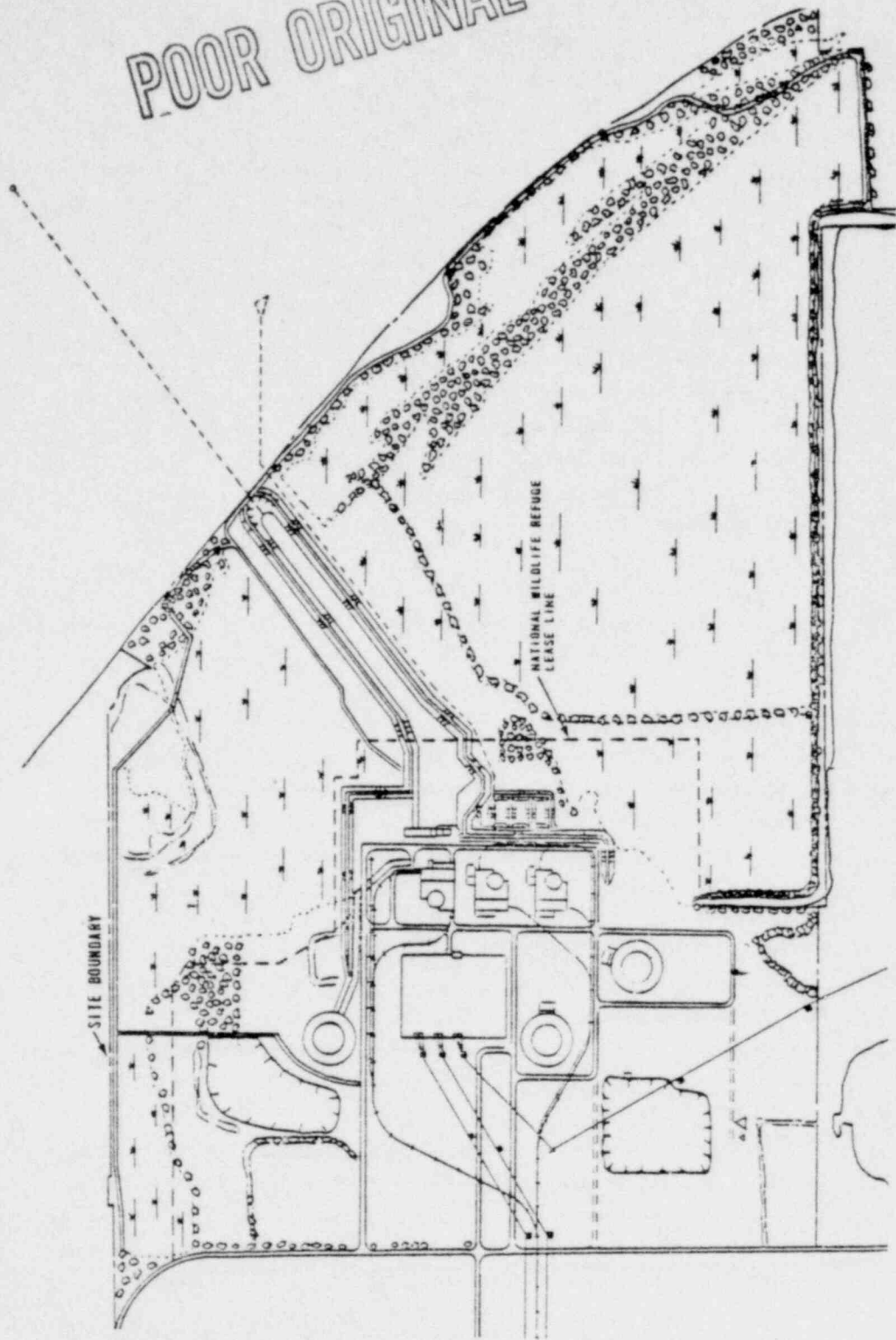


Figure 2.2 Site Plan

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TABLE 2.1

1970 CENSUS AND PROJECTED POPULATIONS

<u>Radius (Miles)</u>	<u>1970</u>	<u>1980</u>	<u>2020</u>
0-5	1,974	1,571	1,936
0-10	13,443	17,740	21,800
0-30	688,157	747,284	1,018,500
0-50	2,034,827	2,244,772	3,107,000

The 1980 and 2020 projected year cumulative populations as a function of distance are shown in Figure 2.3. For reference, the cumulative population corresponding to 500 people per square mile, which is a population density that we use to characterize a moderately populated area, is also shown. The data shown in Figure 2.3 illustrate that the projected population, at distances out to 30 miles from the site, is less than 500 people per square mile at the time of projected plant startup and at projected end of plant life.

We made an independent estimate of the 1970 population within a 50-mile radius of the Davis-Besse site, based upon the Bureau of the Census data, and determined that our value of 2,066,703 people agreed reasonably well with the applicant's value of 2,034,827.

The applicant's projected growth rate to the year 2020, for the area within a 50-mile radius of the site, was compared with the population projections of the Bureau of Economic Analysis for Economic Areas 409, 410 and 411. This comparison showed a projected growth of about 19 percent per decade as compared with about 14 percent per decade estimated by the applicant.

The applicant has indicated the seasonal increases within 5-mile and 10-mile radii of the plant to be 2,191 and 3,921 people, respectively. This increase during the summer months is in summer residences and cottages located mainly along the Lake Erie shoreline. The employment at Erie Industrial Park, located four miles southeast of the site, is approximately 900.

At the present time, the land surrounding the site is of a rural nature. Wildlife refuges cover a large section of the shoreline on both sides of the plant site. The closest industrial areas are four miles southeast of the site. Water sport activities in the area include pleasure boating, sport fishing, duck hunting and swimming. Lake Erie is used for commercial fishing and shipping.

According to the applicant, the Crane Creek State Park (shown in Figure 2.4), located three miles from the site, has an average daily summertime attendance of 2,500 and a peak attendance of 5,000. The Magee Marsh Wildlife Area, located 2-1/2 miles from the site, has an annual attendance of 48,000 with a peak daily attendance of 1,500.

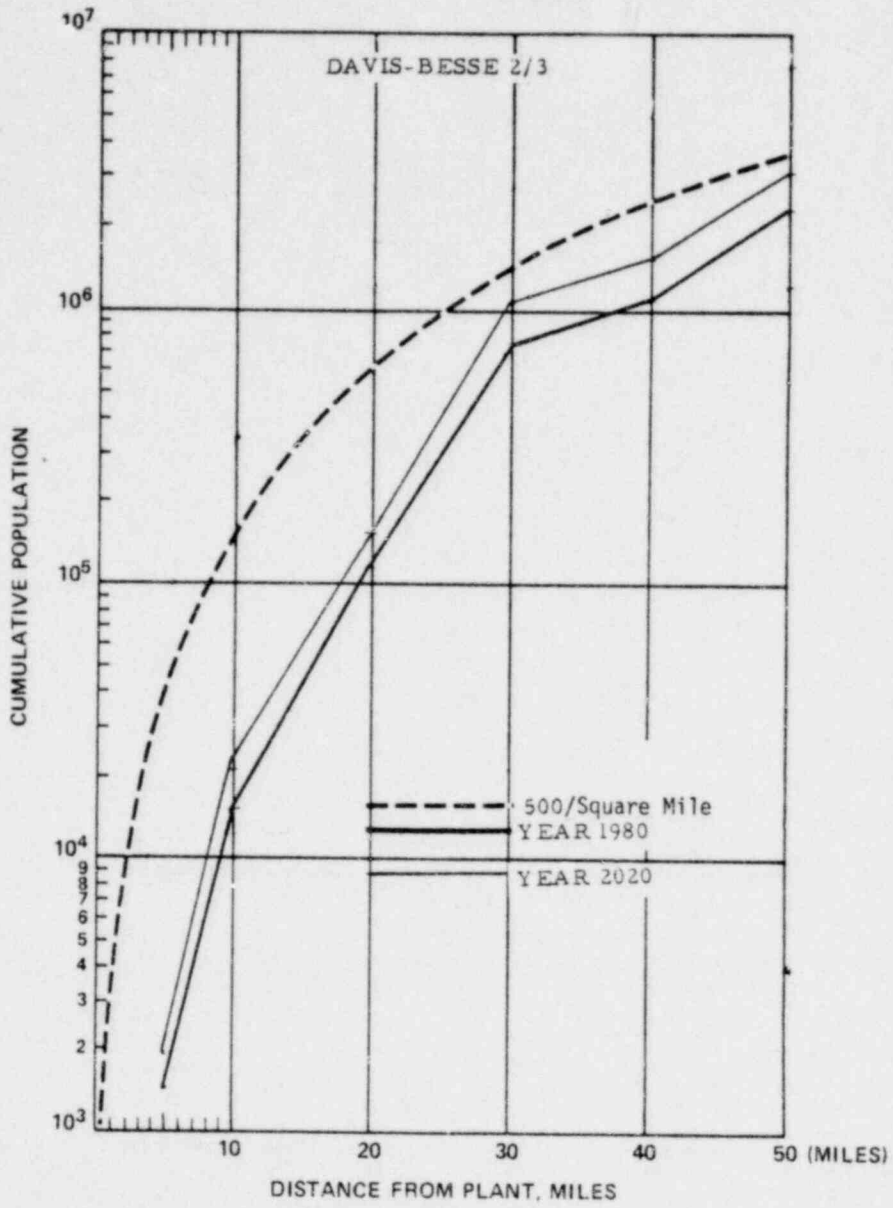
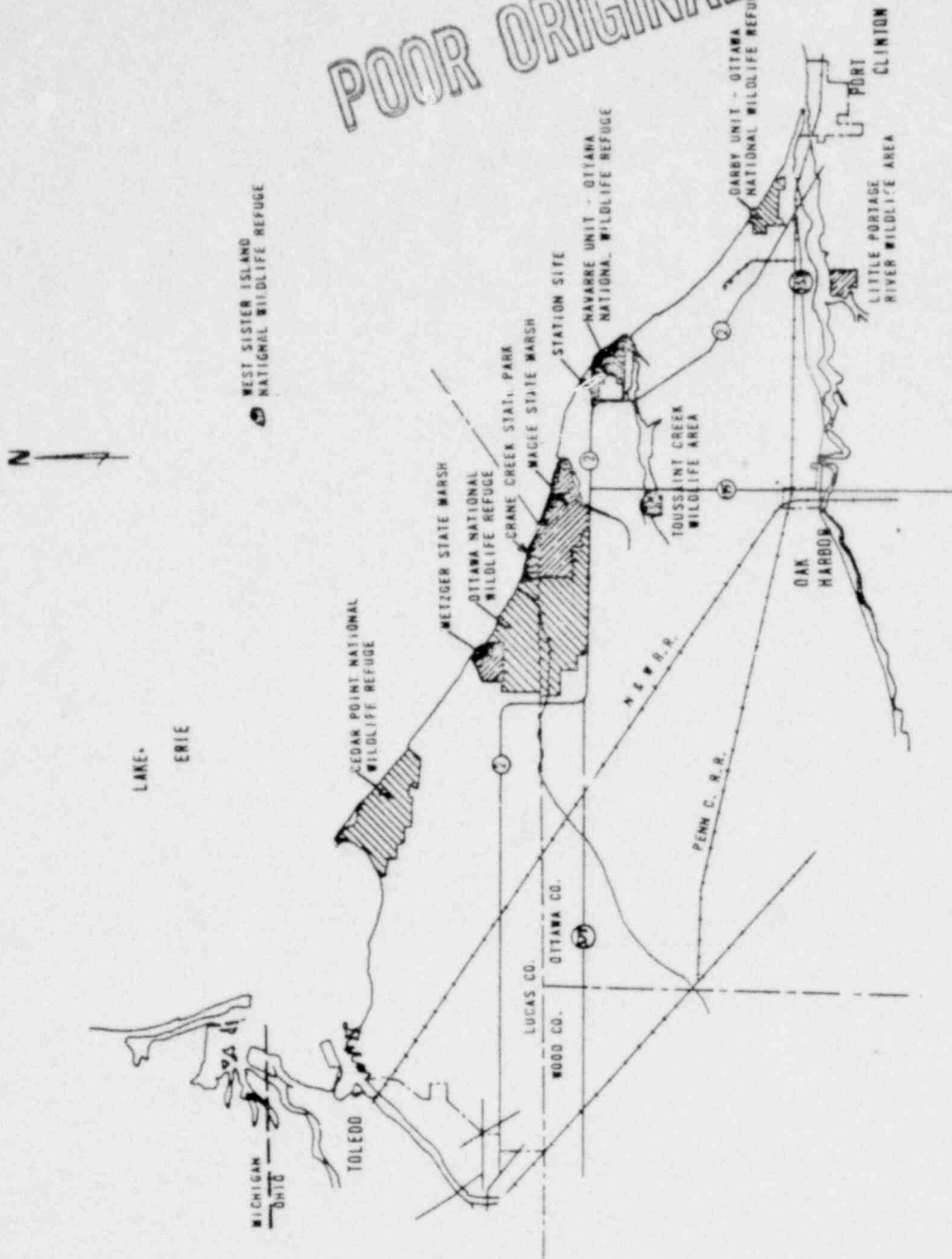


FIGURE 2.3 Cumulative Population Distribution

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DAVIS-BESSE NUCLEAR POWER STATION
UNITS NO. 1, 2, & 3
FEDERAL AND STATE WILDLIFE AREAS
FIGURE 2.4

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We conclude that the transient population within five miles of the site is low and, when added to the resident population, will give an acceptably low population density of less than 128 people per square mile.

The applicant has selected a low population zone radius of two miles. The 1970 Census report for this area showed a population of 502 people. The applicant originally designated the nearest population centers, as defined in 10 CFR Part 100, to be the cities of Toledo, Ohio, and Sandusky, Ohio, both located approximately 20 miles from the plant site. The applicant revised the Preliminary Safety Analysis Report in Revision No. 14, and has designated the city of Fremont, Ohio, located 17 miles south of the plant, as the nearest population center. Although Fremont had a 1970 Census population of 18,490 people, the population is projected to exceed 25,000 people by the year 2020. We concur with the applicant's revised designation of Fremont as the nearest population center.

Since the revised population center distance is over one and one-third times the low population zone radius of two miles, the applicant's selection of the low population zone is in compliance with 10 CFR Part 100.

2.1.4 Conclusion

On the basis of (1) the applicant's specified population center distance, minimum exclusion area, and low population zone, (2) our analysis of the onsite meteorological data from which atmospheric dilution factors were calculated (see Section 2.3 of this report), and (3) our calculated potential radiological dose consequences of design basis accidents discussed in Section 15.0 of this report, we have concluded that the exclusion area, low population zone, and population center distance meet the guidelines of 10 CFR Part 100 and are, therefore, acceptable.

2.2 Nearby Industrial, Transportation and Military Facilities

Industrial activities near the Davis-Besse site are concentrated in the Erie Industrial Park, about five miles southeast of the site. This is the closest industrial area to the site. The Erie Industrial Park has 20 firms of miscellaneous types, with a total employment of approximately 900 people.

One of the firms located in the Erie Industrial Park is Cadillac Gage Company. This firm operates a facility at this location for the test firing of ordnance. The maximum amount of ammunition stored is equivalent to 10,000 pounds of high explosives. We have concluded that the ammunition, stored in bunkers, does not pose a threat to the safety of the proposed Davis-Besse facilities because of the distance of the Erie Industrial Park from the site.

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The closest military facility is Camp Perry, which abuts the eastern boundary of the Erie Industrial Park. Camp Perry is used for Ohio National Guard and U.S. Army training, and is the site of the annual National Rifle Matches during July and August. There are about 70 permanent Army and National Guard personnel, with short-term population increases of about 500 in the summer and on weekends. Weapons firing is also conducted at Camp Perry.

The firing of ordnance from Camp Perry and Cadillac Gage Company is directed toward impact areas located in Lake Erie. These impact areas lie within areas that have been designated by the U.S. Army Corps of Engineers as restricted areas. The nearest boundary of these restricted areas is approximately 1.5 miles to the east of the Davis-Besse plant.

The use of these restricted areas was evaluated at the time of the Davis-Besse Unit 1 construction permit review to determine the effect of such use on the Davis-Besse facility. The results of this evaluation appear in Section 3.6 of our Safety Evaluation Report for Davis-Besse Unit 1, November 2, 1970 (Docket No. 50-346). The following updates the information presented in that Safety Evaluation Report. This update information has also been published in our Report on the Site Suitability for Units 2 and 3, dated November 7, 1975.

Since publication of the Unit 1 Safety Evaluation Report, use of Restricted Area III has been discontinued, and the area is no longer designated as a restricted area. The only presently designated restricted areas are Areas I and II, shown in Figure 2.5. In addition, Thompson-Ramo-Woolridge, Inc., is no longer involved with testing at the Erie Industrial Park.

Cadillac Gage Company directs the firing of its weapons toward Restricted Area II. The closest boundary of Area II is 1.5 miles east of the plant, but the firing fan is limited by gun stakes to five degrees east and west of north. The closest impact point of ordnance fired from Cadillac Gage Company is, therefore, about two miles from the Davis-Besse station site.

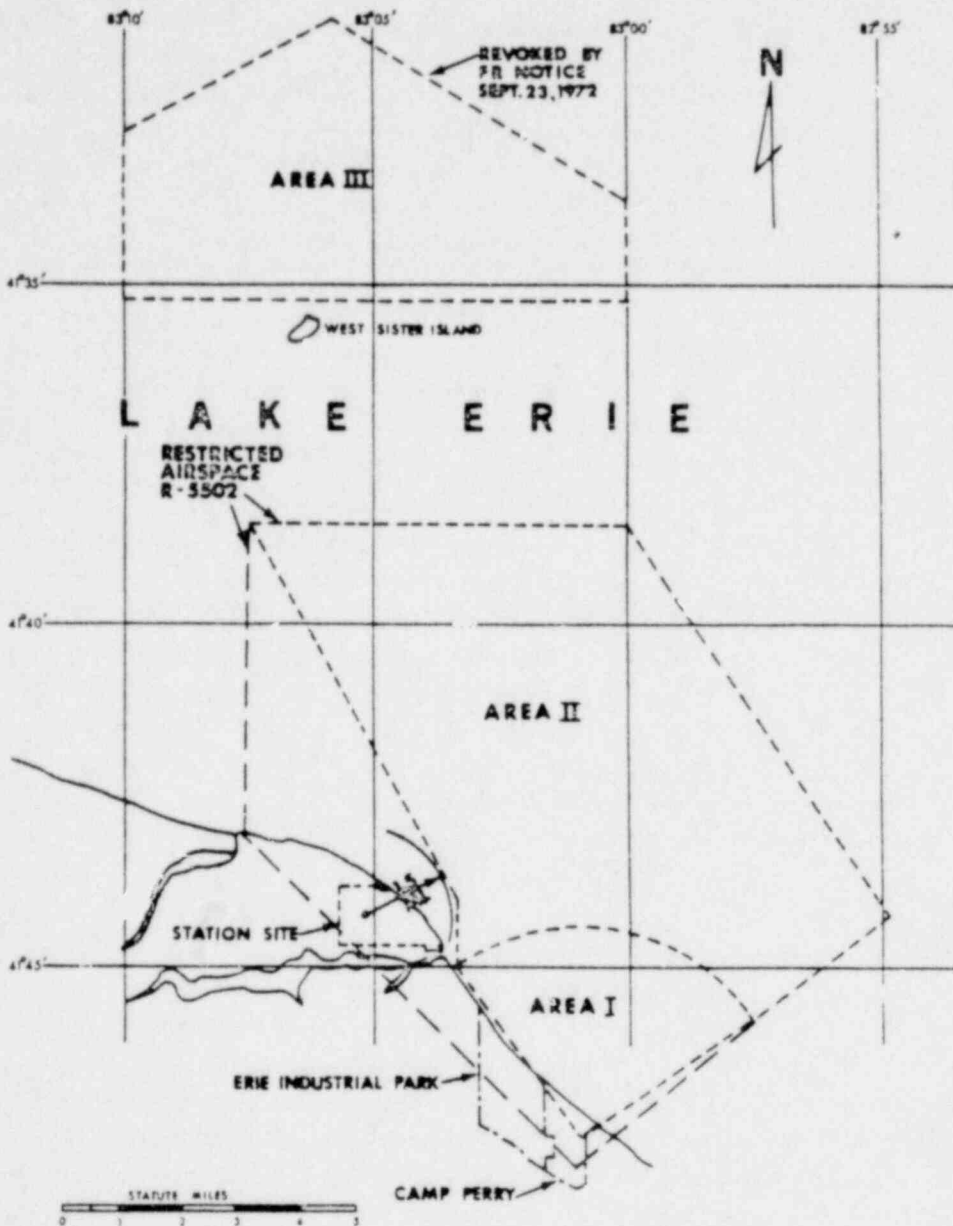
Camp Perry directs the firing of its weapons toward both Restricted Areas I and II. Small arms firing is directed toward Restricted Area I, which has its nearest boundary approximately 1.8 miles from the Davis-Besse plant. The firing of 40-millimeter anti-aircraft guns is directed toward Restricted Area II, but the firing fans limit the possible impact area to a distance of about 2.5 miles from the Davis-Besse station site. In addition, the projectiles carry destruct charges and fuses to prevent surface impact of intact projectiles. These destruct charges limit the maximum range of projectile to approximately two-thirds the distance from the firing area to the station.

On the basis of our review of the present use of the restricted areas, we conclude that the activities associated with these areas have not changed since the Unit 1

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DAVIS-BESSE NUCLEAR POWER STATION
RESTRICTED AREA BOUNDARIES
NOVEMBER 1972
FIGURE 2.5

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construction permit review in a way that would increase the hazard to the site. Since no plant structures are to be located in the restricted areas and testing activities will be limited by the firing fans and destruct charges, we conclude that the activities in those areas pose no danger to the proposed Davis-Besse Units 2 and 3.

There are no oil or gas pipelines within five miles of the Davis-Besse site. The closest airport with a paved runway is at Port Clinton, 13 miles southeast of the site. Because of shallowness of Lake Erie in the area of the site, the nearest shipping lanes are 20 miles from the site. The nearest railroad is the Penn Central, which runs in an east-west direction five miles south of the site. A spur line, owned by the site owners (see Section 2.1.2 of this report), has been constructed to serve the Davis-Besse plant. The line connects the station to the Norfolk & Western Railroad, 7-1/2 miles southwest of the site. The closest highway is State Highway Route 2, located approximately 2,600 feet from the station structures. In view of the distance to major transportation routes and oil and gas lines, accidents involving the shipment of hazardous materials or the rupture of oil and gas lines would not affect the safety of the nuclear facility.

The nature and extent of activities involving potentially hazardous materials which are conducted at nearby industrial, transportation and military facilities have been evaluated to determine if such activities have the potential for adversely affecting plant safety-related structures. Based upon evaluation of information contained in the Preliminary Safety Analysis Report, as well as information independently obtained by the staff, we conclude that the facility is adequately protected and can be operated with an acceptable degree of safety with regard to potential accidents which may occur as the result of activities at nearby industrial, transportation and military facilities.

2.3 Meteorology

Information concerning atmospheric diffusion characteristics of a nuclear power plant site is required for a determination that postulated accidental, as well as routine operational, releases of airborne radioactivity will be within the Commission's guidelines. Regional and local climatological information, including extremes of climate and severe weather occurrences that may affect safe design and siting of a nuclear plant, is required to assure that safety-related plant design and operating bases are within the Commission's guidelines. Meteorological characteristics of a site are determined by our evaluation of meteorological information, performed in accordance with Standard Review Plan Sections 2.3.1 and 2.3.5.

2.3.1 Regional Climatology

The climate of the site can be described as continental, moderated somewhat by the presence of Lake Erie. During the winter the area is frequently affected by southward movements of continental polar air from Canada. The air conditions are

moderated by the air's crossing of the relatively warm waters of the lake. The moderation produces slightly warmer temperatures, excessive winter cloudiness, and frequent snows. In the summer, Lake Erie also has a moderating effect on temperature extremes, and a "lake breeze" circulation is established along and near the shoreline, further cooling the area.

Severe weather is not uncommon, because the site lies near the principal track of winter and spring storms that move northeast and east through the region.

Thunderstorms can be expected to occur on about 40 days per year, being most frequent in June, July and August. Between 1955 and 1967, the one-degree latitude-longitude square containing the site had 13 occurrences of hail greater than 3/4 of an inch in diameter.

Also during the period 1955-1967, 14 tornadoes were reported in the one-degree latitude-longitude square containing the site, giving a mean annual frequency of 1.1. The computed recurrence interval for a tornado at the plant site is about 1,200 years. April is the month with the highest frequency of tornado occurrences. Four waterspouts have been reported on Lake Erie within 50 miles of the site between 1951 and 1973. The design basis tornado characteristics for these plants conform to the recommendations of Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants," for this region of the country.

The applicant has examined meteorological data from the National Weather Service station at Toledo, for the period January 1946 through December 1973, to select the appropriate design basis meteorological conditions for the ultimate heat sink, as recommended in Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants." The meteorological conditions selected by the applicant are conservative, and we consider them to be acceptable for ultimate heat sink design considerations.

The fastest mile wind speed reported at Toledo was 87 miles per hour in March 1948. The applicant selected an operating basis wind speed (defined as the "fastest mile" wind speed at a height of 30 feet above the ground with a return period of 100 years) of 90 miles per hour, which is acceptable for the site.

In the period 1936-1970, there were about 28 periods of atmospheric stagnation reported in the site area covering a total of about 115 days.

2.3.2 Local Meteorology

Climatological data from Toledo, Sandusky, and the onsite meteorological measurements program have been used to assess local meteorological characteristics.

Mean monthly temperatures at the site may be expected to range from about 28 degrees Fahrenheit in January to about 74 degrees Fahrenheit in July. Extreme temperatures of 105 degrees Fahrenheit and -15 degrees Fahrenheit have been reported at Sandusky,

for the period between 1936 and 1965. Over the same period, the annual average precipitation in the site area has been about 34 inches, with about 60 percent occurring in the period April through September; the maximum 24-hour rainfall reported at Sandusky was 5.6 inches; the annual average snowfall was about 29 inches; and the maximum 24-hour snowfall at Sandusky was 12.3 inches.

Wind data obtained from the 20-foot level of the original 300-foot onsite meteorological tower (see Section 2.3.3 below), for the period December 1969 through November 1970, indicate prevailing winds from the west-southwest, southwest, and south-southwest occurring about 38 percent of the time. Winds from the southeast and south-southeast occurred least frequently, less than three percent for each direction. Calms occurred about 2.5 percent of the time.

Wind data obtained from the 35-foot level of the 340-foot tower (see Section 2.3.3 below), for the period August 1974 through August 1976, also indicate prevailing winds from the west-southwest, southwest, and south-southwest occurring about 37 percent of the time. Winds from the east-southeast and southeast occurred least frequently, about 3.5 percent for each direction. Calms occurred less than 0.1 percent of the time during this period.

2.3.3 Onsite Meteorological Measurements Program

A meteorological program, using a 300-foot tower, was initiated in October 1968. Wind speed and direction were measured at the 20-, 100-, and 300-foot levels; vertical temperature gradient was measured between 5 feet and 145 feet and between 145 feet and 297 feet. Dewpoint temperatures were measured at five feet. This tower was instrumented prior to issuance of Regulatory Guide 1.23, "Onsite Meteorological Programs." The construction of the Unit 1 structures and a change in grade elevation caused interference with the collection of the more recent wind speed and direction data from this tower. However, data collected during the period December 1969 through November 1970 do not exhibit any interference problems.

The current onsite meteorological measurements program, operational since August 1974, requires the use of two meteorological towers, 340 feet and 35 feet tall, located about 2,000 feet southwest of the facility containment buildings. A temporary 35-foot tower was in operation from December 1973 to August 1974. On the 340-foot tower, wind speed and direction are measured at the 250-foot and 340-foot levels; vertical temperature difference measurements are made between the 35-foot and 250-foot levels and between the 35-foot and 340-foot levels; and ambient dry bulb temperatures are measured at the 35-foot and 340-foot levels. Precipitation is measured at ground level. The 35-foot tower is used for 35-foot wind speed and direction measurements. The current meteorological measurements program meets the recommendations of Regulatory Guide 1.23 and is, therefore, acceptable.

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The applicant has provided data from the 300-foot tower for the period December 1969 through November 1970. These data, obtained from the 20-foot level, were in the form of joint frequency distributions of wind speed and direction by atmospheric stability class, which is defined by the vertical temperature gradient between 145 feet and 5 feet. Data recovery for this period was 82 percent. The applicant has also provided data from the 340-foot tower for the periods August 4, 1974 through August 3, 1975; August 4, 1975 through August 3, 1976; and August 4, 1974 through August 3, 1976. These data, obtained from the 35-foot level, were in the form of joint frequency of wind speed and wind direction by atmospheric stability class, which is defined by the vertical temperature gradient between 250 feet and 35 feet. Data recovery for these periods was 93 percent.

We have calculated relative concentration (X/Q) values using each set of onsite data. We have some reservations about the quality of the onsite meteorological data collected during the period December 1969 through November 1970, primarily because of the lack of instrument calibration during the data collection period. Also, the lower temperature sensor for the measurement of vertical temperature gradient during this period was only five feet above the surface, which would tend to bias the resultant atmospheric stability distribution towards extremely stable and extremely unstable conditions. Therefore, the relative concentration values presented in Sections 2.3.4 and 2.3.5 below are based upon data collected during the period from August 1974 to August 1976.

The applicant has described the Unit 1 control room monitoring program for measuring pertinent meteorological parameters, and we had found this program to be acceptable during our review of the Unit 1 operating license application. During our review of the Units 2 and 3 Final Safety Analysis Report, we will require the applicant to demonstrate that the operators of Units 2 and 3 will have adequate access to the onsite meteorological data.

2.3.4 Short-Term (Accident) Diffusion Estimates

In the evaluation of short-term (0-2 hours at the exclusion distance and 0-8 hours at the low population zone distance) accidental airborne releases from buildings and vents, we assumed a ground-level release with a building wake factor, cA , of 1,650 square meters. Relative concentration values, for the various time periods following an accidental release, were calculated using the diffusion model described in Revision 2 of Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors," modified to reflect increased lateral plume meander and direction-dependent variations of dispersion conditions, wind frequencies, and exclusion area boundary. Because this method represents a deviation from the guidance provided in Standard Review Plan Section 2.3.4, an explanation of the bases for the deviation is provided below.

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A diffusion model to provide a realistic, yet conservative evaluation of dispersion, while reflecting the results of field dispersion observations, was used to evaluate short term accidental releases at the Davis-Besse site. Credit was allowed for reduction in relative concentrations due to effluent plume meander under stable atmospheric conditions with low wind speeds and for building wake effects. We believe that the use of such considerations is applicable on a general basis as a result of the dispersion experiments conducted at several sites proposed for nuclear power plants as well as at a location containing a plant already in operation. These experiments have demonstrated the existence of lower effluent concentrations, under light wind and stable atmospheric conditions, than those that would have been observed using diffusion estimates presented by Gifford (1968).

The direction dependent evaluation for the exclusion boundary and low population zone provides a constant probability level for X/Q values for individuals as reflected in the wind directional frequencies and the directional dependence of diffusion conditions.

The model and procedures used to calculate short-term (accident) X/Q values, as described in Standard Review Plan Section 2.3.4, would provide the following values:

<u>Time Period</u>	<u>Distance</u>	<u>X/Q seconds per cubic meter)</u>
0-2 hours	Exclusion Boundary (635 meters)	2.8×10^{-4}
0-8 hours	Low Population Zone (3200 meters)	2.5×10^{-5}
8-24 hours	LPZ	1.8×10^{-5}
1-4 days	LPZ	9.0×10^{-6}
4-30 days	LPZ	3.2×10^{-6}

Based upon the modified model, the relative concentration value for the 0-2 hour time period was evaluated to be 2.1×10^{-4} seconds per cubic meter at an exclusion distance of 935 meters north of Unit 2. The relative concentration values for various time periods at the outer boundary of the low population zone (3200 meters) are:

<u>Time Periods</u>	<u>X/Q (seconds per cubic meter)</u>
0-8 hours	3.0×10^{-5}
8-24 hours	2.1×10^{-5}
1-4 days	1.0×10^{-5}
4-30 days	3.4×10^{-6}

2.3.5 Long-Term Diffusion Estimates

Reasonable estimates of average atmospheric dispersion conditions, to be used in estimating routine airborne releases of radioactivity at the site, were made using

the two years of onsite data and a model for long-term releases that applies the methods described in Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors." The model evaluates routine releases from the plant vent and turbine building vent which are continuous. In addition, periodic releases from the plant vent were evaluated. An estimate of maximum increase in calculated relative concentration due to recirculation of airflow, using Figure 2 of Regulatory Guide 1.111, which is not considered in the straight-line trajectory model, was included in the calculations. The calculations of doses reported in Section 11.0 of this report also included consideration of radioactive decay of effluents and depletion of the effluent plume.

Relative concentration (X/Q) and relative deposition (D/Q) values were evaluated at various points of interest. The highest undecayed-undepleted values of X/Q as well as D/Q values for each type of location of interest are given in Table 2.2 for continuous and periodic releases.

2.3.6 Conclusions

The applicant has provided an adequate description of the regional and local meteorological conditions of importance to the safe design and siting of Davis-Besse Units 2 and 3. We conclude that two years (August 1974 - August 1976) of onsite meteorological data provide an adequate meteorological description of the site and vicinity, and that these data provide an acceptable basis for calculations of reasonably conservative relative concentration values for assessments of post-accident conditions and annual average atmospheric diffusion conditions.

2.4 Hydrologic Engineering

2.4.1 Hydrologic Description

The site for the Davis-Besse facility is located on the southwestern shore of Lake Erie. The major plant structures will be located approximately in the center of the site, 3,000 feet from the shoreline. All elevations are referenced to the International Great Lakes Datum, which was established in 1955 by the U.S. Department of Commerce. The International Great Lakes Datum is mean sea level minus 1.454 feet. The low water datum for Lake Erie is 568.6 feet International Great Lakes Datum. The original topography of the site was relatively flat, with elevations varying from 568.6 feet International Great Lakes Datum to about 575 feet International Great Lakes Datum. The facility structures will be on an existing slightly elevated upland section that is separated from the lake by an adjacent marsh area and a narrow beach ridge that lies between the marsh and the lake. Plant grade is to be 584 feet International Great Lakes Datum, with entrance levels to structures at 585 feet International Great Lakes Datum. The southern site boundary borders the Toussaint River, which is approximately 3,000 feet from the major facility structures. A wave protection dike has been installed along the northern and eastern sides of these

TABLE 2.2

CONTROLLING RELATIVE CONCENTRATION (X/Q) AND RELATIVE DEPOSITION (D/Q) RESULTING FROM ROUTINE RELEASES

	<u>DIRECTION</u>	<u>DISTANCE MILES</u>	<u>X/Q (seconds per cubic meter)*</u>	<u>D/Q (inverse meters squared)</u>	
Site Boundary	A	N	.45	1.7×10^{-5}	--
	B	N	.45	3.3×10^{-5}	--
	C	N	.45	3.8×10^{-5}	--
Residence/ Garden	A	W	.60	5.3×10^{-6}	6.3×10^{-8}
	B	W	.60	1.1×10^{-5}	1.3×10^{-7}
	C	W	.60	1.2×10^{-5}	1.5×10^{-7}

A = Continuous

B = Gaseous waste system (15 purges per year, for 8 hours each)

C = Containment purge (24 purges per year, for 2 hours each)

* No decay, no depletion

structures at an elevation of 591 feet International Great Lakes Datum. All surface water from these elevated areas will be collected and carried in storm drains to ditches that empty into either the marsh area or the Toussaint River. An intake canal has been installed between the intake structure at the plant site and the beach ridge. The intake canal is connected to Lake Erie with an 8-foot diameter, underground and underwater pipe that extends 3,000 feet out into the lake. This is the single source of cooling water for the service water system for normal operation. During emergency operation, an existing, below-ground rock quarry, and a seismic Category I open forebay area ahead of the intake structure, will serve as assured sources of water in case of an extreme lowering of the lake due to meteorological conditions, or in case of collapse of the intake canal or the submerged pipes (see Figure 1.1).

Lake Erie is the primary source of potable water in the area. The five nearest lake water users are located between 3.6 and 12 miles from the plant cooling water discharge.

The Sand Beach community, with 122 residences, is located along the beach ridge, commencing at the northern site boundary at the shoreline. Approximately 50 percent of these residences obtain household water, for all purposes, from beach wells located in the lakefront sand. These wells are three to six feet deep, are located 10 to 20 feet from the lake shoreline and, for all practical purposes, may be considered surface water supplies.

2.4.2 Flood Potential

Several potential flood producing sources were investigated by the applicant and reviewed by us. The potential sources include Lake Erie, the Toussaint River, and the site drainage in the vicinity of safety-related structures.

- (1) Lake Erie - The applicant investigated the probable maximum stillwater lake levels, based upon lake levels plus wind tides and transverse seiche. The maximum historical lake level, 573.5 feet International Great Lakes Datum, was recorded in June 1973. Maximum calculated wind tide was 9.3 feet, due to a probable maximum meteorological event, based upon a procedure developed by Platzman. A probable maximum transverse seiche of one foot was also added to a lake level of 573.4 feet International Great Lakes Datum (0.1 foot lower than record) to yield a maximum stillwater lake level of 583.7 feet International Great Lakes Datum, 0.3 foot lower than the yard grade of 584.0 feet International Great Lakes Datum. The maximum wave height at the plant is governed by the maximum depth of water between the lake shore and the plant. The maximum wave that can be supported at the maximum stillwater lake level is 8.5 feet. This wave will produce a maximum runup of 6.6 feet on the 3-to-1 dike slope. This yields a maximum runup elevation of 590.3 feet International Great Lakes Datum, which is 0.7 foot below the top-of-dike elevation of 591.0 feet

International Great Lakes Datum. The lakeside face of the dike will be protected with a three-foot layer of random-placed angular quarry stone on a six-inch layer of granular material two inches or less in diameter.

- (2) The Toussaint River - The river empties into Lake Erie southeast of the site. The stream has a drainage area of about 143 square miles and an average slope of about two feet per mile. The stream is ungaged and there are no dams on it. The lower six miles of the stream are much wider than the remainder, and flow is controlled by the level of Lake Erie. The applicant conservatively estimated peak flow rate for the probable maximum flood, based on probable maximum precipitation, to be 78,500 cubic feet per second. It was conservatively assumed that there would be no flow to the lake during the probable maximum flood, and the maximum stage associated with this "dammed up" condition would be 579.0 feet International Great Lakes Datum, well below plant grade level of 584.0 feet International Great Lakes Datum.
- (3) Site Drainage - The proposed site drainage systems have been designed such that local probable maximum precipitation will not constitute a threat to the safety-related structures. The applicant conservatively assumed that all site storm drainage systems are blocked and filled with water at the start of the probable maximum precipitation. Runoff under this assumption would reach 584.5 feet International Great Lakes Datum, which is 0.5 foot above the high point of all roads and grounds, and 0.5 foot below the floor grade of all safety-related buildings.

All seismic Category I structures will have a 2.5-foot parapet at the periphery of their roofs. Those roofs that have penetrations will have curbs around the penetrations and will have horizontal roof drains with their invert at least 12 inches below the top of the curb. All roofs will have regular roof drainage systems. The horizontal drain pipes will be designed to drain the local probable maximum precipitation should all the regular roof drains become clogged.

We have reviewed the applicant's flood design considerations, and conclude that all safety-related structures will be designed to be safe from all flood potentials up to the probable maximum magnitude. The applicant has satisfied the recommendations of Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants," and, therefore, we conclude that the flood design considerations are acceptable.

2.4.3 Design Basis Water Levels

The design basis stillwater level is elevation 583.7 feet International Great Lakes Datum. The facility will be protected along the north, east, and partially on the south side, by an earthen breakwater built up to elevation 591.0 feet International Great Lakes Datum. The design basis wave runoff on the breakwater is calculated to be elevation 590.3 feet International Great Lakes Datum. All seismic Category I

structures will be designed for the design basis stillwater level of elevation 583.7 feet International Great Lakes Datum. The design basis for hydrostatic roof loading is 1.5 feet of water, which is the maximum possible depth below the horizontal roof drains.

2.4.4 Ice Flooding

The applicant has concluded that ice flooding of safety-related structures from either Lake Erie or the Toussaint River will not present any hazards, because of the distance between the site and the relevant water bodies and because of the freeboard between the yard grade and the stillwater lake and river levels. Even if the ice should reach the dikes on the north and east sides of the plant, these dikes are designed to withstand the ice pressures. We have reviewed the applicant's information and conclude that the facility design is acceptable with respect to ice flooding.

2.4.5 Water Supply

All cooling water will be supplied from Lake Erie. The intake canal forebay is of seismic Category I design and will be used as a heat sink reservoir in the event of low water or an accident. The applicant has used a procedure developed by Platzman to determine the maximum wind tide set-down at the site due to a probable maximum meteorological event. The maximum wind tide set-down was calculated as occurring at Toledo. Since the facility site is located about 80 percent of the way from the wind tide node (that point in the lake where no wind tide change in lake level occurs) to Toledo, wind tide variations at the facility site were reduced by 20 percent from Toledo wind tides. This procedure gives a maximum wind tide set-down with west-southwest winds, of 9.3 feet. The maximum variation of record in the mean monthly level of Lake Erie is 1.2 feet below the low water datum. The applicant has used a value of 1.5 feet below the low water datum as a prior condition to the above procedure. A transverse seiche causing an additional set-down of one foot was also assumed. These lake level losses total 11.8 feet, and give a minimum stillwater lake level of 556.8 feet International Great Lakes Datum. In the event the lake level drops below the minimum operating level of 562.0 feet International Great Lakes Datum, the reactor can be brought to a safe shutdown condition using the seismic Category I ultimate heat sink. In order to assure adequate shutdown cooling capability in the event of a low lake level, the technical specifications for operation of Units 2 and 3 will require frequent monitoring of lake level when the level approaches the minimum operating level. The numerical values of level and monitoring frequency will be determined during the operating license review stage.

The ultimate heat sink for Unit 1 is the seismic Category I forebay. The Unit 1 service water will circulate from the forebay through the Unit 1 plant components and back to the forebay. The forebay alone would not be adequate to accommodate the combined heat loads of all three units in the event of an accident in one unit and

the following shutdown of the other two units. Therefore, the water in an onsite rock quarry will be used as a heat sink to supplement the water in the forebay. The quarry has been designated by the applicant as the ultimate heat sink for Units 2 and 3. When the quarry water is needed, it will be pumped into the forebay, where it will mix with the lake water in the forebay, and then will be pumped through the Units 2 and 3 plant components and back to the quarry. The quarry is a below-ground rock quarry with a capacity, at the 21-foot maximum operating depth, of 231 acre-feet and a surface area of 11 acres. Two seismic Category I pipe lines will be provided to carry the quarry water to the seismic Category I intake forebay.

Since the Unit 1 ultimate heat sink will be directly connected to the ultimate heat sink for Units 2 and 3, it was necessary to analyze the two heat sinks as one complex. We have made independent analyses of this complex and, based upon these analyses, we have concluded that there will be sufficient water at an acceptable temperature for emergency shutdown of one unit with coincident normal shutdown of the two other units and for maintaining them in a shutdown condition for at least 30 days. Additional water supply is available from Lake Erie to extend this time beyond 30 days. The computed return water temperature for the service water pumps in the intake forebay is 107 degrees Fahrenheit. This is less than the maximum allowable service water temperature of 115 degrees Fahrenheit and, therefore, is acceptable. We conclude that the ultimate heat sink complies with Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants," and, therefore, is acceptable. See also Sections 2.6.5 and 9.2.3 of this report.

2.4.6 Groundwater

The major groundwater sources in the area are the water-bearing Silurian and Dovanon Formations consisting of thick layers of limestone and dolomite. The major water-bearing confined aquifer is between elevations 500 and 555 feet International Great Lakes Datum. Well yields from the carbonate aquifer vary considerably in the vicinity of the site.

In the vicinity of the site the groundwater gradient is about two feet per mile toward Lake Erie. This is generally the same slope as that of streams in the area. The site is underlain by about 17 feet of glaciolacustrine and till deposits that overlie the Tymochtee Formation. These deposits consist basically of silty clay with very low permeability, which has created artesian groundwater conditions in the water-bearing Tymochtee and underlying carbonate bedrock formations. The applicant has estimated the average groundwater flow velocity to be about five feet per year by assuming the carbonate bedrock aquifer to be homogeneous and isotropic with an average permeability of 1×10^{-2} centimeters per second and an average gradient of two feet per mile. In addition to the beach wells, there are 30 groundwater wells within a three-mile radius of the site. Thirteen wells are no longer being used, and the remaining 17 are used only intermittently for irrigation and sanitation purposes. The applicant has concluded, and we concur, that the possibility of

extensive contamination of the groundwater aquifer from a postulated liquid spill of radioactivity is low because: (1) redundant safety features have been incorporated into the design of the facility; (2) the piezometric gradient and corresponding groundwater flow velocities are small; and (3) the groundwater gradient is toward the lake and there are no users between the site and the lake. Also, the impervious surface deposits, of predominantly clay composition, will retard the rate of flow to the aquifer, and the dissolved radionuclides will react with the clay.

We have made an independent analysis of the potential for groundwater contamination from an accidental liquid spill of radioactivity. The foundation of the radwaste building will be at elevation 545 feet International Great Lakes Datum, and the top of the confined aquifer, in the plant area, is at elevation 555 feet International Great Lakes Datum. Therefore, any postulated accidental liquid spill of radioactivity from tanks in the radwaste building would leak to the confined aquifer, if leakage were possible. However, the hydrostatic pressure associated with the confined aquifer, in the vicinity of the radwaste building, is about elevation 570 feet International Great Lakes Datum. This positive pressure on the radwaste building foundation will preclude leakage from the building into the confined aquifer. We also noted that the groundwater gradient is about two feet per mile toward Lake Erie and that there are no wells between the site and Lake Erie that draw from the confined aquifer.

As a conservative measure, we have calculated the travel times and dilution factors for the three points described below:

- (1) The Lake Erie near-field release point. The interface between Lake Erie and the confined aquifer was conservatively assumed to be 3,000 feet from the lake shoreline. Consequently, contaminated groundwater would travel 6,000 feet horizontally from the facility to Lake Erie and then vertically 30 feet to the lake surface. We estimate a dilution factor of 2750 and a travel time of 72 years.
- (2) The Sand Beach wells. Our calculated dilution factor of 5.5×10^6 and travel time of 72 years for this point are based upon the assumption that radioactive liquid would have to travel from the radwaste building, through the confined aquifer to the Lake Erie interface (approximately 6,000 feet), then 3,000 feet through Lake Erie to the beach wells. The beach wells are not hydraulically connected to the confined aquifer.
- (3) The Lake Erie Industrial Park. This is a surface water source located about five miles east of the plant site. For this point our calculated dilution factor is 9.8×10^6 and travel time is 72 years.

Based upon the above considerations, and upon the results of radiological dose calculations presented in Section 15.5.3 of this report, we conclude that groundwater

contamination due to an accidental liquid spill of radioactivity would be negligible and that the radiological effects to the nearest user would be negligible.

2.4.7 Conclusions

Based upon our review of the flood analysis for the facility site, we conclude that the flood protection measures to be provided are acceptable. Our review included consideration of maximum wind tide and wind waves on Lake Erie due to a probable maximum meteorological event, probable maximum flood levels for the Toussaint River, ice effects on Lake Erie and the Toussaint River, and flood conditions at the site and on rooftops due to a local probable maximum precipitation.

We have made an independent analysis of the potential for contamination of ground-water wells and surface water intakes in the vicinity of the site. We conclude that, in the event of an accidental spill of liquid radioactivity, leakage to the groundwater aquifer is improbable. However, even if leakage to the aquifer were to occur, concentrations at the nearest water user would be a small fraction of those permitted by 10 CFR Part 20 (see Section 15.5.3).

We find the applicant's analysis of low water conditions at the site to be acceptable. Although extreme low lake levels would preclude the use of the intake crib as a source of normal cooling water, the plant will be able to be brought to a safe shutdown condition with the seismic Category I ultimate heat sink.

Based upon our review and independent analyses of the ultimate heat sink, we conclude that the present heat sink design and conditions of operation will be adequate to provide facility cooling water at a temperature less than 115 degrees Fahrenheit (maximum allowable for equipment design) for a 30-day period.

2.5 Geology and Seismology

2.5.1 Introduction

We have reviewed the geologic, seismic and foundation engineering characteristics of the proposed facility site and its environs. The review included consideration of the information presented by the applicant and that in recently published pertinent literature. Our review included a study of the geologic development of the region and the geologic structures produced by the various deformational events. We considered the existence of nearby capable faults which might cause surface displacement or earthquakes to occur at the proposed site. We reviewed the seismic history of the region and the design basis earthquakes proposed by the applicant. We also reviewed the subsurface conditions on the site and the foundation engineering design proposed by the applicant for safety-related structures.

2.5.2 Geology

Our geology review for Units 2 and 3 was based on the review we performed for Davis-Besse Unit 1, the results of which we reported in the Safety Evaluation Report, dated November 2, 1970, for the Davis-Besse Unit 1 construction permit application (Docket No. 50-346).

The site is located in the Great Lakes section of the Central Lowland physiographic province. Tectonically, the site is located on the east flank of the Findlay Arch, an ancient, northeast-trending anticlinal structure that separates the Michigan Basin on the northwest from the Appalachian Basin on the southeast. The trace of the axis of the Findlay Arch is inferred to be about 15 miles west of the site. There are no geologic structures or faults known in the vicinity of the site that could be expected to localize seismicity.

In the area of Davis-Besse Units 2 and 3, about eight to 16 feet of Pleistocene fill and glaciolacustrine silty clay overlie nearly flat-lying Paleozoic sedimentary rocks. Bedrock underlying the site consists of dolomitic strata of the Tymochtee Formation containing interbedded soluble gypsum and anhydrite. The Tymochtee Formation is part of the Bass Islands Group of Silurian age.

We have concluded, based upon our review of the geological information available for Unit 1 and the subsequent updated information received for Units 2 and 3, that there are no geologic faults or other tectonic structures that present a potential hazard to the proposed site. During our review, we asked the applicant to investigate the possible existence of previously unidentified tectonic structures beneath Lake Erie that might be of significance to the seismic design of the nuclear station. No evidence of any structural anomalies beneath central and western Lake Erie were identified.

2.5.3. Seismology

The site lies within the Central Stable Region Tectonic Province described by Eardley. We have accepted this and other large tectonic provinces defined in the literature (King 1964) as guidance in assessing the appropriate seismic design in the eastern United States. Beneath the surface sediments, the Central Stable Region Province is characterized by a series of arches, basins and domes formed during the Paleozoic Era. King describes this area as "platform deposits on Precambrian foldbelts." The site is situated on the east flank of the Cincinnati-Findlay arch structure, which extends from central Tennessee to southern Ontario. In west central Ohio, the Kankakee Arch splays off to the northwest not far from the town of Anna.

Vibratory motion at the site was estimated using bases discussed in Appendix A to 10 CFR Part 100. These include seismicity of tectonic provinces and structures

within 200 miles of the site. Large earthquakes, at greater distances, that might also affect the site were also considered.

- (1) Tectonic Province: Earthquakes of intensity VII within the U.S. portion of the Central Stable Region have occurred in Oklahoma, Kansas, Nebraska, Michigan, Illinois, Indiana and Ohio. Earthquakes of greater intensity within this province have occurred at Anna, Ohio (VII-VIII), Keweenaw Peninsula, Michigan (VIII), and Attica, New York (VIII). Of the latter three events, we consider only the Attica earthquake to have a demonstrated association with local structure. Docekal (1970) suggests that the Michigan earthquake was associated with the Mid-Continent Geophysical Anomaly. Recent investigations by Frantti (1975) have correlated this and several other events, characterized by high local intensity and small felt area, with areas of intense mining activity. In the 1906 event included in Frantti's review, the high intensity was related to an actual mine collapse over a tunnel. In a 1905 event in Calumet, Michigan, no surface or subsurface failure was found even though intensity VIII was reached at several points. These events appear to be a case of earthquakes induced by mining activity.

The applicant has found a spatial relationship between the Anna, Ohio, earthquake (and related smaller events) and a bifurcation in the Cincinnati Arch. He believes that this event may be related to a local structural anomaly or weakness. This, however, has not yet been demonstrated so as to be acceptable to us. Thus, we consider a Modified Mercalli intensity VII-VIII to be a reasonably conservative value for the site safe shutdown earthquake, based upon the seismicity of the Central Stable Region tectonic province and the seismicity and geological structure of the site vicinity.

- (2) Tectonic Structure: A series of earthquakes in the vicinity of Anna, Ohio, includes 31 felt events during the past 100 years, 4 of intensity VII and 1 of intensity VII-VIII. These earthquakes occurred in a small cluster at the junction of the Cincinnati Arch with its northeast branch, the Findlay Arch. Although this is seismically the most active point on the structure, it is not the only place along the arch where earthquakes have occurred. Smaller events (intensity VI or less) have occurred along the Cincinnati-Findlay Arch in Tennessee, in southern Ohio, and in the Toledo area. A study done for the Fort Calhoun site (Docket No. 50-548) lists 75 events that have occurred along the Cincinnati-Findlay structure. Another study, conducted for the Marble Hill site (Docket Nos. 50-546/547), presented evidence of faulting associated with the cluster of seismicity near Anna, Ohio. However, pending the outcome of ongoing seismic monitoring and additional geophysical studies in the area, we regard the postulated association between the Anna, Ohio, earthquake activity and the identified faults as being inconclusive. In the initial report of a seismograph station recently installed in the Anna area, the only prominent local or near local event to occur was a magnitude 3.4 earthquake astride the

Findlay Arch in westernmost Lake Erie. This event was felt in parts of eastern Michigan and southern Ontario. It is presently not possible to determine whether these earthquakes are indeed related to the Findlay Arch or are part of a random or undetermined pattern of seismicity.

The applicant has considered, and partially rejected and partially accepted, the possibility that the Anna, Ohio, earthquakes may be related to the Findlay Arch. In his discussion of the site safe shutdown earthquake, he has assumed that: (1) an earthquake larger (VIII-IX) than that indicated by historical seismicity could occur only at Anna, and (2) an earthquake equivalent in size to the largest historical earthquake VII-VIII could occur on the Findlay Arch at the site.

We consider the correlation of earthquake activity with the Cincinnati-Findlay Arch or with localized faulting to be unresolved.

- (3) Large Distant Earthquakes: The largest earthquake to occur in the mid-continent region of the United States was the intensity XI-XII New Madrid earthquake of 1812. This earthquake has been associated with the central Mississippi Valley seismic zone. It is our position that present knowledge would allow the earthquake zone in which such an earthquake could occur to extend as far as Vincennes, Indiana, in the Wabash Valley. Assuming an intensity XI-XII earthquake entered at Vincennes, and utilizing Gupta and Nuttli's intensity distance relationship, results in intensity VIII at the site, with an epicentral distance of 300 miles.

The above analysis leads to a maximum local intensity VII-VIII at the site from nearby sources. Utilizing the Trifunac and Brady intensity acceleration curve yields a peak acceleration of 0.2g.

Nuttli's (1973) analysis of intensity and ground motion at distances greater than 100 miles, for central U.S. earthquakes, indicates lower peak accelerations and relatively stronger motions at long periods (around one second). Previous reviews of the Callaway site (Docket No. 50-483) showed that, even at the closer epicentral distances of 155 miles, the response spectrum of Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," scaled to 0.2g, would adequately envelope the response spectra expected for a New Madrid type earthquake. Therefore, we have concluded as a result of our review that, for the safe shutdown earthquake, the appropriate reference ground acceleration for seismic design of Davis-Besse Units 2 and 3 structures is 0.2g. The applicant, using a maximum intensity of VII-VIII and the Gutenberg-Richter intensity acceleration relationship, arrived at a maximum acceleration of 0.15g. It is our position that effective acceleration values should be at least as large as those predicted by the trend of the mean of Trifunac and Brady.

In response to this position the applicant submitted, with a letter dated August 29, 1977, several arguments as to why a reference acceleration of 0.15g is sufficiently conservative. The applicant maintains that the controlling tectonic province earthquake, i.e., the intensity VII-VIII Anna, Ohio, earthquake of March 8, 1937, has a causal relationship with local faulting in the Anna area, and was, in truth, an intensity VII event whose damage effects were confused with and enhanced by an intensity VII event that occurred in the Anna region on March 2, 1937. As stated above, we consider the correlation of earthquake activity with local faulting to be unresolved. In addition, we only accept those intensities listed in the Earthquake History of the United States and those changes agreed upon by the appropriate governmental body (U.S. Geological Survey or National Oceanic and Atmospheric Administration) charged with assigning these intensities. The intensity assigned to the March 8, 1937 event is VII-VIII. The applicant maintains that, even if he accepts an intensity VII-VIII at the site, a more recent correlation of acceleration with intensity, presented by O'Brien, Murphy and Lahoud, indicates that the appropriate peak acceleration used to anchor the Regulatory Guide 1.60 response spectrum would be between 0.13g and 0.14g rather than 0.20g. The staff's evaluation is that the material and analysis presented by O'Brien, et al., do not warrant such a change.

The applicant maintains that, following the procedure used in Appendix 2.5K to the WPPSS No. 3 Project (Docket No. 50-508), real earthquake records deconvolved from the surface of a thick soil deposit (the conditions at Anna) to a rock or thin soil outcrop (the conditions at the Davis-Besse site) result in a reduction of ground motion. On the basis of the material presented, we cannot accept this argument.

Our position is unchanged, and was reiterated to the applicant in our letter of February 3, 1978. In response, the applicant submitted a letter dated March 7, 1978, in which he maintains his position that 0.15g is an adequately conservative reference acceleration for the safe shutdown earthquake. However, in his March 7, 1978 letter, the applicant has committed to using the 0.20g reference acceleration for design purposes. We consider this matter to be resolved.

Appendix A to 10 CFR Part 100 defines the operating basis earthquake as that earthquake that could reasonably be expected to affect the plant site during the operating life of the plant. Probabilistic techniques, taking into account historical seismicity and regional geology, appear to us to be the best method for arriving at quantitative estimates of the operating basis earthquake. Short or inaccurate earthquake histories and an insufficient understanding of seismogenic geological structure can place constraints on the use of such techniques. One recent study by Algermissen and Perkins (1976) has estimated, for different locations in the United States, the maximum horizontal acceleration in rock that have a 90 percent probability of not being exceeded in 50 years. The peak value in a region consisting of most of Ohio and southeast Michigan is 0.07g. The mean return period associated with this acceleration level is 475 years. This is equivalent to an 8 percent probability of exceeding a level of 0.07g at the site during the 40-year life of the

plant. This peak value corresponds to intensity VI, utilizing the Trifunac and Brady intensity acceleration curve. Intensity VI is not an unreasonable estimate, considering that the maximum intensity felt at the site, as determined from historical records, was intensity V. The applicant has proposed a maximum horizontal acceleration of 0.08g for the operating basis earthquake. We consider this to be a sufficiently conservative value in that a more severe earthquake could not reasonably be expected to affect the plant site during the operating life of the plant and conclude that it meets the requirements of 10 CFR Part 100, Appendix A.

2.6 Foundation Engineering

2.6.1 Subsurface Conditions

The station structures will be founded on sedimentary rock. Overburden soils at the site consist of a layer of glacial deposits ranging in depth to 22 feet. The top three to 12 feet are recent glaciolacustrine soils classified as stiff silty clay. These soils have been desiccated and are, therefore, much stiffer than most glacial lake deposits of this type. These upper soils have an unconfined compressive strength of about 3.5 tons per square foot and an average standard penetration resistance of 15 blows per foot. Underlying the lacustrine material is three to 12 feet of glacial till, described as hard silty sandy clay with less than 10 percent gravel. The till has an average unconfined compressive strength of eight tons per square foot and an average standard penetration resistance of about 30 blows per foot.

Bedrock consists of horizontally stratified argillaceous and gypsiferous dolomite of the Tymochtee Formation. The rock has an upper 8- to 10-foot thick mantle of massive dolomite that contains small anhydrite inclusions and vugs. The lower limit of this zone is generally between elevations 550 and 555 feet International Great Lakes Datum. Underlying the dolomite mantle is soft-to-hard, thinly-bedded dolomite containing many laminae of gypsum, anhydrite and shale.

The piezometric surface at the site is at about elevation 571 feet International Great Lakes Datum, which is about 10 feet above top-of-rock. A piezometric head of 10 feet is confined in the foundation rock by the overlying impervious glacial till.

2.6.2 Solution Cavities

The dolomite foundation rock at the site is susceptible to solutioning. Evidence of solutioning and cave development can be seen along the western shore of South Bass Island, approximately 15 miles east of the site. The initial foundation investigations for Davis-Besse Unit 1 (Docket No. 50-346) revealed only the presence of minor solution activity at the site. About 15 percent of the borings and rock probes in the subsurface investigations program for Unit 1 encountered cavities. Most of the cavities were small, on the order of one foot in depth. Additional investigations were completed during construction of Unit 1, primarily in the foundation areas of

seismic Category I and other major structures. This was a verification measure taken to assure that there were no significant solution cavities in these foundations that were previously undetected. The program did not identify the presence of any significant features in the area of seismic Category I structures, but a zone of major solution was found along the northern perimeter of the foundation for the Unit 1 cooling tower. Solution cavities were found ranging up to three feet in depth and up to 15 feet in width. The solutioning was found to be concentrated in a shallow zone between elevations 546 and 556 feet International Great Lakes Datum, and this zone was successfully pressure-grouted.

The foundation investigation program for Units 2 and 3 was completed during construction of Unit 1 at about the same time the foundation verification program for Unit 1 was being carried out. The program was similar to that used for Unit 1, and included borings spaced on 50-foot centers, rock probes, and geophysical exploration. In addition, an exploratory excavation was made to inspect and evaluate a surface depression (collapse feature) in the Units 2 and 3 area. Solution cavities in the excavation were observed as mostly occurring along joints. Fissures were formed up to several feet wide, increasing in width to about 10 feet at joint intersections where the overlying soil and rock material had collapsed into the feature and partly filled the void. Solutioning did not appear to extend below elevation 534 feet International Great Lakes Datum.

About 70 percent of the exploratory borings and rock probes completed in the area for Units 2 and 3 encountered cavities, mostly in the zone between elevation 545 and 555 feet, roughly corresponding to the level where features were found in the foundation for the Unit 1 cooling tower. Most of these cavities were shallow (one foot or less) and, typically, two or three cavities were found in a single boring, indicating possible solutioning at different elevations along bedding planes. Cavities were detected primarily by observing losses in drilling fluid and the sudden drop of drill rods. Many of these solution features are thought to be filled with soil, although this could not be determined from the core boring data. It appears that moderate to severe solutioning, defined by the continuity of cavities from boring to boring, exists in the zone of rock between elevations 545 and 555 feet International Great Lakes Datum throughout a major portion of the Units 2 and 3 area, but is probably concentrated in the vicinity of the observed surface depression. The intensity of solutioning appears to decrease, away from the surface depression area, as shown in Figure IV-13 of the Preliminary Safety Analysis Report, Appendix 2C. However, we suspect that there may be other similar areas of solutioning at the site apparently detached from this zone, such as those found in the foundation of the Unit 1 cooling tower. This possibility should not be precluded, and will be further investigated and evaluated during construction. Because of these conditions, the staff will work with the applicant during construction to evaluate actual field conditions and to make any adjustments needed in the applicant's foundation investigation and treatment program to assure the safety of the plant.

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2.6.3 Additional Foundation Investigations and Treatment During Construction

The applicant has committed to conduct additional subsurface investigations, during construction, in foundation areas of all seismic Category I structures. The program is to consist of additional core borings, drill hole probes and geophysical exploration using surface resistivity techniques and microgravimetric surveys. Details of the exploratory program proposed by the applicant are described in the Preliminary Safety Analysis Report, Appendix 2C.

We reviewed the program that was proposed initially and found it to be acceptable, except for the exploration of foundations for lighter seismic Category I structures. Our position required additional borings to be made to investigate certain seismic Category I structures that will be founded above elevation 545 feet International Great Lakes Datum. The basis for this position is discussed further in Section 2.6.4 of this report. For these lightly loaded foundation areas, we recommended a minimum boring coverage equivalent to a grid with borings spaced 20 feet on centers, and extending to elevation 540 feet International Great Lakes Datum. We noted that percussion drilling, in place of core drilling, is acceptable, provided proper logging techniques are used. Also, the exploration borings may be used in conjunction with the proposed foundation grouting program under foundation grades above elevation 545 feet, as discussed below.

We notified the applicant of our concerns. The applicant responded by letter dated March 7, 1978, providing us with a more detailed plan for investigating and treating foundations for lighter structures. We accept the proposed plan subject to the following modifications:

- (1) At least three exploratory drill holes should be included in the bedrock verification program for the service water valve room, three for the ultimate neat sink valve room, and three for each borated water storage tank.
- (2) All exploratory drilling should extend down to or below elevation 540 feet International Great Lakes Datum.
- (3) All drill holes and borings should be pressure-grouted according to the specifications of the foundation grouting program.

Because foundation conditions at this site are complex, it may be necessary during construction to conduct further investigations beyond the program now proposed. Additional borings and exploratory excavations may be used to further explore, inspect and evaluate unanticipated solution features discovered during construction and during the course of the scheduled investigation program. Plant facility excavations that expose the foundation rock will be thoroughly evaluated during construction to assess the effectiveness of foundation treatment. We will observe actual field conditions, and will review with the applicant the results of the

additional foundation investigations and remedial foundation treatment. The applicant has committed (1) to notifying us at least 45 days prior to the commencement of any grouting work, and (2) to providing, before work commences on the foundation mats, a final foundation report for review during construction while the major excavations are still open. We will use this information during site visits to assess the need for any additional exploration or for possible redesign of foundations.

The applicant's procedures for pressure grouting presented in the Preliminary Safety Analysis Report, Revision 18, are acceptable. However, grout holes should be logged and used for exploration as well as for grouting purposes. Logging should consist of measuring changes in drilling penetration rates, and noting rod drops and changes in the color of cuttings. This logging can provide useful information regarding subsurface conditions and can aid in detecting anomalous zones and solution features that may have low grout takes. We required and the applicant agreed to perform the logging indicated above and use the results of this logging to detect solution features that may not be detected by grout takes alone.

2.6.4 Foundations for Structures
Major Seismic Category I Structures

The containment building and major portions of the auxiliary building will be founded in rock below the level of significant solutioning. The containment building will be founded on a circular bowl-shaped concrete mat at about elevation 528 feet at its center and about elevation 540 feet along its periphery. A large portion of the auxiliary building (Area 7 and a major part of Area 8, as shown in the Preliminary Safety Analysis Report, Figure 3.8-15) will be founded on a common mat at about elevation 541 feet.

The applicant originally proposed to establish the foundation level, for two other areas of the auxiliary building, above the zone of significant solutioning. Area 6 was to be placed on a mat foundation at about elevation 582 feet, supported by fill concrete down to the top-of-rock at elevation 555 feet. A portion of Area 8 was to be founded on spread footings supported at the top-of-rock. Foundation areas in the solutioned rock above elevation 540 feet were to be grouted.

Based upon our review of the subsurface data presented through the Preliminary Safety Analysis Report, Revision 20, we have concluded that foundation conditions between elevation 545 feet and elevation 560 feet are too poor and complex to warrant the positive judgment that the subsurface structure can adequately support major plant structures that impart heavy to moderately-heavy foundation loads. The distribution and frequency of cavities encountered in the subsurface boring program appear to define continuous zones of solutioning, and weathering along bedding planes, at and below the contact between the near surface massive dolomite and the lower thinly bedded dolomite. A widespread and nonuniform distribution of soft and

weathered material along bedding planes could cause structures to experience differential settlement, cracking of foundations, and overstressing of pipes and structural components. Solution features are unpredictable and can have significant impact on structure performance. Therefore, it is prudent to excavate such solutioned zones where feasible or, where not feasible, to expose them by excavation to permit their direct inspection and evaluation. Excavation is recommended because the zone of significant solutioning is defined and is of shallow depth. Removal of the capping rock will permit direct inspection of the solutioning encountered in the lower, thinly-bedded dolomite. Experience has shown that actual field conditions in carbonate rock are often worse than anticipated, even when a relatively large number of borings have been made. We concluded that the solutioned rock zone in these foundation areas should be removed to elevation 545 feet and be replaced with compacted engineered fill or fill concrete. In response to this conclusion, the applicant has committed, by letter dated March 7, 1978, to overexcavate and backfill with concrete those foundation areas for major structures that will be founded above elevation 545 feet International Great Lakes Datum.

Under dynamic loading, the maximum foundation contact stress is estimated to be 15 kips per square foot for the containment building and seven kips per square foot for the auxiliary building.

The service water intake structure will be an extension of the intake structure for Unit 1. It will be founded on a mat foundation in rock at elevation 543 feet, which is below the zone of solution cavities. The combined static and dynamic loading for this structure is reported to impose a maximum contact stress, on the rock foundation, of 26 kips per square foot.

The ultimate heat sink pump house structure will be founded on a common mat on two levels. The intake area will be founded on rock at about elevation 540 feet, while the area that houses the service water piping leading to the pumps will be founded at about elevation 562 feet. The applicant originally proposed to grout the foundation rock between elevations 540 feet and 562 feet. However, in response to our position that this was unacceptable, the applicant has committed to overexcavate the foundation rock to elevation 545 feet and replace it with fill concrete, as in the case of the shallow foundation levels of the auxiliary building. Under dynamic loading, the maximum foundation contact stress is estimated to be 15 kips per square foot.

Based upon the subsurface conditions described in the Preliminary Safety Analysis Report and upon the applicant's commitments, we conclude that the foundation rock can adequately support the proposed structures discussed above, under the specified foundation loads. We expect that settlement of these structures will be negligible.

The nonseismic Category I turbine and office buildings foundations will be treated by grouting. The impact on safety, from settlement of these structures, is not

known. Therefore, we required and the applicant agreed to monitor the settlement of these structures.

Other Seismic Category I Structures

The electrical and service water manholes will have mat foundations supported on glacial till, glaciolacustrine soil, or structural fill. Each foundation will be drilled to verify the competency of subsurface conditions. We find this acceptable.

The service water piping will be founded on seismic Category I structural backfill extending to glacial till. The recent glaciolacustrine soils above the till will be removed. We find this to be acceptable.

The service water valve rooms in the intake structure will have mat foundations founded on rock, or on fill concrete extending to rock, between elevations 561 and 567 feet. The maximum combined static and dynamic foundation loading of these structures is reported to be four kips per square foot.

The borated water storage tanks, which are each 47 feet in diameter, will be founded east of the containment structure on a mat foundation at elevation 585 feet. The tanks will be supported by about 25 feet of engineered fill extending to the top of rock. The maximum combined static and dynamic loading for this structure is reported to be 7.5 kips per square foot.

The applicant estimates that the settlement of the service water piping, the service valve rooms, the borated water storage tanks, and the manholes, which impose light foundation loads, will be small. However, because the foundation rock in these areas will be treated by grouting, we required that the applicant monitor the settlement of each of these structures during construction. Observed settlement during construction will provide a direct means of evaluating the effectiveness of the grouting program and for predicting the performance of foundations during plant operation. The applicant has committed to carrying out a settlement monitoring program for these structures.

We conclude, based upon the subsurface conditions described by the applicant, that the foundations for the service water piping, the service water valve rooms and the borated water storage tanks will be able to support adequately the proposed structures under the specified foundation loads.

2.6.5 Earth Construction

The lower part of the ultimate heat sink pond for Units 2 and 3 will consist of a rock quarry that is below the normal groundwater level and is naturally filled with water. A 3,000-foot long earth embankment will be constructed around the periphery

of the quarry to impound water to the natural groundwater level of the site, elevation 571 feet International Great Lakes Datum. A preliminary plan, with sections of the quarry and embankment slope, was submitted for our review as part of the Preliminary Safety Analysis Report, Revision 19.

The Preliminary Safety Analysis Report does not contain enough detailed information, regarding the conditions of the quarry slopes, to permit us to complete our review of the design of the dike and rock slopes of the quarry. We are particularly interested in the stability of the slopes in the vicinity of the water intake. We notified the applicant of our concerns, and he has committed to provide a final report, for our review, on the conditions of the rock quarry slopes within 200 feet of the ultimate heat sink pump house. This report will be available during construction, once the area has been dewatered. We find this acceptable at the construction permit stage because remedial measures can be taken, if needed, during construction to stabilize the rock slopes.

3.0 DESIGN CRITERIA FOR STRUCTURES, SYSTEMS AND COMPONENTS

3.1 Conformance with General Design Criteria

In Preliminary Safety Analysis Report Section 3.1.1, the applicant provides a discussion of how Davis-Besse Units 2 and 3 conform to each of the General Design Criteria for Nuclear Power Plants, Appendix A to 10 CFR Part 50. In that discussion, the applicant identifies several deviations from General Design Criteria 55 and 56 relating to the design arrangement of certain isolation valves of the containment isolation system. We have reviewed these deviations, and our evaluation and acceptance is given in Section 6.2.3 of this report.

On the basis of our review of the discussion of each of the General Design Criteria in Preliminary Safety Analysis Report Section 3.1.1, and our review of other sections of the Preliminary Safety Analysis Report, we conclude that the proposed facility will be designed, constructed and operated to meet the requirements of the General Design Criteria, with the exceptions noted in the preceding paragraph.

3.2 Classification of Structures, Components and Systems

3.2.1 Seismic Classification

Criterion 2 of the General Design Criteria requires that nuclear power plant structures, systems, and components important to safety be designed to withstand the effects of earthquakes without loss of capability to perform their safety function. These plant features are those necessary to assure (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shut down the reactor and maintain it in a safe shutdown condition, and (3) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10 CFR Part 100.

Structures, systems, and components important to safety that will be designed to withstand the effects of a safe shutdown earthquake and remain functional have been identified in the Preliminary Safety Analysis Report in an acceptable manner and classified as seismic Category I items, in conformance with Regulatory Guide 1.29, "Seismic Design Classification." As an alternate to including seismic Category I component cooling water lines to the reactor coolant pumps, the applicant has committed to providing safety grade instrumentation that will sense loss of cooling water flow to each pump and will shut down the affected pumps. We find this to be an acceptable alternate to the provisions of Regulatory Guide 1.29 (see Section 9.2.2 of this report). All other structures, systems, and components that may be required for operation of the facility will be designed to other than seismic Category I requirements. Included in this classification are those portions of

otherwise seismic Category I systems that will not be required to perform a safety function. Structures, systems, and components important to safety, that will be designed to withstand the effects of a safe shutdown earthquake and remain functional, are identified in an acceptable manner in Section 3.2.1 of the Preliminary Safety Analysis Report.

In our review, the basis for acceptance has been conformance of the applicant's designs, design criteria and design bases, for structures, systems, and components important to safety, with the Commission's regulations as set forth in General Design Criterion 2, and with Regulatory Guide 1.29 and industry codes and standards.

We conclude that structures, systems and components important to safety, that will be designed to withstand the effects of a safe shutdown earthquake and remain functional, have been properly classified as seismic Category I items in conformance with the Commission's regulations, the applicable regulatory guides, and industry codes and standards, and are, therefore, acceptable. Design of these items in accordance with seismic Category I requirements provides reasonable assurance of (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shut-down the reactor and maintain it in a safe shutdown condition, and (3) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of 10 CFR Part 100.

3.2.2 System Quality Group Classification

Criterion 1 of the General Design Criteria requires that nuclear power plant systems and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.

We have reviewed the applicant's classification system for pressure-retaining components, such as pressure vessels, heat exchangers, storage tanks, pumps, piping, and valves, in fluid systems important to safety and the assignment by the applicant of safety classes to those portions of systems required to perform safety functions.

The applicant has applied Quality Groups A, B, C and D, defined in Regulatory Guide 1.26, "Quality Group Classifications and Standards," to those fluid-containing components that will be part of the reactor coolant pressure boundary or of other fluid systems important to safety, where reliance is placed on these systems (1) to prevent or mitigate the consequences of accidents and malfunctions originating within the reactor coolant pressure boundary, (2) to permit shutdown of the reactor and maintain it in a safe shutdown condition, and (3) to contain radioactive material. These fluid systems have been identified and classified in Tables 3.2-1 and 3.2-3 of the Preliminary Safety Analysis Report, and on system piping and instrumentation diagrams in the Preliminary Safety Analysis Report, in conformance with Regulatory Guide 1.26.

Fluid system pressure-retaining components important to safety that are classified Quality Group A, B, or C will be constructed to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code as follows:

<u>Quality Group</u>	<u>Component Code</u> <u>ASME Section III, Division 1</u>
A	Class 1
B	Class 2
C	Class 3

Quality Group A components will comply with Section 50.55a of 10 CFR Part 50. Quality Groups B and C components will comply with Subsection NA-1140 of the ASME Code.

Components that are classified Quality Group D will be constructed to the following codes as appropriate: ASME Boiler and Pressure Vessel Code, Section VIII, Division 1, and American National Standards Institute (ANSI) Standard B31.1-1973.

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

The applicant's proposed means for isolating the steam generators, in the unlikely event of a main steam or feedwater line rupture, include reliance on valves classified Quality Group D. As discussed in Section 6.2.1 of this report, we require certain valves in the main feedwater and main steam lines to be Quality Group B pending completion of the staff's generic study, "PWR Main Steam Line Break -- Core, Reactor Vessel, and Containment Building Response." Subject to resolution of the classification of these valves, we conclude that fluid system pressure-retaining components important to safety will be designed, fabricated, erected and tested to quality standards that are in conformance with the Commission's regulations, the applicable regulatory guides, and industry codes and standards, and are, therefore, acceptable.

3.3 Wind and Tornado Design

3.3.1 Wind Loadings

All seismic Category I structures exposed to wind forces will be designed to withstand the effects of the design wind. The design wind specified by the applicant has a velocity of 90 miles per hour, based on a recurrence period of 100 years.

The procedures that will be used to transform the wind velocity into pressure loadings on structures, and the associated vertical distribution of wind pressures and gust factors, are in accordance with ASCE Paper No. 3269. We have reviewed this document and have found the methods therein to be acceptable.

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

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

3.3.2 Tornado Loadings

In accordance with Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants," for Region I, all seismic Category I structures exposed to tornado forces will be designed to resist a tornado of 290 miles per hour rotational wind velocity and a 70 miles per hour translational wind velocity. The simultaneous atmospheric pressure drop will be three pounds per square inch in 1.5 seconds.

The procedures that will be used to transform the tornado wind velocity into pressure loadings are similar to those used for the design wind loadings as discussed in Section 3.3.1 of this report. The tornado missile effects will be determined using procedures discussed in Section 3.5 of this report. The total effect of the design tornado on seismic Category I structures will be determined by appropriate combinations of the individual effects of the tornado wind pressure, pressure drop, and tornado-associated missiles. Structures will be arranged on the plant site and protected in such a manner that collapse of structures not designed for the tornado will not affect other safety-related structures.

The procedures utilized, to determine the loadings on structures induced by the design basis tornado specified for the plant, are acceptable since the procedures provide a conservative basis for engineering design to assure that the structures will withstand such environmental forces.

The use of these procedures provides reasonable assurance that, in the event of a design basis tornado, the structural integrity of the plant structures that have to be designed for tornadoes will not be impaired and, consequently, safety-related systems and components located within these structures will be adequately protected and may be expected to perform their necessary safety functions. Conformance with

these procedures is an acceptable basis for satisfying the applicable requirements of General Design Criterion 2.

3.4 Water Level (Flood) Design

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 of this report.

The maximum probable static water level for the plant site was calculated to be 583.7 feet International Great Lakes Datum. The maximum water level due to wave runup on the wave protection dike, installed along the north, east and, partially, on the south side of the site, was calculated to be 590.5 feet International Great Lakes Datum. Protection against wave runup for the plant will be by the dike, the top of which will be at elevation 591.0 feet International Great Lakes Datum.

The seismic Category I structures for Units 2 and 3 will have no openings below grade elevation 585.0 feet International Great Lakes Datum. All structures will be protected below grade by waterproof membranes and water stops. Consequently, the design maximum flood level of 583.7 feet International Great Lakes Datum will be adequate to protect the plant structures and safety-related equipment installed therein from external flood water and is, therefore, acceptable.

The hydrostatic effect of the design flood level will be considered in the design of all seismic Category I structures exposed to the water head.

The grades of all the seismic Category I structures will be located above the maximum probable flood elevation, and the wave protection dike will be provided on the lake sides of the station. We, therefore, conclude that phenomena, such as flood current, wind wave or seiches that are associated with dynamic water forces, will not affect the plant structures.

The procedures to be utilized to determine the loadings on seismic Category I structures induced by the design flood or highest groundwater level specified for the plant are acceptable because these procedures provide a conservative basis for engineering design to assure that the structures will withstand such environmental forces.

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

3.5 Missile Protection

3.5.1 Missile Selection and Protection Criteria

The facility design requirements consider the possibility of missiles being generated from pressurized piping and vessels, rotating equipment, and tornadoes. Protection of safety-related components and structures will be provided by orientation and separation from missile generating sources, and by the use of adequate barrier or energy absorbing materials. Engineered safety feature systems will be separated in a manner such that the failure of one train cannot cause the failure of the other, or that the failure of any plant component which brings about the need for these engineered safety feature systems does not render the safety system inoperative.

We have concluded that nearby industrial, transportation, and military facilities do not represent a hazard requiring protective features in the facility design. See Section 2.2 of this report. We have categorized the other missiles considered by the applicant as (1) missiles generated by postulated failures of facility equipment, (2) missiles generated by postulated failure of the turbine generator, and (3) missiles generated by postulated tornadoes. The results of our review of the applicant's missile selection is as follows:

(1) Facility Equipment-Generated Missiles

Missiles that could be generated by postulated failures of equipment within the containment are listed in Preliminary Safety Analysis Report Section 3.5.2.1 and Table 3.5-1. These include appurtenances to pressurized systems, e.g., nuts, bolts, studs, control and drive assemblies and instrumentation nozzles. The possibility of missiles being generated due to overspeed of the reactor coolant pump is being reviewed by the staff as a generic issue (see Section 5.4.1 of this report).

The applicant has tabulated a list of characteristics of selected equipment-generated missiles. Additional equipment-generated missiles may be identified during the detailed plant design. The method of characterization of missiles and a classification scheme is described. The methodology to be used for predicting missile characteristics including equations, is provided.

As a result of our review, we conclude that the applicant's proposed design criteria and bases are in conformance with General Design Criterion 4 as it relates to structures that house essential systems and to the systems being capable of withstanding the effects of internally-generated missiles; with Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," as it relates to protection of spent fuel pool systems and spent fuel assemblies from internal missiles; and with Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants," as it relates to the design of the intake structure to withstand the effects of internal missiles, and are, therefore, acceptable.

(2) Turbine Missiles

The steam turbine-generators will be arranged in non-peninsular orientation with respect to their respective reactor containments. The applicant indicates that the probability of unacceptable consequences due to low-trajectory turbine missiles is greater than 1×10^{-7} per turbine year. Our review indicates that the only safety-related target that is vulnerable to direct-strike turbine missiles is the upper portion (above elevation 660 feet) of the reactor containment building. This building will consist of a 2-1/2-foot thick reinforced concrete outer wall and a free standing inner steel shell. Based on information presently available, we estimate the probability of a turbine missile damaging primary system piping and equipment to be about 1.4×10^{-6} per turbine year.

The value of 1.4×10^{-6} is greater than our current acceptance criteria (1×10^{-7}). Therefore, we have asked the applicant to prepare and submit a more detailed strike and damage analysis, with respect to destructive overspeed turbine missiles penetrating the upper portions of the reactor containment wall. This analysis will determine the need for additional measures to protect the primary system piping and equipment. Penetration of structures and other missile barriers will be evaluated on the basis of the review procedures outlined in Standard Review Plan 3.5.3, "Barrier Design Procedures," and on the most penetrating missile as described by the turbine vendor.

The applicant has stated, in a letter dated September 16, 1977, that the turbine missile strike and damage probability analysis will be performed at a future time, when data from turbine missile tests are made available. These tests are being performed under the sponsorship of the Electric Power Research Institute. The report on the probability analysis is expected to be completed and submitted for our review in late 1978 or early 1979. On the basis of this report, the staff will determine the extent of the protection needed against turbine missiles. The applicant has committed to provide the necessary protection as dictated by the analysis and the staff review thereof. Furthermore, the applicant committed to continue to review the physical features of the plant to assure that the design and arrangement will be such as to permit incorporation of any structural barriers that might be found necessary to assure an acceptably low level of damage from potential turbine missiles.

However, we require the applicant to commit that, should the results of the Electric Power Research Institute tests, or a generic resolution of missile penetration modeling, be unavailable prior to construction of affected structures, the protection requirements against turbine missiles will be evaluated in terms of the current concrete penetration modeling involving the

use of equations that are commonly referred to as the "modified NDRC formulas" and that are given as Equations 5 through 10 in "A Review of Procedures for the Analysis and Design of Concrete Structures to Resist Missile Impact Effects," by R. P. Kennedy. We will report the resolution of this matter in a supplement to this report.

(3) Tornado Missile Protection

Criterion 2 of the General Design Criteria requires that structures, systems and components important to safety be designed to withstand the effects of natural phenomena, such as tornadoes, without loss of capability to perform their safety functions. In consideration of this requirement we have specified a spectrum of selected design basis missiles that we conclude are acceptable for the design of tornado missile barriers in NUREG-075/087, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 3.5.1.4.

The applicant originally proposed missile velocities that are the same as those given in Standard Review Plan 3.5.1.4, Revision 1, with the exception of the automobile, where the applicant proposed a lower velocity. At our request, the applicant has committed, in Preliminary Safety Analysis Report Revision 13, to designing Davis-Besse Units 2 and 3 to provide protection against the NRC tornado missile spectrum given in Standard Review Plan 3.5.1.4, Revision 1. Compliance with the NRC spectrum given in Standard Review Plan 3.5.1.4, Revision 1, is acceptable for preventing missile penetration and spalling and to assure acceptable overall structural response.

3.5.2 Barrier Design Procedures

The seismic Category I structures, systems and components will be shielded from, or designed to withstand, 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 in the Preliminary Safety Analysis Report, indicating that the procedures, that will be 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 will be accomplished in two steps. In the first step, the potential damage that could be done by the missile, in the immediate vicinity of impact, will be investigated. This will be accomplished by estimating the depth of penetration of the missile into the impacted structure. Furthermore, secondary missiles will be prevented by fixing the target thickness well above that determined for penetration. In the second step

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of the analysis, the overall structural response of the target, when impacted by a missile, will be 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, will be combined with other applicable loads, as is discussed in Section 3.8 of this report.

The applicant committed to using a ductility ratio of 10 or less for analysis of structures impacted by missiles. The staff required that the applicant extend this commitment to impactive and impulsive loads except compartment pressurization loads. For loads resulting from compartment pressurization, the elastic analysis should be applied. The applicant has committed to meeting these requirements.

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

The use of these procedures provides reasonable assurance 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 will, therefore, be adequately protected against the effects of missiles and may be expected to perform their intended safety functions. Conformance with these procedures is an acceptable basis for satisfying the applicable requirements of General Design Criteria 2 and 4.

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

The provisions for protection against the dynamic effects of postulated pipe ruptures and the resulting discharge of fluid provide that, in the event of the occurrence of the combined loadings imposed by an earthquake of the magnitude specified for the safe shutdown earthquake and a concurrent single pipe break of the largest pipe at one of the design basis break locations inside containment, the following conditions and safety functions will be accommodated and assured:

- (1) The magnitude of the design basis loss-of-coolant accident cannot be aggravated by potential multiple failures of piping.
- (2) The reactor emergency core cooling systems can be expected to perform their intended function, assuming a single failure.

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(3) Systems and components important to safety will be adequately protected.

The analytical method for determining pipe motion subsequent to rupture, and the pipe restraint dynamic interaction, as described in the applicant's Preliminary Safety Analysis Report, is sufficiently detailed to reflect the structural characteristics of the piping system.

The design criteria to be used for identifying high energy fluid piping, and for postulating pipe break locations and flow areas, will be consistent with the criteria set forth in Regulatory Guide 1.46, "Protection Against Pipe Whip Inside Containment," and we find them to be acceptable.

On the basis of our review, we have concluded that the criteria used for the identification, design and analysis of piping systems inside containment, where postulated breaks may occur, constitute an acceptable design basis for satisfying the applicable requirements of Criteria 1, 2, 4, 14 and 15 of the General Design Criteria and, therefore, are acceptable.

3.6.2 Outside Containment

The proposed design should accommodate the effects of postulated pipe breaks and cracks in high energy fluid piping systems outside containment, with respect to pipe whip, jet impingement and resulting reaction forces, and environmental conditions. The general arrangement and the layout of high energy systems may utilize combinations of physical separation, pipe enclosures, pipe whip restraints, and equipment shields to protect against adverse dynamic effects of pipe rupture.

The criteria to be followed in the design of the piping systems and associated components and structures should be consistent with the criteria contained in Branch Technical Positions APCS 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," and MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," and with Regulatory Position C.2 of Regulatory Guide 1.29, "Seismic Design Classification," Revision 2, February 1976.

We have reviewed the proposed arrangement of steam and feedwater piping outside containment to evaluate the adequacy of protection of safety-related equipment against the effects of pipe breaks in these lines. This piping passes close to the control room, the cable spreading room and the spent fuel storage pool. We have also reviewed the applicant's criteria for protection against breaks in these high energy lines. The criteria are unacceptable because they fail to meet the requirements of both Branch Technical Position APCS 3-1 and Position C.2 of Regulatory Guide 1.29. We have notified the applicant of our position by our letter of March 17, 1978. We will review his response when it is received, and will report the results of our evaluation in a supplement to this report.

3.7 Seismic Design

3.7.1 Seismic Input

The input seismic design response spectra, for the safe shutdown and operating basis earthquakes, to be applied at the bottom of the foundations in the design of seismic Category I structures, systems and components, comply with the recommendations of Regulatory Guide 1.60, "Design Response Spectra for Nuclear Power Plants." The specific percentages of critical damping values to be used in the seismic analysis of Category I structures, systems and components, are in conformance with Regulatory Guide 1.61, "Damping Values for Seismic Analysis of Nuclear Power Plants."

The synthetic time history to be used for seismic design of seismic Category I plant structures, systems and components, will be adjusted in amplitude and frequency to obtain response spectra that envelope the design response spectra specified for the site.

Conformance with the recommendations of Regulatory Guides 1.60 and 1.61 provides reasonable assurance that, for an operating basis earthquake whose intensity is .08g and for a safe shutdown earthquake whose intensity is 0.20g, the seismic inputs to seismic Category I structures, systems and components are adequately defined to assure a conservative basis for the design of such structures, systems and components to withstand the consequent seismic loadings.

We conclude that the applicant's proposed seismic input criteria are acceptable for seismic design, and that compliance with Regulatory Guides 1.60 and 1.61 is an acceptable basis for satisfying the applicable requirements of Criterion 2 of the General Design Criteria.

3.7.2 Seismic System and Subsystem Analysis

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

The system and subsystem analyses will be performed by the applicant on an elastic basis. Modal response spectrum multidegree of freedom and time history methods will form the bases for the analysis of all seismic Category I structures, systems and components. When the modal response spectrum method is used, governing response

parameters will be combined by the square root of the sum-of-the-squares rule. The absolute sum of the modal responses, or its equivalent, will be used for modes with closely spaced frequencies. The square root of the sum-of-the-squares of the maximum codirectional responses will be used in accounting for three components of the earthquake motion for both the time history and response spectrum methods. Floor spectra inputs to be used for design and test verifications of structures, systems, and components will be generated from the time history method, taking into account variation of parameters by peak widening. A vertical seismic system dynamic analysis will be employed for all structures, systems, and components where analysis shows significant structural amplification in the vertical direction. Torsional effects and stability against overturning will be considered.

The seismic analysis of piping will be performed using the criteria outlined in Bechtel topical report BP-TOP-1, "Seismic Analysis of Piping Systems," Revision 3, dated January 1976. This document has been reviewed and approved by the staff.

We conclude that the seismic system and subsystem analysis procedures and the criteria proposed by the applicant provide an acceptable basis for the seismic design of seismic Category I structures, systems and components.

3.7.3 Seismic Instrumentation

The installation of seismic instrumentation in the reactor containment structure and at other seismic Category I structures, systems, and components will permit the recording of data on seismic ground motion as well as data on the frequency and amplitude relationship of the response of major structures and systems. A prompt readout of pertinent data at the control room can be expected to yield sufficient information to guide the operator on a timely basis for the purpose of evaluating the seismic response in the event of an earthquake. Data to be obtained from such installed seismic instrumentation will be sufficient to determine that the seismic analysis assumptions and the analytical model used for the design of the plant are adequate and that allowable stresses are not exceeded under conditions where continuity of operation is intended.

The applicant originally proposed a seismic instrumentation system that contains many exceptions to Regulatory Guide 1.12, "Instrumentation for Earthquakes." Peak strain gauges were to be used in place of triaxial peak accelerographs, and response spectrum recorders are not to be supplied as discrete instruments.

The proposed system did not provide the information needed for evaluating the seismic response of the plant in the event of an earthquake.

We informed the applicant that the proposed system is not acceptable. By letter dated February 28, 1978, the applicant committed to providing a seismic instrumentation system that conforms to the recommendations of Regulatory Guide 1.12.

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 are described in the Preliminary Safety Analysis Report. Together with the above-mentioned commitment, they comply with the recommendations of Regulatory Guide 1.12. On this basis, the proposed seismic instrumentation program is acceptable.

3.8 Design of Seismic Category I Structures

3.8.1 Containment

The containment will consist of a free-standing steel shell located within a separate reinforced concrete reactor building. The containment will be 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 to be included in the design will consist of an appropriate combination of dead and live loads; thermal loads; and seismic and loss-of-coolant accident induced loads, including pressure and jet forces.

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

The criteria that will be used in the analysis, design, and construction of the steel containment structure, for anticipated loadings and postulated conditions that may be imposed upon the structure during its service lifetime, are in conformance with established criteria, codes, standards and guides that are acceptable to us.

The use of these criteria as defined by applicable codes, standards and guides; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control programs, and special construction techniques; and the testing and in-service surveillance requirements, provide reasonable assurance that, in the event of earthquakes and of various postulated accidents occurring inside and outside the containment, the structure will withstand the specified conditions without impairment of structural integrity or safety function. A seismic Category I concrete shield building will protect the steel containment from the effects of wind, tornadoes, and various postulated accidents occurring outside the shield building. Conformance with these criteria constitutes an acceptable basis for satisfying the applicable requirements of General Design Criteria 2, 14, 16 and 50.

3.8.2 Concrete and Structural Steel Internal Structures

The containment internal structures will consist of a shield wall around the reactor, secondary shield walls and other interior walls, compartments and floors. The major code to be used in the design of concrete internal structures is

ACI 318-71, "Building Code Requirements for Reinforced Concrete." For steel internal structures, the AISC specification, "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," will be used.

The containment concrete and steel internal structures will be 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 to which each structure may be subjected, and include all loads that may act simultaneously. The design and analysis procedures that will be used for the internal structures are the same as those approved on previously licensed applications, and are in accordance with procedures delineated in the ACI 318-71 code and the AISC specification, for concrete and steel structures, respectively.

The containment internal structures will be designed and proportioned to remain within limits established by us under the various load combinations set forth in Standard Review Plan 3.8.3. These limits are based on the ACI 318-71 code and the AISC specification, for concrete and steel structures, respectively, modified as appropriate for load combinations that are considered extreme.

The materials of construction and their fabrication, construction and installation, will be in accordance with the ACI 318-71 code and the AISC specification.

The criteria that will be used in the design, analysis and construction of the containment internal structures, to account for anticipated loadings and postulated conditions that may be imposed upon the structures during their service lifetime, are in conformance with established criteria, and with codes, standards, and specifications acceptable to us.

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

3.8.3 Other Seismic Category I Structures

Seismic Category I structures, other than containment and its interior structures, but including the shield building that surrounds the containment, will all be of structural steel and concrete. The structural components will consist of slabs, walls, beams and columns. The major code to be used in the design of concrete

seismic Category I structures is ACI 318-71, "Building Code Requirements for Reinforced Concrete." For seismic Category I steel structures, the AISC specification, "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," will be used.

The concrete and steel seismic Category I structures will be designed to resist various combinations of dead loads; live loads; environmental loads including winds and tornadoes; loads due to the safe shutdown earthquake and the operating basis earthquake; 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 will be used for these seismic Category I structures are the same as those approved on previously licensed applications, and are in accordance with procedures delineated in the ACI 318-71 code and in the AISC specification, for concrete and steel structures, respectively.

The various seismic Category I structures will be designed and proportioned to remain within limits established by us under the various load combinations set forth in Standard Review Plan 3.8.4. These limits are based on the ACI 318-71 code and the AISC specification, for concrete and steel structures, respectively, modified as appropriate for load combinations that are considered extreme.

The materials of construction, and their fabrication, construction and installation, will be in accordance with the ACI 318-71 code and the AISC specification.

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

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

3.8.4 Foundations

Foundations of seismic Category I structures are described in Section 3.8.5 of the Preliminary Safety Analysis Report. The foundations of major structures will be reinforced concrete of the mat type. The major code to be used in the design of these concrete mat foundations is ACI 318-71. These concrete foundations will be designed to resist various combinations of dead loads; live loads; environmental loads including winds, tornadoes, and earthquakes, and loads generated by postulated ruptures of high energy pipes.

The design and analysis procedures that will be used for these seismic Category I foundations are the same as those approved on previously licensed applications, and are in accordance with procedures delineated in the ACI 318-71 code. The various seismic Category I foundations will be designed and proportioned to remain within limits established by us under the various load combinations set forth in Standard Review Plan 3.8.5. These limits are based on the ACI 318-71 code modified as appropriate for load combinations that are considered extreme. The materials of construction, and their fabrication, construction and installation, will be in accordance with the ACI 318-71 code.

The criteria that will be used in the analysis, design, and construction of all the seismic Category I foundations, to account for anticipated loadings and postulated conditions that may be imposed upon each foundation during its service lifetime, are in conformance with established criteria, codes, standards, and specifications acceptable to us.

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

3.9 Mechanical Systems and Components

3.9.1 Dynamic System Analysis and Testing

The applicant will perform a piping preoperational vibration dynamic effects test program to check the vibration performance of piping important to safety. The test program will be conducted, during startup and initial operation, on safety-related ASME Code Class 1, 2 and 3 piping systems and all other high energy piping systems; on seismic Category I portions of moderate energy piping systems located outside

containment; and on their restraints. We find the preoperational vibration effects test program to be acceptable for the reasons stated below.

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

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

With regard to flow-induced vibration testing of reactor internals, the applicant has referenced Oconee Nuclear Station Unit 1 (Docket No. 50-269) as the prototype reactor on which the instrumented prototype vibrational testing has been performed. The test program, which is described fully in Topical Reports BAW-10038, "Prototype Vibration Measurement Program for Reactor Internals - 177 Fuel Assembly Plant," Revision 1, and BAW-10039, "Prototype Vibration Measurement Results for 177 - Fuel Assembly Two Loop Plant," meets the recommendations of Regulatory Guide 1.20, "Vibration Measurements on Reactor Internals." The core mechanical design and flow characteristics of the proposed Davis-Besse Units 2 and 3 are considered sufficiently similar to those of Oconee Unit 1 to justify the designation of Oconee 1 as a valid prototype. Any minor differences in the design in order to accommodate the 17x17 fuel assemblies will be identified and fully justified in the Final Safety Analysis Report, when more detailed design information will be available.

The preoperational vibration assurance program, as planned for the reactor internals, provides an acceptable basis for verifying the design adequacy of these internals under test loading conditions that will be comparable to those experienced during operation. We conclude that the combination of tests, predictive analysis and post-test inspection will provide adequate assurance that the reactor internals can withstand the flow-induced vibrations of reactor operations without loss of structural integrity during their service life.

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The conduct of the preoperational vibration test constitutes an acceptable basis for demonstrating design adequacy of the reactor internals in fulfilling the applicable requirements of General Design Criteria 1 and 4 and in conforming with the recommendations of Regulatory Guide 1.20.

3.9.2 Analysis Methods for Loss-of-Coolant Accident Loadings

On May 7, 1975, we were informed by a licensee of a pressurized water reactor, Virginia Electric and Power Company, that the asymmetric loading resulting from a postulated pipe rupture in the reactor coolant system had not been taken into account in the original design of the reactor pressure vessel support system for the North Anna Units 1 and 2 (Docket Nos. 50-338 and 339).

The loading results from the forces induced on the reactor internals caused by differential pressure conditions within the vessel immediately following a postulated loss-of-coolant accident. In addition, the asymmetric loading, from transient differential pressures that would exist around the exterior of the reactor vessel from the same postulated pipe rupture, was not included in the original design analysis. However, the symmetric loadings from such a postulated pipe rupture were included in the original analysis of the reactor pressure vessel supports.

We have asked the applicant to provide additional information that addresses the analysis of these loadings. We have not completed our review of the applicant's response. We will report on this matter in a supplement to this report.

3.9.3 ASME Code Class 2 and 3 Components and Supports

All seismic Category I pressure retaining systems, components and equipment outside of the reactor coolant pressure boundary, including active pumps and valves, will be designed to sustain normal loads, anticipated transients, the operating basis earthquake, and the safe shutdown earthquake, within stress limits that are comparable to those outlined in Regulatory Guide 1.48, "Design Limits and Loading Combinations for Seismic Category I Fluid System Components." The specified design basis combinations of loading, as applied to the design of the safety-related ASME Code Class 2 and 3 pressure-retaining components and supports in systems classified as seismic Category I, will provide reasonable assurance that in the event (1) an earthquake should occur, or (2) an upset, emergency or faulted plant transient should occur during plant operation, the resulting combined stresses imposed on the system components will not exceed the allowable design stress and strain limits for the materials of construction. Limiting the stresses under such loading combinations will provide a conservative basis for the design of the system components to withstand the most adverse combinations of loading events without an unacceptable loss of structural integrity. The design load combinations and associated stress and deformation limits specified for all ASME Code Class 2 and 3 components and

supports, including the active pumps and valves, constitute an acceptable basis for design in satisfying Criteria 1, 2 and 4 of the General Design Criteria.

The applicant will conduct component test programs, supplemented by analytical predictive methods, to provide assurance that active pumps and valves can withstand postulated seismic loads, in combination with other significant loads, without loss of structural integrity, and can perform the "active" function (i.e., valve closure or opening, or pump operation), when a safe plant shutdown is to be effected or the consequences of an accident are to be mitigated. A commitment to develop and utilize a component operability assurance program satisfactory to the staff constitutes an acceptable basis for implementing the requirements of General Design Criterion 1 as related to operability of ASME Code Class 2 and 3 active pumps and valves.

The criteria to be used in developing the design and mounting of the safety and relief valves of ASME Code Class 2 systems will provide adequate assurance that, under discharging conditions, the resulting stresses will not exceed the allowable design stress and strain limits for the materials of construction. Limiting the stresses, under the loading combinations associated with the actuation of these pressure relief devices, provides a conservative basis for the design of the system components to withstand these loads without loss of structural integrity and impairment of the overpressure protection function.

The criteria to be used for the design and installation of overpressure protection devices in ASME Code Class 2 systems are consistent with Regulatory Guide 1.67, "Installation of Overpressure Protection Devices," and constitute an acceptable design basis for meeting the applicable requirements of Criteria 1, 2, 4, 14 and 15 of the General Design Criteria.

3.9.4 Control Rod Drive Systems

The design criteria, and the testing program that will be conducted to verify the mechanical operability and life cycle capabilities of the reactivity control system, are in conformance with established criteria, codes, standards and specifications, and we find them to be acceptable. The use of these criteria provide reasonable assurance that the system will function reliably when required, and form an acceptable basis for satisfying the mechanical reliability requirements of Criterion 21 of the General Design Criteria.

3.9.5 Inservice Testing of Pumps and Valves

The applicant has stated that the inservice test program for all Code Class 1, 2 and 3 pumps and valves will meet the requirements of the ASME Boiler and Pressure Vessel Code, Section XI, Subsections IWP and IWV, respectively. We conclude that the inservice inspection program for this equipment is consistent with the requirements

of 10 CFR Part 50.55a and constitutes an acceptable basis for satisfying the applicable portions of General Design Criteria 37, 40, 43 and 46.

3.10 Seismic Qualification of Seismic Category I Electrical, Instrumentation, and Control Equipment

Instrumentation and electrical components required to perform a safety function will be designed to meet seismic Category I design criteria. Seismic requirements established by the seismic system analysis will be incorporated into equipment specifications to assure that the equipment purchased or designed will meet seismic requirements equal to or in excess of the requirements for seismic Category I components, either by appropriate analysis or by qualification testing.

The applicant has proposed a seismic qualification program that includes both the Babcock and Wilcox Company equipment and the equipment in the balance of plant.

The applicant has committed to utilizing the methods of Topical Report BAW-10082 (Part I), "Seismic Qualification Methods," which has been found acceptable by the staff for the equipment in the Babcock and Wilcox Company scope of supply.

With regard to the balance-of-plant equipment, the applicant has committed to a program in accordance with IEEE 344-1975, "Guide for Seismic Qualification of Class I Electric Equipment for Nuclear Power Generating Stations."

Based on the above commitments, we conclude that the seismic qualification program is acceptable.

3.11 Environmental Qualification of Electrical Equipment

The safety-related equipment located inside the containment, and required to function after a postulated loss-of-coolant accident or main steam line break, has been identified by the applicant. The most severe environmental conditions that will be imposed upon the equipment inside containment during and subsequent to a design basis accident are also listed with corresponding design values for these conditions. We require the qualification methodology to be in conformance with IEEE Standard 323-1974, "Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations."

The applicant has proposed an environmental qualification program that includes both the Babcock and Wilcox Company equipment and the equipment in the balance-of-plant scope.

For the Babcock and Wilcox Company scope of supply, the applicant has referenced Parts 2 and 3 of Topical Report BAW-10082, "Qualification Methods." However, these parts of the report are only in the draft stage and, as a result, do not demonstrate

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Babcock and Wilcox's conformance to IEEE 323-1974 as supplemented by Regulatory Guide 1.89, "Qualification of Class IE Equipment for Nuclear Power Plants." The applicant, therefore, has committed to the generic resolution to be achieved between us and Babcock and Wilcox on the topical report (Parts 2 and 3), while reserving the right to meet the qualification requirements of IEEE 323-1974 by implementing a program similar to that discussed below for balance-of-plant Class IE equipment if resolution is not reached on the referenced topical report by the Final Safety Analysis Report stage. We have been working with Babcock and Wilcox on the generic resolution, and we consider the progress made to date to be adequate to give reasonable assurance that resolution will be accomplished in time to permit application of the resolution to the qualification program. Therefore, we find this commitment acceptable.

With regard to the balance-of-plant equipment, the applicant has proposed to meet IEEE 323-1974 with a qualification program that is identical to the Gulf States River Bend Nuclear Power Plant proposal (Docket Nos. 50-458 and 50-459). The staff has already reviewed this proposal, and has found it acceptable for the Preliminary Safety Analysis Report stage of the River Bend application. Therefore, we conclude that the proposed qualification program is acceptable for Davis-Besse Units 2 and 3. However, the development of the program has not progressed as far as that for the Babcock and Wilcox scope of supply. Therefore, in order to assure that an acceptable qualification program, that is consistent with the objectives of IEEE Standard 323-1974, will be carried out for balance-of-plant Class IE equipment, the staff required that the applicant commit to provide a technical report one year prior to submittal of the Final Safety Analysis Report identifying (1) how each piece of Class IE equipment has been, or will be, qualified; (2) the acceptance criteria; (3) test procedures; and (4) test results, if available, or a schedule for submittal of test results. In a letter dated March 7, 1978, the applicant made such a commitment. Therefore, we conclude that the environmental qualification program can be implemented in an acceptable manner.

As noted above, the applicant has stated that all safety-related electrical equipment located within the containment will be qualified under the procedures of IEEE Standard 323-1974 to demonstrate availability for service in the most severe environment. However, on the basis of previous reviews, we have some concern with respect to the equipment that will be required to mitigate the consequences of a main steam line break. The applicant has stated that, in all cases, safety-related equipment located within the containment and required following a loss-of-coolant accident, will be qualified to the most severe environmental conditions for a loss-of-coolant accident. In addition, safety-related equipment that is required following a main steam line break will be qualified to the containment pressure and to the most severe temperature that exists for the particular piece of equipment during the time period that the equipment must operate.

Based on these commitments, we conclude that the criteria for the qualification of equipment inside containment are acceptable. Our evaluation of calculational models to be used to determine the environmental conditions is given in Section 6.2.1 of this report.

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4.0 REACTOR

4.1 Summary

Criterion 10 of the General Design Criteria requires that the reactor core and associated systems be designed to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. We have reviewed the information provided in the Preliminary Safety Analysis Report in support of the proposed reactor design. Our evaluation is contained in the following sections.

The facility nuclear steam supply system will be designed to operate at a maximum core thermal output of 2772 megawatts, with sufficient margin to allow for transient operation and instrument error, without causing damage to the core and without exceeding the pressure settings of the safety valves in the coolant system.

The core will be cooled and moderated by light water at a pressure of 2,235 pounds per square inch gauge. The reactor coolant will contain soluble boron for neutron absorption. The concentration of the boron will be varied, as required, to control relatively slow reactivity changes, including the effects of fuel burnup. Additional boron, in the form of burnable poison rods, will be employed to establish the desired initial reactivity. Part-length control rod assemblies may be used for axial power shaping, and full-length control rods will be used for reactor shutdown.

The design of the Davis-Besse Units 2 and 3 reactors is similar to that of Davis-Besse Unit 1 and of the Rancho Seco Nuclear Generating Station (Docket No. 50-312). We have approved both of the latter plants for operation. Each of these reactors produces 2772 thermal megawatts from a core consisting of 177 fuel assemblies.

The design of Davis-Besse Units 2 and 3 differs from that of the others in that Units 2 and 3 will use the Babcock & Wilcox Mark C fuel assemblies (17 x 17 rod matrix, 264 fueled rods) whereas the others use Mark B assemblies (15 x 15 matrix, 208 fueled rods). The 17 x 17 fuel design results in lower linear heat generation rates and fuel rod temperatures.

The design for Davis-Besse Units 2 and 3 provides four vent valves in the core support shield above the outlet nozzles. In the event of a cold leg break, these valves provide a flow path from the upper core region to the inlet annulus. Steam is vented from the upper plenum, thus reducing the backpressure and permitting a more rapid reflood of the core.

4.2 Fuel Design

4.2.1 Description

Davis-Besse Units 2 and 3 will contain the Mark C fuel assembly, which is an advanced design by Babcock & Wilcox incorporating the 17 x 17 square array of fuel rods. The basic Mark C fuel assembly will consist of 264 fuel rods, 24 control rod guide tubes, one instrumentation tube assembly, eight spacer grids and two end fittings. The guide tubes, spacer grids, and end fittings will form a structural cage for the rods and tubes. The guide tubes will be rigidly attached to the upper and lower end fittings. The upper end fitting will position the upper end of the fuel assembly in the upper grid plate structure, and will provide the means for the handling equipment to grasp the fuel assembly. Holddown springs and a spider are integral parts of each upper end fitting, serving to provide a positive holddown margin to oppose hydraulic forces. The lower end fitting will position the fuel assembly in the lower core grid plate. The lower end fitting grillage will provide a support surface for the bottom end of the fuel rod.

The spacer grids will be constructed from Inconel strips that are slotted and fitted together in an "egg-crate" fashion. Each grid has 36 strips -- 18 perpendicular to 18 -- which form the 17 x 17 lattice. The square cells formed by the interlaced strips will provide support for the fuel rods in two perpendicular directions through contact points on each wall of each cell.

The contacts will be in the form of protruding dimples, which are integrally punched from the strips on the walls of each square opening. On each of the two end spacer grids, the peripheral strip will be extended and mechanically attached to the respective end fitting.

The Zircaloy guide tubes will provide a guidance envelope for the control rods, which are moved in and out of the fuel assembly during operation. They also will provide the structural continuity for the fuel assembly. Threaded sleeves will be welded to each end of the guide tube. Guide tubes will be attached to each end fitting by nuts lock-welded to the sleeves. Guide tubes will be positioned between end fittings by the spacer grids.

The Zircaloy instrumentation tube will serve as a channel to guide, position, and contain the incore instrumentation within the fuel assembly. The instrumentation tube will be located on the centerline of the fuel assembly and will be axially retained at the lower end fitting. The spacer tubes will fit around the instrumentation tube between spacer grids and will restrict axial movement of the spacer grids.

Each fuel rod will be comprised of cladding, fuel pellets, end caps, and internal support components. The fuel will be in the form of sintered and ground pellets of low enriched uranium dioxide. Pellet ends will be dished and chamfered. The

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pellets will be inserted in Zircaloy-4 tubing that will be sealed by Zircaloy-4 end caps welded at each end of the tubes. All fuel rods will be internally pressurized with helium, which will reduce the differential pressure across the cladding and will improve heat transfer within the rods. The level of prepressurization is designed both to preclude cladding tensile stresses resulting from a net internal pressure and to reduce cladding flattening ("creep-down") during normal operation.

Above the fuel column in each rod will be a spring that separates the fuel from the fuel rod end cap. This spring will maintain the fuel column in place during shipping and handling. In operation, the spring permits thermal differential growth and axial movement between the fuel column and the cladding. Below each fuel column there will also be a spring that axially locates the bottom of the fuel column and separates the fuel from the lower fuel rod end cap. This spring is designed to deflect under high column loads to reduce axial strain in the cladding. This 2-spring, 2-plenum design is unique to Babcock & Wilcox fuel. The other United States fuel manufacturers use only an upper plenum and spring. We have reviewed this design with Babcock & Wilcox and have found this feature acceptable.

The Mark C fuel assembly design (17 x 17 array) is mechanically similar to the Babcock & Wilcox Mark B fuel assembly (15 x 15 array). A comparison of critical dimensions is indicated in Table 4-1 of this report. The differences are essentially geometric and will result in a lower linear power density for the Mark C design than for the Mark B design. The lower power density produces lower average and maximum centerline fuel temperatures.

The evaluation of the Babcock & Wilcox Mark C fuel mechanical design has been based upon the assessment of mechanical tests, in-reactor operating experience with prototype assemblies, and engineering tests. Additionally, the in-reactor performance of the Mark C fuel design will be evaluated with surveillance programs carried out by Babcock & Wilcox and individual utilities as described in Section 4.2.4 of this report.

4.2.2. Thermal Performance

In our evaluation of the thermal performance of the reactor fuel, we assume that densification of the uranium dioxide fuel pellets may occur during irradiation in light water reactors. The initial density of the fuel pellets, and the size, shape and distribution of pores within the fuel pellets, influences the densification phenomenon. Briefly stated, in-reactor densification (shrinkage) of oxide fuel pellets (1) may reduce gap conductance, and hence increase fuel temperatures, because of a decrease in pellet diameter; (2) increases the linear heat generation rate because of the decrease in pellet length; and (3) may result in gaps in the fuel column as a result of pellet length decreases -- these gaps produce local power spikes and the potential for cladding creep collapse.

TABLE 4-1

DIMENSIONAL COMPARISON OF BABCOCK & WILCOX FUEL DESIGNS

	Mark B <u>(15x15)</u>	Mark C <u>(17x17)</u>
Number of fuel rods per fuel assembly	208	264
Number of guide tubes per assembly	16	24
Number of instrument tubes per assembly	1	1
Fuel rod outside diameter, inches	0.430	0.379
Cladding thickness, inches	0.0265	0.0235
Fuel rod pitch, inches	0.568	0.501
Fuel assembly pitch spacing, inches	8.587	8.587
Guide tube outside diameter, inches	0.530	0.465
Instrument tube spacer sleeve outside diameter, inches	0.554	0.480
Fuel pellet outside diameter, inches	0.370	0.324
Fuel pellet length, inches	0.70	0.375
Fuel stack length, inches	144	143

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The engineering methods to be used by Babcock & Wilcox, to analyze the densification effects on fuel thermal performance, have been submitted in BAW-10083, "Babcock & Wilcox Model for Predicting In-Reactor Densification," Revision 1, November 1976, and have been approved by the staff for use in licensing. The results of our review are given in two staff reports: (1) "Review of Revision 1 to BAW-10083P, December 1976; and (2) "Review of B&W Densification Model," September 1975. Additional information on densification methods can be found in NUREG-0085, "The Analysis of Fuel Densification," July 1976.

The approved Babcock & Wilcox model for fuel densification has been incorporated, along with companion models for fuel swelling, gas release, gap closure, and cladding creep, in a Babcock & Wilcox computer code, called TACO. This code was written to calculate gap conductance, fission gas pressure, and stored energy, over the lifetime of the fuel rod. A description of the TACO code had been submitted in BAW-10087, "TACO - Fuel Pin Performance Analysis," June 1976. Our review has prompted several modifications of this code, including a revision of the fission gas release model to account for enhanced release at high exposures. Thermal performance calculations will be performed by Babcock & Wilcox with the approved version of the TACO code.

Topical report BAW-10084, "Program to Determine In-Reactor Performance of B&W Fuels-Cladding Creep Collapse," May 1974, describes the analytical procedure and supporting data developed by Babcock & Wilcox to determine the minimum time for fuel cladding to collapse under operating conditions. This topical report was accepted for use in safety analyses related to licensing, subject to provisions specified in our evaluation report, "A Generic Review of the B&W Cladding Creep Collapse Analysis Topical Report BAW-10084," August 1974. The computer code used to perform these calculations is referred to as CROV. In our evaluation of the CROV method, we stated that it was acceptable for use in safety analyses for licensing provided that (1) the creep related material properties are similar to those characteristic of current Babcock & Wilcox cladding, (2) the initial ovality input to CROV both bounds the as-fabricated cladding and is not less than 0.0005 inch (ovality equals maximum outside diameter minus minimum outside diameter), and (3) the results of the long-term, in-reactor confirmatory tests continue to be favorable.

We recently reviewed BAW-10084, Revision 1, October 1976, of the CROV report. The revised report was unchanged from the original except for the creep correlation and its effects on predicted times to cladding collapse. In our evaluation report, "Program to Determine In-Reactor Performance of B&W Fuels-Cladding Creep Collapse," December 1976, we concluded that the revised cladding creep correlation provides a more accurate description of the creep phenomenon expression, reduces some of the former uncertainties, eliminates the need for some assumptions, and still conservatively represents appropriate creep data. The revised report has, therefore, been found acceptable for use in safety analyses related to licensing,

subject to the same conditions noted above for the original report. Following our review of even more recent information, which includes comparisons of CROV predictions for Mark C 17 x 17 fuel cladding creep collapse with actual creep data from Mark C cladding, we have concluded that CROV is also suitable for creep response calculations for Mark C fuel assemblies such as those to be used in Davis-Besse Units 2 and 3.

In the Preliminary Safety Analysis Report, the applicant has referenced a November 1974 issue of BAW-10084. We have notified the applicant that our acceptance of his proposed analytical procedures is contingent upon his use of the October 1976 Revision 1 of that report, and he has since committed to using Revision 1. Reference should be made in the Final Safety Analysis Report to the then most recent CROV report that includes the results of our CROV review to that time.

Based on the foregoing, we have concluded that the analytical methods to be employed by Babcock & Wilcox to consider thermal performance of the fuel are acceptable at the construction permit stage of review.

4.2.3 Mechanical Performance

Although most of our operating experience with Babcock & Wilcox fuel has been with the Mark B 15 x 15 fuel assemblies, substantially all of the in-reactor operating experience with Babcock & Wilcox fuel rods and assemblies is applicable to the Davis-Besse Mark C 17 x 17 fuel design, since the 17 x 17 fuel assembly is only a slight mechanical extrapolation from the 15 x 15 fuel assembly. The current use of similar fuel rods and assemblies has yielded operating experience that provides confidence in the acceptable performance of the Mark C fuel assembly design. In addition, Babcock & Wilcox is irradiating two Mark C demonstration assemblies in the Oconee 2 reactor. Nondestructive and post-irradiation examinations are being performed at each refueling outage. The plan for irradiation of these two 17 x 17 demonstration assemblies is described in BAW-1424, "Irradiation of Two 17 x 17 Demonstration Assemblies in Oconee 2, Cycle 2 - Reload Report." January 1976.

Many of the tests and analyses previously performed for the Mark B fuel assembly are applicable for the Mark C assembly. The development program for the Mark C design is drawing upon the experience gained in the Mark B development program. Babcock & Wilcox has stated that the Mark C program results will be used to demonstrate that Mark B tests and analyses are applicable for the safety analysis of the Mark C fuel. Where necessary, supplemental tests have been planned to demonstrate the performance of the Mark C design. In the event that any experimental results fall outside the design values used in the analysis of Mark C assembly performance, changes in the Mark C design may be required.

The Mark C assembly program objectives are to obtain data for analytical models, to confirm analytical predictions, and to verify the adequacy of the design.

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Included in this effort are mechanical and flow tests and critical heat flux and reflood heat transfer testing. These programs are described in Preliminary Safety Analysis Report Section 1.5 and in more detail in the topical report BAW-10097, "Mark C (17 x 17) Fuel Assembly - Research and Development - Revision 2," July 1975. We reviewed this topical report and found it acceptable for reference in licensing applications, provided that Babcock & Wilcox submits semiannual status reports (which began in January 1976) on the Mark C research and development programs. In our evaluation report on the Mark C research and development program, however, we also noted that a thorough safety review of the Mark C design, and issuance of an operating license for a Mark C plant, would require additional information in some areas that were not addressed in the research and development topical report. One such area is rod bowing.

Rod-to-rod gap spacing measurements have been taken through three cycles of operation on the lead Mark B plant (Oconee 1) on fuel assemblies that have experienced 26,000 megawatt days per tonne of uranium burnup. In addition, visual examinations of peripheral rods for bowing have been performed at five different plants, and bow profile measurements have been taken on rods on the lead Mark B plant. The schedule and scope of the 15 x 15 examination program includes three-cycle examination of 15 x 15 fuel assemblies in the lead Mark B plant. End-of-core-1 and end-of-core-2 examinations have been completed, and the data and evaluations have been reported in a letter, K. E. Suhrke to D. F. Ross, September 10, 1976.

Methods used by Babcock & Wilcox to analyze the effects of rod bowing in Mark B (15 x 15) fuel have been reviewed and discussed in staff reports, "Babcock & Wilcox Rod Bow Model," November 26, 1975; and "Babcock & Wilcox Rod Bowing," April 5, 1976. The models and procedures to be used in rod bowing analyses, for thermal and hydraulic considerations, are presented in a staff memorandum, "Revised Interim Safety Evaluation Report on the Effects of Fuel Rod Bowing on Thermal Margin Calculations for Light Water Reactors," February 16, 1977. Further revised methods for Mark C (17 x 17) fuel, which may be presented in a supplement to the Mark C topical report or as a separate report, will be reviewed at the operating license stage of review.

We conclude, on the basis of the above programs, that there is reasonable assurance that the effects of rod bowing in Mark C fuel assemblies can and will be properly accounted for prior to the issuance of an operating license for Davis-Besse Units 2 and 3, and we find the current information and commitments acceptable at the construction permit stage of review.

In addition to rod bowing, several other fuel performance issues are being studied as generic issues. These items are discussed in the following paragraphs of this section, and include the mechanical response to seismic and loss-of-coolant accident forces, the potential for water logging rupture, fretting and wear,

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fatigue, and pellet/cladding interaction. Although each of these issues has been identified as an area requiring further study, we conclude that current information available from Babcock & Wilcox is acceptable at the preliminary design stage of review. The basis for our conclusion as to each item is set forth below.

With respect to the seismic and loss-of-coolant analyses, we have requested Babcock & Wilcox to provide in-depth safety analysis of the seismic and postulated loss-of-coolant accident response of the Mark C (17 x 17) fuel assembly. We have received Topical Report BAW-10133, "Mark C LOCA-Seismic Analysis," and have found that there is not enough information in that report to permit us to perform a thorough review. We have, therefore, asked Babcock & Wilcox to provide substantially more technical information. We expect to complete our review prior to fabrication of the fuel assemblies for Davis-Besse Units 2 and 3, thus permitting inclusion of design changes that might be required as a result of our review.

The applicant has indicated that the effects of waterlogging rupture are not severe and should not result in failure propagation. We have reviewed the safety aspects of waterlogging failure, not only as a result of our Preliminary Safety Analysis Report review activity, but also as a consequence of our inquiry into the broader issue of pellet/cladding interaction as a potential failure mechanism. A survey of the available information, which includes (1) test results from SPERT and the Japanese test reactor NSRR, and (2) observations of waterlogging failures in commercial reactors, indicates that the assessment of the consequences of waterlogging failures made by Babcock & Wilcox is correct. Our review of the consequences of waterlogging failures has not resulted in the identification of a safety-related incident to date (see "Evaluation of the Behavior of Waterlogged Fuel Rod Failures in LWRs," NUREG-0303, March 1978). We conclude this matter is acceptably addressed for the construction permit stage of review.

Limitations on power rate changes could also affect pellet/cladding interaction, which is being reviewed as a generic item. The Babcock & Wilcox fuel rod design incorporates features directed at reducing cladding strain due to pellet/cladding interaction. These include pellet chamfering, rod prepressurization, plenum regions at both the top and bottom of the fuel rod, and greater cladding thickness-to-diameter ratio in Mark C fuel than in Mark B fuel. Based on the available experimental and commercial reactor data, the design features adopted by Babcock & Wilcox should result in a reduction in pellet/cladding interaction failures or a delay of such failures until late in the fuel design life. While the failure thresholds are probably lower at high burnup than at low burnup, the fuel operating parameters are also less severe. Our review of the consequences of pellet/cladding interaction failures has not resulted in the identification of a safety-related incident to date. We conclude that this matter is acceptably addressed for the construction permit stage of review, and we will continue our generic study of this matter.

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In the treatment of fuel assembly fretting and wear, Babcock & Wilcox asserts that, for Mark B fuel, the potential for fretting is low so long as a pressure contact between fuel rod and spacer grid is maintained throughout life. This condition is proposed as a design limit for Mark C, as well as for Mark B, assemblies because of their similarity in design. A test program, outlined in Preliminary Safety Analysis Report Section 1.5.3.1.2, is being performed to verify the adequacy of this limit for the Mark C fuel assembly. Fuel assembly resistance to fretting and wear is to be evaluated on the basis of test results from exposure at reactor operating conditions in the control rod drive line facility, which is described in the Topical Report BAW-10097, "Mark C (17 x 17) Fuel Assembly-Research and Development, Revision 2." In addition, absence of fretting and wear will be confirmed for a range of flow rates, temperatures and pressures in both the control rod drive line facility and the cold water facility. We have requested a topical report on the results and interpretation of test data. In the topical report, Babcock & Wilcox should also specify the criteria used in judging the applicability and adequacy of the tests. We conclude this matter is acceptably addressed for the construction permit stage of review.

In the BSAR-205 Preliminary Safety Analysis Report discussion of cycling and fatigue, Babcock & Wilcox states that "fatigue analyses, based on conservative assumptions, will be performed to show that design limits . . . are met." In view of this statement, we had, as a part of our BSAR-205 review, requested a topical report on the cycling and fatigue analyses. This topical report was to have been submitted for review at least one year prior to the date of submittal of a Final Safety Analysis Report on a plant using Mark C fuel. We have subsequently been notified, however, that this information is contained in the Final Safety Analysis Report for the Bellefonte Nuclear Plant (Docket 50-438/439). Therefore, we have withdrawn our request for the topical report, and will review the Bellefonte information on cycling and fatigue. Since this subject will not be resolved by a generic topical report, it should be fully addressed in the Davis-Besse Units 2 and 3 Final Safety Analysis Report. This is sufficient time, before fuel fabrication, to incorporate any design changes that may be needed as a result of our review.

4.2.4 Fuel Surveillance

Performance of the fuel during operation will be indirectly monitored, by measurement of the activities of both the primary and secondary coolants, for compliance with technical specification limits. For new fuel designs, for which there is no operating experience, we require that a supplemental fuel surveillance program be conducted. The supplemental fuel surveillance program is directed at monitoring the behavior of the actual fuel systems as they perform in-reactor, thus demonstrating the adequacy of the conclusions reached in the design evaluation.

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We are, therefore, requiring a supplemental fuel surveillance program for the first two plants with Mark C fuel. Based on existing construction schedules, we expect the first two plants to be Bellefonte Unit 1 and Washington Nuclear Project Unit 1. The details of the surveillance program requirements are provided in our letter, D. F. Ross (NRC) to K. E. Suhrke of Babcock & Wilcox, September 20, 1976. The program will consist of a visual inspection of all the peripheral rods in the initial-core fuel assemblies as they are discharged into the spent fuel pool. The visual inspection will include observations for cladding defects, fretting, rod bowing, corrosion and deposition, and geometric distortion. If any anomalies are detected during the visual examination, further investigation will be performed, including destructive examination of a fuel assembly or individual fuel rods as required.

If the fuel surveillance programs are not in progress at the operating license stage of review for Davis-Besse Units 2 and 3, we will consider imposing the program on Davis-Besse Units 2 and 3.

4.2.5 Fuel Mechanical Design Evaluation Conclusion

Babcock & Wilcox is conducting a development program in order to verify the design analyses and to demonstrate that the Mark C (17 x 17) fuel will perform successfully. We have reviewed the design and are continuing to monitor and evaluate the results of the ongoing development program.

We conclude that there is reasonable assurance that the cladding integrity of the Mark C (17 x 17) fuel will be maintained, that significant amounts of radioactivity will not be released, and that neither accidents nor earthquake-induced loads will result in either an inability to cool the fuel or interference with control rod insertion. Our conclusion is based on (1) analytical results, (2) operating experience with similar Mark B (15 x 15) fuel, (3) increased thermal margins of 17 x 17 fuels, (4) technical specifications that will be in effect to limit offgas and effluent activity, and (5) the on-going development and demonstration test program.

4.3 Nuclear Design

4.3.1 Design Bases

Our review of the nuclear design of Units 2 and 3 was based on information supplied by the applicant in the Preliminary Safety Analysis Report, as amended, and discussions with the reactor supplier, Babcock and Wilcox. The nuclear design features of this reactor are similar to those of Davis-Besse Unit 1 and Rancho Seco Unit 1. The major difference between these plants and Davis-Besse Units 2 and 3 is the use of the Mark C (17x17) fuel assembly in Units 2 and 3 instead of the Mark B (15x15) fuel assembly that is used in the other two plants.

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The proposed design bases comply with Criteria 10 through 13, 20, and 25 through 28 of the General Design Criteria. Acceptable fuel design limits are specified, a negative prompt feedback coefficient is specified, and power oscillations will be precluded by design or will be detected and suppressed by the control system. Design bases require a control and monitoring system that will automatically initiate a rapid reactivity insertion to prevent exceeding fuel design limits in normal operation, anticipated transients, or accident conditions. The control system will be designed so that a single malfunction or single operator error will cause no violation of fuel design limits. A chemical boron shim system will be provided that is capable of bringing the reactor to cold shutdown conditions, and the control system, when combined with emergency core cooling system operation, will control reactivity changes during accident conditions. Reactivity accident conditions will be limited so that no damage to the reactor coolant system occurs.

We find the design bases presented in the Preliminary Safety Analysis Report to be acceptable.

Descriptions of the first fuel cycle enrichment and burnable poison distributions, physics of the fuel burnout process, soluble boron concentrations, delayed neutron fraction, and neutron lifetimes have been provided. The values for these parameters are consistent with the design bases and are acceptable.

4.3.2 Power Distribution

We have reviewed the methods used by Babcock & Wilcox to calculate power distributions for both steady state and transient conditions. See Section 4.3.7 of this report. These methods have been compared with experimental results to determine the uncertainty in the core peaking factor for plants with 177 fuel assemblies. In addition, the power distributions at various core powers have been compared to calculated values during startup testing of several Babcock & Wilcox plants with 177 fuel assemblies. These comparisons have shown that the proposed use of a 7.5 percent uncertainty on the calculated core peaking factor is conservative. For one plant (Rancho Seco Unit 1) our consultant, Brookhaven National Laboratory, has performed an independent audit of heat generation rates at beginning of life. The audit calculation agreed, to within approximately 3.5 percent, with those calculated by Babcock & Wilcox. In addition, the peaking factors will be measured at several power levels, in the startup tests for this facility, to provide further confirmation of the calculated peaking factors.

Monitoring of power distributions for this facility will be performed by neutron detectors outside the reactor vessel (excore detectors) in the form of axially split ionization chambers, or by self-powered neutron detectors inside the reactor core (incore detectors). The former will be used for the reactor protection system and may be used for monitoring operating conditions. The incore detectors will be used to determine the axial imbalance and quadrant tilt of the core and

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for calibrating certain functions of the excore system, and may be used for monitoring operating conditions. Both sets of instrumentation will be normalized to the calorimetric value of total reactor power. Functionally identical systems have been successfully employed on other Babcock & Wilcox reactors, and we find their use acceptable.

Peaking factor limits are determined by the requirement that fuel design limits not be exceeded during normal operation or anticipated transients, and that a fuel clad temperature of 2200 degrees Fahrenheit not be exceeded during a loss-of-coolant accident. During operation, peaking factors will be controlled by the application of limits on control rod insertion and on axial imbalance. Operating experience with Babcock & Wilcox reactors has shown that operation within these limits (which are included in the plant technical specifications) is sufficient to assure that peaking factor limits are not violated.

On the basis that calculational techniques used have been shown to yield good results in the past for 15 x 15/177 fuel assembly designs, and our review indicates that these calculational techniques are just as appropriate, with necessary adjustments, for the 17 x 17/177 fuel assembly designs, we conclude that acceptable predictions of core power distributions have been made for Davis-Besse Units 2 and 3. On the basis of satisfactory operating experience in Babcock & Wilcox reactors and our review of the instrumentation proposed for Davis-Besse Units 2 and 3, we conclude that adequate instrumentation will exist to monitor power distributions during plant operation.

4.3.3 Reactivity Coefficients

Babcock & Wilcox has presented, in the Preliminary Safety Analysis Report, values of various reactivity coefficients that are used in the analysis of normal, transient, and accident conditions. Included are values for moderator temperature, Doppler, and power coefficients of reactivity as well as soluble poison worths. The calculation methods used to obtain these quantities have been reviewed in Section 4.3.7 of this report. These quantities are routinely measured at various reactor power levels as part of the startup program. Past experience has shown that measured power coefficient and moderator temperature coefficients are usually within about 10 percent of calculated values. Experience has also shown that values that are used in safety analyses are always conservative relative to the measured values.

With respect to operation with a positive moderator temperature coefficient, a technical specification requirement that this coefficient be negative above 95 percent of rated power is typically imposed and will be included in the technical specification for this facility. Analyses show that transients or accidents initiated below 95 percent power do not exceed safety limits even for positive temperature coefficients within technical specification limits.

Based on our review and approval of the proposed calculational methods, and on the fact that there has been good agreement between calculation and measurement in operating Babcock & Wilcox reactors, we conclude that suitably conservative values of reactivity coefficients have been provided. Startup tests for Davis-Besse Units 2 and 3 will be performed to confirm that conservative reactivity coefficients were used in the safety analysis.

4.3.4 Control Requirements

To allow for changes in reactivity due to reactor heatup, load following, and fuel burnup with consequent fission product buildup, a significant amount of excess reactivity will be designed into the core. This excess reactivity will be controlled by a combination of soluble boron and control rods.

Soluble boron will be used to control reactivity changes due to:

- (1) Moderator deficit from ambient to operating temperatures
- (2) Equilibrium xenon and samarium buildup
- (3) Fuel depletion and fission product buildup throughout cycle life -- that part not controlled by burnable poison
- (4) Transient xenon resulting from load following.

Full-length regulating rods will be used to control reactivity changes due to:

- (1) Moderator deficit from hot zero power to full power
- (2) Power level changes (Doppler).

Lumped burnable poison rods will be used for radial flux shaping and to control part of the reactivity change due to fuel burnup. Part-length control rods will be used to maintain an axially balanced power distribution.

Babcock & Wilcox has provided data to show that adequate control will exist to satisfy the above requirements, with enough additional control to provide a hot shutdown effective multiplication factor less than or equal to 0.99 during the initial and equilibrium cycles with the most reactive control rod stuck out of the core. Comparisons between calculated and measured control rod worths have been made for silver-indium-cadmium control rods in 177 (15 x 15) fuel assembly plants. The agreement obtained was well within the 10 percent error assigned to total rod worth in the analysis. In addition, control rod worth measurements, with particular emphasis on shutdown margins, will be a part of the startup program for Davis-Besse Units 2 and 3 and will serve to confirm calculated rod worths and shutdown margins.

The soluble boron, or chemical shim, system will have sufficient capability to shut down the reactor and to maintain it in the cold shutdown state at any point in core life, in conformance with the requirement of Criterion 26 of the General Design Criteria.

On the basis of our review, which has included the considerations described above, we conclude that the applicant has made suitably conservative assessments of reactivity control requirements for the reactors and that adequate reactivity control systems will be provided to assure the shutdown capability required by Criteria 26 and 27 of the General Design Criteria.

4.3.5 Control Rod Patterns and Reactivity Worth

The full-length control rods will be divided into two classes - shutdown (or safety) rods and regulating rods. The safety rods will always be completely out of the core when the reactor is at operating condition. Load (core power) changes will be made with regulating rods or soluble boron or both. Regulating rod insertion will be controlled by power-dependent insertion limits given in the technical specifications. These limits will assure that:

- (1) There is sufficient negative reactivity available to permit the rapid shutdown of the reactor with adequate margin.
- (2) The worth of any control rod that might be ejected, in the unlikely event of failure of a pressure barrier in a control rod drive mechanism, is not greater than that which has been shown to have acceptable consequences in the safety analysis.
- (3) The overall peaking factor does not exceed that used in the safety analysis as the initiating value for transients or accidents.

We have reviewed the calculated rod worths and the methods used by Babcock & Wilcox to obtain the worths. Our consultant, Brookhaven National Laboratory, has performed independent calculations of regulating bank worths, and their calculations agree, to within about two percent, with those calculated by Babcock & Wilcox.

The effects of fuel densification on peaking factors will be reflected in the technical specifications on rod insertion limits. These effects will be considered in the operating license review, since "as-built" fuel characteristics must be used. Babcock & Wilcox has an acceptable model for fuel densification that will be applied. It is expected that modifications may be made in this model as more data are obtained. These modifications will be reviewed generically, and we will use the latest acceptable model to account for densification effects at the operating license stage of review.

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On the basis of our review, we have concluded that the rod groupings proposed for the facility reactors satisfy the requirements for safe shutdown, ejected rod worth, and power distribution control, and are acceptable.

4.3.6 Stability

The stability of the core with respect to xenon oscillations has been analyzed. Azimuthal xenon oscillations are predicted to be damped, but sustained axial xenon oscillations may occur under certain conditions, if no remedial action is taken. The stability of 177 fuel assembly reactors was investigated during the startup tests for the Oconee Unit 1 reactor. A diagonal (combination of azimuthal and axial) oscillation was induced at 75 percent power, and the reactor response was monitored for 72 hours. The azimuthal component was strongly damped, but the axial component was divergent. At 70 hours into the transient, the part-length rods were used to return the axial imbalance to near zero, where it was successfully held.

While the Oconee reactor is rodded and the Davis-Besse reactors will be unrodded, the behavior of the reactors is expected to be similar. Startup tests at Rancho Seco, which operates unrodded, showed that its transient behavior was similar to that of Oconee 1.

On the basis of this demonstration of the azimuthal stability of a similar reactor and the ability of the control system to suppress axial oscillations, we conclude that the reactors will not experience uncontrolled oscillation.

4.3.7 Analytical Methods

We have reviewed the analytical methods used by Babcock & Wilcox to perform core design. The major design tool is PDQ-7, a diffusion theory code with industry-wide usage. Cross sections for use with this code are prepared in a manner similar to that used by others in the industry. Comparisons between calculated and measured design parameters have been made during startup tests on six reactors designed by Babcock & Wilcox. In all cases, the comparisons have been satisfactory. On the basis of our review, we conclude that the analytical methods used for the design of Davis-Besse Units 2 and 3 are acceptable.

4.4 Thermal and Hydraulic Design

4.4.1 Design Criteria and Parameters

The principal criterion for the thermal-hydraulic design of a reactor is avoidance of thermally-induced fuel damage during normal steady-state operation and during anticipated operational occurrences. Babcock & Wilcox uses the following design limits to satisfy this criterion:

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- (1) The fuel rod cladding, fuel pellets, and fuel rod internals must be designed so that the fuel-to-clad characteristics assure that the maximum fuel temperature does not exceed the fuel melting limit at 112 percent design overpower at any time during core life. The fuel melting temperature is 5080 degrees Fahrenheit at beginning-of-life, and decreases linearly to 4800 degrees Fahrenheit at end-of-life (43,000 megawatt days per tonne of uranium).
- (2) The minimum allowable departure from nucleate boiling ratio during steady-state operation and anticipated transients is 1.32, based on the BAW-2 correlation.
- (3) Hydraulic stability is required during all steady-state and operational transient conditions.

We find these criteria for the avoidance of fuel damage to be acceptable.

We have reviewed the thermal-hydraulic design parameters for operation at 2772 megawatts-thermal. A comparison of the thermal-hydraulic performance for Davis-Besse Units 2 and 3 and for Davis-Besse Unit 1 is presented in Table 4.2.

The hydraulic analysis has been based on vessel model flow tests, as described in BAW-10037, "Reactor Vessel Model Flow Tests." BAW-10037 has been accepted for reference in licensing submittals.

As with Davis-Besse Unit 1, the reactor internals for Units 2 and 3 will include four core support internal vent valves located on a common plane in the upper core support weldment above the outlet nozzles. These valves will provide a direct flow path between the upper core region and the inlet annulus in the event of a loss-of-coolant accident from an inlet (cold leg) line break. This flow path will provide for pressure equalization by the venting of steam to the break and will permit the emergency core cooling water to reflood at a higher rate. For Unit 1, we had required further consideration of the effect of stuck-open internal vent valves on the thermal hydraulic design and core cooling characteristics. Babcock & Wilcox responded on a generic basis by submitting a report, "B&W Operating Experience of Reactor Internal Vent Valves." We have evaluated this report and concluded that a flow penalty due to internal vent valve leakage need not be applied. The applicant must, however, implement a program of inspection and test of the valves at each refueling. The technical specifications will include this requirement.

The margin to departure from nucleate boiling at any point in the core is expressed in terms of the departure from nucleate boiling ratio. This ratio is defined as the heat flux required to produce departure from nucleate boiling at the calculated local coolant conditions divided by the actual local heat flux.

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TABLE 4.2

THERMAL AND HYDRAULIC PARAMETERS

	<u>Davis-Besse Units 2 and 3</u>	<u>Davis-Besse Unit 1</u>
Reactor core heat output, thermal megawatts	2772	2772
System pressure, nominal, pounds per square inch	2200	2200
Minimum departure from nucleate boiling ratio at design power	3.05 (BAW-2)	1.79 (W-3)
Minimum departure from nucleate boiling ratio at design overpower (112 percent)	2.09 (BAW-2)	1.41 (W-3)
Total reactor coolant flow, millions of pounds per hour	131.32	131.32
Core coolant average velocity, feet per second	15.9	15.74
Coolant temperature, degrees Fahrenheit		
vessel inlet	555.4	5. 4
core outlet	611.7	611.7
Total heat transfer surface in core, square feet	55,251	49,734
Average heat flux, British thermal units per hour per square foot	166,613	185,090
Maximum heat flux, British thermal units per hour per square foot	442,060	544,200
Maximum thermal output at design power, kilowatts per foot	12.85	18.28
Maximum fuel centerline temperature at design power, degrees Fahrenheit	3532	4060

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The margin to departure from nucleate boiling will be chosen to provide a 95 percent probability, with 95 percent confidence, that departure from nucleate boiling will not occur on fuel rods having the (calculated) minimum departure from nucleate boiling ratio during normal operation and anticipated operational occurrences.

The departure from nucleate boiling correlation used for the design of this core is the BAW-2 correlation. The BAW-2 correlation was derived from data on six-foot long heated rods that simulated the rod diameter and spacing of 15 x 15 fuel assemblies. Babcock & Wilcox has stated that they will perform tests on full-length heated rods of the 17 x 17 geometry with uniform and nonuniform axial heat flux. Results from these tests, and the statistical analysis of the results, must confirm the thermal-hydraulic design prior to issuance of an operating license for Davis-Besse Unit 2 or 3.

Prevention of departure from nucleate boiling, for steady-state operation and anticipated transients, will assure that the hot spot of the fuel cladding is at a temperature only slightly greater than that of the coolant and that the fuel cladding will maintain its integrity.

Another parameter that influences the thermal-hydraulic design of the core is the rod-to-rod bowing within fuel assemblies. During the Oconee 1 (Docket number 50-269) refueling, six 15 x 15 assemblies were examined visually and dimensionally. The water channel and line scan measurements indicated a maximum rod bow of approximately 30 thousandths of an inch. Babcock & Wilcox states that the observed rod bow is accommodated within the current 17 x 17 design, and that they are pursuing a program to demonstrate this. Babcock & Wilcox plans to develop, in the near future, bow correlations and predictive techniques to analyze the data and the predicted bow from a thermal-hydraulic standpoint, and has committed to provide results of the localized effect of rod bow on the departure from nucleate boiling for the 17 x 17 fuel assembly design. As discussed in Section 4.2.3 of this report, we conclude that the information regarding rod bowing now available from Babcock & Wilcox is acceptable at the preliminary design stage of review.

Protective action to prevent departure from nucleate boiling in the core will be provided in part by the reactor protection system's calculating module. We will review the design and implementation of the protective software used in the calculating module prior to issuance of an operating license. Our review of the reactor protection system (RPS-II) is discussed in Section 7.2 of this report.

4.4.2 Conclusions

On the basis of our review of the thermal-hydraulic analytical techniques and available supporting experimental data, we conclude that there is reasonable

assurance that (1) the proposed thermal-hydraulic design will account for departure from nucleate boiling and fuel centerline temperature limitations in a satisfactory manner, and (2) the conservatism in the thermal-hydraulic design procedures can be verified. Therefore, we conclude that the presently available information on the preliminary thermal-hydraulic design of the reactor is acceptable for the construction permit stage of review.

However, we will require that several items be resolved prior to the issuance of an operating license. These are:

- (1) Development of critical heat flux data with full-length heaters for the BAW-2 correlation. This data base should include both uniform and nonuniform axial heat flux tests on full-length heaters with the 17 x 17 rod diameter and spacing. If necessary, the BAW-2 correlation should be modified to agree with the data.
- (2) Statistical analysis of the critical heat flux data to verify that the minimum departure from nucleate boiling ratio complies with the 95/95 design criterion.
- (3) Review and approval of the vessel model flow test topical report for the 205 fuel assembly configuration.
- (4) Review and approval of the HYTRAC and CHATA codes.
- (5) Review and approval of the as-built dimensions of the core flow distribution plate.
- (6) Review and approval of the reactor protection system protective software.

In the event that the analytical methods are determined not to be conservative during the final design review, appropriate restrictions on operation can be established at the operating license stage.

4.5 Reactivity Control System

4.5.1 Functional Design

Reactivity control will be provided by control rod assemblies, axial power shaping rod assemblies, and burnable poison rod assemblies. Additional control will be provided by the addition of soluble boron to the reactor coolant by the makeup and purification system.

The control rod system will consist of 64 full-length control rod assemblies, eight axial power shaping rod assemblies with absorber material in the lower one-quarter of the rod only, and 116 full-length burnable poison rod assemblies, each containing 24 absorber rods.

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The lower quarter of each axial power shaping rod will contain silver-indium-cadmium alloy positioned within a cold worked austenitic stainless steel tube that comprises the basic rod structure. Each burnable poison rod assembly will have a section of sintered aluminum oxide/boron carbide mixture in pellet form. The burnable poison rod will be clad with coldworked Zircaloy-4 tubing, and will have Zircaloy-4 upper and lower end pieces. All control assemblies will have 24 control elements that fit into the fuel assembly guide tubes.

The control rods will be used to compensate for reactivity changes due to variations in operating conditions of the reactor, such as power and temperature changes. The part-length axial power shaping rod assemblies will be used to maintain an axially balanced power distribution. The burnable poison assemblies are designed to control the reactivity change due to fuel burnup and fission product buildup and also to reduce the amount of soluble boron required in the reactor coolant.

The control rod worth will be sufficient to provide the required one percent shutdown margin for a hot shutdown, while the soluble boron, provided from the makeup and purification system, will be used to provide the required margin for a cold shutdown.

For the accident analysis (see Preliminary Safety Analysis Report Table 15.1-2), a rod drop time of 1.7 seconds, for two-thirds insertion, was used. This is consistent with rod drop times previously reviewed and approved for the Babcock & Wilcox silver-indium-cadmium control rod.

The functional design of the reactivity control systems meets the applicable requirements of General Design Criteria 20, 23, 25, 26, 27, 28, and 29 and is, therefore, acceptable.

Additional objectives of our review were to determine that the design, fabrication, and construction of the control rod drive mechanisms will provide structural adequacy and that suitable life cycle testing programs have been utilized to prove operability under service conditions.

The applicant has provided a sufficiently detailed description of the facility control rod drive mechanism, including the functional aspects and principal components. In addition, the applicant has specified design load combinations and corresponding stress limits for normal, upset, emergency, and faulted load conditions. The pressure boundary parts will be designed and fabricated according to ASME Boiler and Pressure Vessel Code Section III rules, and applicable code cases, for class 1 nuclear components. The applicant had referenced topical report BAW-10029, "Control Drive Mechanism Test Program," Revision 1, to verify operability and life tests for the control drive mechanism. Revision 3 of the

referenced topical report has been reviewed by the staff and found to be an acceptable reference. We required the applicant to update the referenced revision to Revision 3, which the applicant did by letter dated June 26, 1978. During the operating license stage of review, we will require that the applicant:

- (1) Demonstrate that the postulated seismic or pipe break events, for the Davis-Besse Units 2 and 3 application, envelope the loads used in the topical report.
- (2) Address the difference between the expected (usually 40-year) plant life and the life testing done by Babcock & Wilcox to demonstrate at least 20 years of expected life for the control rod drive mechanisms.

The design criteria and the testing program conducted in verification of the mechanical operability and life cycle capabilities of the reactivity control system are in conformance with established criteria, codes, standards and specifications acceptable to us. The use of these criteria provides reasonable assurance that the system will function reliably when required, and forms an acceptable basis for satisfying the mechanical reliability requirements of Criterion 27 of the General Design Criteria.

4.5.2 Structural Materials

The mechanical properties of structural materials selected for the control rod system components, that will be exposed to the reactor coolant, satisfy Appendix I of Section III of the ASME Boiler and Pressure Vessel Code, or Part A of Section II of the Code. They also satisfy our position, stated in NUREG-75/087, "Standard Review Plan," Section 4.5.1, that the yield strength of cold worked austenitic stainless steel should not exceed 90,000 pounds per square inch.

The controls imposed upon the austenitic stainless steel of the system satisfy the recommendations of NRC Interim Position on Regulatory Guide 1.31, "Control of Stainless Steel Welding," and of Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel." The compatibility of all control rod system materials with the reactor coolant satisfies the criteria for Articles NB-2160 and NB-3120 of Section III of the ASME Boiler and Pressure Vessel Code. Both martensitic and precipitation-hardening stainless steels will be given tempering or aging treatments in accordance with the NUREG-75/087, "Standard Review Plan." Cleaning and cleanliness controls will be in accordance with ANSI Standard N.45.2.1-1973, "Cleaning of Fluid Systems and Associated Components for Nuclear Power Plants," and Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants."

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Conformance with the codes, standards, and regulatory guides indicated above, and with the Standard Review Plan positions on the allowable maximum yield strength of cold worked austenitic stainless steel and on minimum tempering or aging temperatures of martensitic and precipitation-hardened stainless steels, constitutes an acceptable basis for meeting the requirements of Criterion 26 of the General Design Criteria.

4.6 Reactor Pressure Vessel Internals

We have reviewed the information presented in the Preliminary Safety Analysis Report on:

- (1) The physical and design arrangements of all reactor internals structures, components, assemblies, and systems, including the manner of positioning and securing such items with the reactor pressure vessel, the manner of providing for axial and lateral retention and support of the internals assemblies and components, and the manner of accommodating dimensional changes due to thermal and other effects.
- (2) The design loading conditions that will provide the basis for the design of the reactor internals to sustain normal operation, anticipated operational occurrences, postulated accidents, and seismic events, including all combinations of design loadings that will be accounted for in the design of the core support structure, such as operating pressure differences and thermal effects, seismic loads, and transient pressure loads associated with postulated loss-of-coolant accidents.
- (3) Each combination of design loadings categorized with respect to the "normal," "upset," "emergency," or "faulted" conditions as defined in Section III of the ASME Boiler and Pressure Vessel Code and the associated design stress intensity or deformation limits. The design loadings include the safe shutdown earthquake and operating basis earthquake loads.
- (4) The design bases for the mechanical design of the reactor vessel internals, including limits such as maximum allowable stresses, deflection, cycling, and fatigue limits, and core mechanical and thermal restraints for positioning and holddown purposes.

Additional discussion of the analytical evaluations and verification testing to be completed for the reactor pressure vessel internals is included in Section 3.9.1 of this report.

The applicant has committed to perform a dynamic system analysis of the reactor internals and of the connected piping system. This analysis will be provided with the Final Safety Analysis Report. The dynamic system analysis will be performed

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to confirm the structural design adequacy of the reactor internals and the unbroken piping loops to withstand the combined dynamic effects of the postulated occurrence of a loss-of-coolant accident and a safe shutdown earthquake.

We have reviewed descriptions of the analytical methods presented in the Preliminary Safety Analysis Report and find that they provide reasonable assurance that the combined stresses and strains in the components of the reactor coolant system and reactor internals will not exceed the allowable design limits for the materials of construction as specified in Appendix F to Section III of the ASME Boiler and Pressure Vessel Code. We conclude that there is reasonable assurance that the resulting deflections or displacements of any structural elements of the reactor internals will not significantly impair core cooling.

The assurance of structural integrity of the reactor internals under the postulated safe shutdown earthquake and the most severe loss-of-coolant accident conditions provides added confidence that the design can be expected to withstand a spectrum of lesser pipe breaks and seismic loading combinations. Limiting the stresses and deformations under such loading combinations provides an acceptable basis for the design of these structures and components to withstand the most adverse loading events which have been postulated to occur during the service lifetime without loss of structural integrity or impairment of function.

We conclude that the design procedures and criteria to be used by Babcock & Wilcox in the design of the reactor internals constitutes an acceptable basis for satisfying the applicable requirements of Criteria 1, 2, 4 and 10 of the General Design Criteria.

The facility design includes a loose parts monitoring system to detect the presence of loose parts in the reactor core and the primary coolant system. Included in the detection system will be piezoelectric accelerometers located in the upper and lower portions of the reactor vessel, on the reactor coolant pumps, and on the top and bottom of each steam generator. Other detectors will be used to monitor components outside the reactor coolant system. Appropriate readout equipment and tape recorders will be provided for analysis of the detector signals. We conclude that the system, as described in the Preliminary Safety Analysis Report, is acceptable.

The materials for construction of components of the reactor internals have been identified by specification, and we have found them to be in conformance with the requirements of Section III of the ASME Code.

The materials for reactor internals exposed to the reactor coolant have been identified and all of the materials are compatible with the expected environment, as proven by extensive testing and satisfactory performance. General corrosion on all materials is expected to be negligible.

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The controls that will be imposed on reactor coolant chemistry provide reasonable assurance that the reactor internals will be adequately protected during operation from conditions that could lead to stress corrosion of the materials and loss of component structural integrity.

The controls imposed upon components constructed of austenitic stainless steel, as used in the reactor internals, satisfy the recommendations of Nuclear Regulatory Commission Interim Position on Regulatory Guide 1.31, "Control of Stainless Steel Welding," Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," and Regulatory Guide 1.66, "Nondestructive Examination of Tubular Products." Material selection, fabrication practices, examination procedures, and protection procedures, performed in accordance with these recommendations, provide reasonable assurance that the austenitic stainless steel used for reactor internals will be in a metallurgical condition that precludes susceptibility to stress corrosion cracking during service. The use of materials proven to be satisfactory by actual service experience, and conformance with the recommendations of these regulatory guides, constitute an acceptable basis for meeting the applicable requirements of General Design Criteria 1 and 14.

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5.0 REACTOR COOLANT SYSTEM

5.1 Summary

Section 50.2(v) of 10 CFR Part 50 defines the reactor coolant pressure boundary as all those pressure-containing components of pressurized water-cooled nuclear power reactors, such as pressure vessels, piping, pumps and valves, that are:

- (1) Part of the reactor coolant system, or
- (2) Connected to the reactor coolant system, up to and including:
 - (a) The outermost containment isolation valve in system piping that penetrates primary reactor containment,
 - (b) The second of two valves normally closed during normal reactor operation in system piping that does not penetrate the primary reactor containment, and
 - (c) The reactor coolant system safety and relief valves.

The reactor coolant system contains the reactor vessel, including the control rod drive mechanism housings, the reactor coolant side of the two steam generators, the four reactor coolant pumps, a pressurizer, and the interconnecting piping and valves associated with these components. A description of the reactor coolant system and its normal operating modes is contained in the Preliminary Safety Analysis Report, Section 5.1.

The proposed Babcock & Wilcox nuclear steam supply system design for Davis-Besse Units 2 and 3 will incorporate a pressurized water reactor in a closed-cycle reactor coolant system. The reactor coolant system, located entirely within the containment, will circulate water in a split two-loop configuration, removing heat from the reactor core and transferring it to two vertical once-through steam generators. Each coolant loop will consist of a 36-inch inside diameter hot leg pipe between the reactor vessel outlet and the steam generator inlet, two 28-inch inside diameter cold leg pipes between the steam generator outlet and the inlets of the two reactor coolant pumps, and two 28-inch inside diameter cold leg pipes between the outlets of the two reactor coolant pumps and the two inlets to the reactor vessel. The reactor coolant system design does not include loop stop valves. The pressurizer will be connected to one of the hot legs by a 10-inch schedule 140 surge line, and to one of the cold leg pump discharge lines by a 2-1/2-inch schedule 160 spray line.

During operation, the reactor coolant system will transfer the heat, generated in the core, to the steam generators where steam will be produced to drive the turbine-generator. Borated demineralized water will be circulated in the system at a flow rate, pressure, and temperature consistent with achieving the design reactor core thermal-hydraulic performance. The water will also act as a radiation shield, and as a neutron moderator and reflector. The reactor coolant system design is essentially the same as that on other nuclear steam supply systems designed by Babcock & Wilcox.

5.2 Integrity of the Reactor Coolant Pressure Boundary
5.2.1 Design Criteria

Criterion 4 of the General Design Criteria requires that structures, systems and components important to safety be designed to accommodate the effects of normal operation, maintenance, testing, and postulated accidents. We reviewed the design of the reactor coolant pressure boundary components to determine whether component quality will be commensurate with the importance of the safety function of the reactor coolant pressure boundary.

We determined that the design loading combinations, specified under Section III of the ASME Code for Class 1 components, have been appropriately categorized with respect to the plant conditions identified as "normal," "upset," "emergency," or "faulted." The design limits proposed by the applicant for these plant conditions are consistent with the criteria recommended in Regulatory Guide 1.48, "Design Limits and Loading Combinations for Seismic Category I Fluid System Components." Use of these criteria for the design of reactor coolant pressure boundary components will provide reasonable assurance that, in the event of an earthquake at the site or of other system upset, emergency or faulted condition, the resulting combined stresses imposed upon the system components will not exceed the allowable design stresses and strain limits for the materials of construction.

Limiting the stresses and strains under such loading combinations provides a basis for the design of the system components for the most adverse loadings postulated to occur during the service lifetime without loss of the system's structural integrity. The design load combinations and associated stress and deformation limits specified for ASME Code Class 1 components and supports constitute an acceptable basis for design in satisfying the related requirements of Criteria 1, 2 and 4 of the General Design Criteria.

The applicant has identified the active components within the reactor coolant pressure boundary for which operation is required to safely shut down the plant and maintain it in a safe condition in the event of a safe shutdown earthquake or design basis accident. The applicant has agreed to utilize an operability assurance program, in addition to stress and deformation limits, to qualify active valves. Such a program will include valve testing, or a combination of tests and

predictive analysis, supplemented by seismic qualification testing of valve operator systems, to provide assurance that active components (1) will withstand the imposed loads associated with normal, upset, emergency and faulted plant conditions without loss of structural integrity, and (2) will perform the "active" function under conditions comparable to those expected when safe plant operation or shutdown is to be effected, or when the consequences of a seismic transient or of an accident are to be mitigated.

The applicant's commitment to develop and utilize a component operability assurance program found acceptable to us constitutes an acceptable basis for implementing the requirements of Criterion 1 of the General Design Criteria as related to the operability of ASME Code Class 1 active valves.

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

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

5.2.2 Codes and Standards

Components of the reactor coolant pressure boundary, as defined by the rules of 10 CFR Part 50, Section 50.55a, have been properly identified and classified in the Preliminary Safety Analysis Report as ASME Section III, Class 1, components. These components within the reactor coolant pressure boundary will be constructed in accordance with the requirements of the applicable codes and addenda as specified by 10 CFR Part 50, Section 50.55a, Codes and Standards.

The applicant has identified the ASME Code Cases whose requirements will be applied in the construction of pressure-retaining ASME Section III, Code Class 1, components within the reactor coolant pressure boundary (Quality Group Classification A). These code cases, listed in Table 5.2-2 of the Preliminary Safety Analysis Report, are in accordance with those code cases in Regulatory Guide 1.84, "Code Case Acceptability ASME Section III Design and Fabrication," and Regulatory Guide 1.85, "Code Case Acceptability ASME Section III Materials," and are acceptable.

We conclude that construction of the components of the reactor coolant pressure boundary, in conformance with the ASME Code and the Commission's regulations, provides adequate assurance that component quality will be commensurate with the importance of the safety function of the reactor coolant pressure boundary and is acceptable.

5.2.3 Overpressure Protection

The pressure relief system will prevent overpressurization of the reactor coolant pressure boundary under the most severe transients, and will limit the reactor pressure during normal operational transients.

Overpressure protection will be provided by two safety valves and one electrically-actuated relief valve. These valves will discharge to the pressurizer quench tank through a common header from the pressurizer. The reactor coolant system safety valves, in conjunction with the steam generator safety valves and the reactor protection system, will protect the reactor coolant system against overpressure in the event of a complete loss of heat sink. The design specifications and qualification for these valves are identical to those previously approved for Davis-Besse Unit 1. The three pressure-relieving valves will be designed in accordance with the ASME Code, Section III, Class 1.

The relief valve will have a capacity of 100,000 pounds per hour at 2,250 pounds per square inch gauge. It will be pressure-loaded and pilot-operated, and is designed to accommodate reactor power level changes of 10 percent. Unnecessary safety valve action will be prevented by operation of the relief valve.

Each of the two safety valves will have a capacity of 336,000 pounds per hour. The combined capacity of the two valves is twice that required by the most severe pressure transient, which is a control rod withdrawal at low power. This transient was analyzed assuming no direct reactor trip, no operator action, and no credit for actuation of the relief valve or the turbine bypass system. The peak pressure of 2,670 pounds per square inch absolute is below the ASME Code limit of 110 percent of design pressure.

We conclude that the overpressure protection design will limit the peak reactor coolant system pressure, following the worst transient from normal conditions, to the ASME Code allowable of 110 percent of design pressure, in conformance with General Design Criterion 15 and is, therefore, acceptable.

A number of transients have occurred in operating pressurized water reactors in which the limits of 10 CFR Part 50, Appendix G, have been exceeded during startup and shutdown operations. The Babcock & Wilcox design provides a unique feature to minimize the occurrence and limit the consequences of such events. During reactor cooldown, nitrogen is injected into the pressurizer when the system is

depressurized to approximately 50 pounds per square inch. This vapor space is maintained during long-term cooling. While the decay heat removal system is in operation, safety valves in that system are set to relieve at 450 pounds per square inch gauge, providing an automatic limit to a pressure transient. In addition, the presence of the nitrogen bubble increases the time available for the operator to respond to the transient.

We have identified five requirements for the overpressure protection system for periods of low reactor vessel temperature. The design must meet the following requirements:

- (1) Credit for operator action. No credit shall be taken for operator action until 10 minutes after the operator is made aware that a transient is in progress.
- (2) Single failure criteria. The pressure protection system shall be designed to protect the vessel, given any event initiating a pressure transient. Redundant or diverse pressure protection systems will be considered as meeting the single failure criteria.
- (3) Testability. Provisions for periodic testing of the overpressure protection system(s) and components shall be provided. The program of tests and frequency or schedule thereof will be selected to assure functional capability when required.
- (4) Seismic design and IEEE 279 criteria. Ideally, the pressure protection system(s) should meet both seismic Category I and IEEE 279 criteria. The basic objective, however, is that the system(s) shall not be vulnerable to an event that both causes a pressure transient and causes a failure of equipment needed to terminate the transient.
- (5) Reliability. The system(s) provided must not reduce the reliability of the emergency core cooling system or decay heat removal systems.

In order to assure that the Appendix G pressure limit will be greater than the maximum reactor coolant system pressure during periods of low reactor vessel temperatures, the applicant must commit to designing the overpressure protection in accordance with the staff's requirements specified in items (1) through (5), above. We will report the resolution of this matter in a supplement to this report.

5.2.4 Reactor Coolant Pressure Boundary Materials

Criteria 1 and 14 of the General Design Criteria require that the reactor coolant pressure boundary be designed, fabricated, erected, and tested so as to have an extremely low probability of a rapidly propagating failure or of a gross rupture.

In addition, they require that the reactor coolant pressure boundary be tested to quality standards commensurate with the importance of the safety function to be performed.

Our review included an assessment of the compatibility of the reactor coolant pressure boundary construction materials with the reactor coolant, contaminants, and radiolytic products to which the system will be exposed. The extent of the corrosion of ferritic low alloy steels and carbon steels in contact with the reactor coolant was reviewed. In addition, we reviewed the controls that will be used to prevent cracking of austenitic stainless steels, and the fracture toughness and welding requirements for ferritic materials.

The materials used for construction of components of the reactor coolant pressure boundary, including the reactor vessel and its appurtenances, have been identified by specification and found to be in conformance with the requirements of Section III of the ASME Code. Special requirements adopted by the applicant with regard to control of residual elements in ferritic materials, to reduce the sensitivity of the material to irradiation embrittlement, have been identified and are considered acceptable.

The materials of construction that will be exposed to the reactor coolant have been identified, and all of the materials are compatible with the expected environment, as proven by extensive testing and satisfactory performance. General corrosion of all materials, except carbon and low alloy steel, will be negligible. For carbon and low alloy steels, conservative corrosion allowances will be provided for all exposed surfaces in accordance with the requirements of the ASME Code, Section III.

The external nonmetallic insulation to be used on austenitic stainless steel components will conform with the requirements of Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steels."

Further protection against corrosion problems will be provided by control of the chemical environment. The composition of the reactor coolant will be controlled, and the proposed maximum contaminant levels have been shown, by tests and service experience, to be adequate to protect against corrosion and stress corrosion problems. The controls to be imposed on reactor coolant chemistry are in conformance with the recommendations of Regulatory Guide 1.44, "Control of Sensitized Stainless Steel," and provide reasonable assurance that the reactor coolant pressure boundary components will be adequately protected, during operation, from conditions that could lead to stress corrosion of the materials and loss of structural integrity of a component.

The instrumentation and sampling provisions for monitoring reactor coolant water chemistry will provide adequate capability to detect significant changes on a timely basis.

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The use of materials of proven performance and the conformance with the recommendations of the Regulatory Guides constitutes an acceptable basis for satisfying the requirements of General Design Criteria 14 and 31.

The controls imposed on welding preheat temperatures and weld cladding satisfy the recommendations of Regulatory Guide 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel," and Regulatory Guide 1.43, "Control of Stainless Steel Weld Cladding of Low-Allow Steels." Adoption of these recommendations provides reasonable assurance that cracking of components made from low alloy steels will not occur during fabrication, and minimizes the possibility of subsequent cracking due to the retention of residual stresses in the weldment.

The welding procedures used for ferritic steels in limited access areas comply with the objectives of Regulatory Guide 1.71, "Welder Qualification for Areas of Limited Accessibility." The ultrasonic method for examination of ferritic steel tubular products conforms to Regulatory Guide 1.66, "Nondestructive Examination of Tubular Products." The fabrication practices and examination procedures performed in accordance with these recommendations will provide reasonable assurance that welds in the reactor coolant pressure boundary will be satisfactory in locations of restricted accessibility, and that unacceptable defects in components of the reactor coolant pressure boundary will be detected regardless of shape, size or orientation.

Conformance with the above Regulatory Guides and Commission regulations constitutes an acceptable basis for meeting the requirements of General Design Criteria 1 and 14.

Within the reactor coolant pressure boundary, no components of austenitic stainless steel will have a yield strength exceeding 90,000 pounds per square inch, in accordance with the staff position.

The controls imposed upon components constructed of austenitic stainless steel, used in the reactor coolant pressure boundary and for the reactor vessel and its appurtenances, satisfy the recommendations of Branch Technical Position MTEB 5-1 on Regulatory Guide 1.31, "Control of Stainless Steel Welding," Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water Cooled Nuclear Power Plants," Regulatory Guide 1.71, "Welder Qualification for Areas of Limited Accessibility," and Regulatory Guide 1.66, "Nondestructive Examination of Tubular Products."

Materials selection, fabrication practices, examination procedures, and protection procedures, performed in accordance with these recommendations, provide reasonable assurance that the austenitic stainless steel in the reactor coolant pressure boundary will be free from hot cracking (microfissures) and will be in a

metallurgical condition that precludes susceptibility to stress corrosion cracking during service. Conformance with the Regulatory Guides and staff position constitutes an acceptable basis for meeting the requirements of General Design Criteria 1 and 14.

5.2.5 Inservice Inspection Program

Criterion 32 of the General Design Criteria requires that components, which are part of the reactor coolant pressure boundary, be designed to permit periodic inspection and testing of important areas and features, in order to assess their structural and leaktight integrity. Inservice inspection programs are based upon Section XI of the ASME Code, "Rules for Inservice Inspection of Nuclear Power Plant Components."

The applicant states that his inservice inspection program for Class 1, 2 and 3 components will be in accordance with the proper edition and addenda of the ASME Boiler and Pressure Vessel Code, as required by paragraph (g) of Section 50.55a of 10 CFR Part 50.

The design of the reactor coolant system will incorporate provisions for access for inservice inspections, in accordance with Section XI of the ASME Code. Consideration has been given to the inspectability of the system in the design of components, in the equipment layout, and in the support structures. Suitable equipment will be developed to facilitate the remote inspection of those areas of the reactor vessel not readily accessible to inspection personnel.

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

5.2.6 Reactor Coolant Pressure Boundary Leakage Detection

A limited amount of leakage is to be expected from components forming the reactor coolant boundary. Components, such as valve stem packing, circulating pump shaft seals, and flanges, are not completely leak tight. This type of leakage (identified leakage) will be monitored, limited, and separated from other leakage (unidentified).

Unidentified leakage may be an indication of a small, through-wall flaw developed in the primary coolant boundary. Changes in the unidentified leakage may represent a change in flaw size; therefore, this is a subject of safety concern.

The plant design for leakage detection will incorporate several approaches for leakage collection, separation, isolation, and detection.

During steady-state operation, the total leakage from the primary system will be monitored by the makeup tank level. Since the pressurizer level will be maintained constant, water required to compensate for leakage will be supplied from the makeup tank inventory. Identified leakage from various sources will be collected in the reactor coolant drain tank. Both the drain tank and the makeup tank will contain level indication. Thus, during steady-state operation, the operator will be able to identify total leakage and differentiate between identified and unidentified leakage.

Features will be provided, in the collection system leading to the reactor coolant drain tank, to identify flow from various sources so that corrective action can be taken if the total identified leakage becomes excessive.

Unidentified leakage within containment will be in the form of liquid or vapor. Liquid will drain to the containment building sump where it will be detectable, as an observable increase in sump level, with a sensitivity of one gallon per minute. Flow rates may be calculated from the frequency of automatic sump pump operation.

Vapor will be condensed on the coils of the containment fan cooler units, and the condensate will drain to the containment sump. Collection tanks for identified leakage and strategically located sumps for unidentified leakage will be provided in the auxiliary building.

Leakage into the secondary system will be indicated by steam line radiation monitors and by the main condenser air ejector radiation monitor. In addition, samples taken from the steam generator and condensate streams will be analyzed for both radioactivity and boron concentration. One gallon per minute leakage into the component cooling water system will be detected, within about 16 minutes, by a radioactivity monitor.

Various devices will monitor conditions in the containment. These will provide indication and, in some cases, alarms based upon rate of change of condition or level achieved. In addition to provisions for monitoring sump levels, cooling coil cooling water temperature differentials and containment temperature and pressure, the applicant has described the following: (1) containment particulate monitor, (2) containment gaseous radiation monitor, and (3) humidity detection.

We reviewed the above, and determined that the requirements of Criterion 30 of the General Design Criteria for detecting leaks are satisfied and the objectives of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," are satisfied. The sensitivity recommendations for detection, as identified in Regulatory Guide 1.45, are not explicitly satisfied and the details of how

the operator will convert some of the indications into leakage rates are not clear. We conclude that the proposed leakage detection system is acceptable at the construction permit stage of review because it meets the objectives of Regulatory Guide 1.45. Implementation of all the requirements, including sensitivity, interpretation of data, and alarms, will be reviewed at the FSAR review stage.

5.3 Reactor Vessel

5.3.1 Reactor Vessel Materials

Criterion 31 of the General Design Criteria requires that the reactor coolant pressure boundary be designed with sufficient margin to assure that, when stressed under operating, maintenance, testing, and postulated accident conditions, the boundary will behave in a nonbrittle manner and the probability of rapidly propagating fracture will be minimized.

We have reviewed material specifications for the reactor vessel and closure studs. We assessed their adequacy, for use in the construction of such components, on the basis of: material, mechanical, and physical properties; the effects of irradiation on these materials; corrosion resistance; and fabricability. We reviewed the welding controls and procedures for low alloy and austenitic steel welds.

We reviewed the fracture toughness of the ferritic materials to be used for the reactor vessel and the appurtenances thereto to assure that such components will behave in a nonbrittle manner and that the probability of rapidly propagating fracture will be minimized under operating, maintenance, testing, and postulated accident conditions. The review included the descriptions of the fracture toughness tests to be performed on all ferritic materials that will be used for the reactor vessel and appurtenances thereto, and considered the acceptability of the proposed transverse Charpy V-notch impact test specimens, dropweight test specimens, and any other test specimens included in the program.

Ferritic materials in the reactor vessel belting region, including welds, will be controlled to minimize the content of copper and phosphorus. The use of controlled composition material for the reactor vessel beltline will minimize the possibility that irradiation will cause serious degradation of its toughness properties. There is no design requirement for thermally annealing the reactor vessel. However, the design does not preclude the feasibility of in-place annealing within the design temperature limitations.

5.3.2 Pressure-Temperature Limits

We reviewed the pressure-temperature limits to be imposed upon the reactor coolant pressure boundary, during operation and testing, to assure that they will provide adequate safety margins against nonductile behavior or rapidly propagating failure

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of ferritic components of the reactor coolant pressure boundary, as required by Criterion 3i of the General Design Criteria.

The reactor will be operated in a manner that will minimize the possibility of rapidly propagating failure to comply with Appendix G, 10 CFR Part 50. The procedures for estimating the operating limitations are described in Appendix G, "Protection Against Non-Ductile Failure," Section III, ASME Boiler and Pressure Vessel Code, 1971 Edition, including Summer 1972 Addenda. Additional conservatism in the pressure-temperature limits for heatup, cooldown, testing, and core operation will be provided because these will be determined assuming that the bellline region of the reactor vessel has already been irradiated.

The use of operating limitations, based on fracture toughness tests conducted in accordance with Appendices G and H, 10 CFR Part 50, will assure adequate safety margins during operation, testing, maintenance, and postulated accident conditions. Compliance with these recommendations constitutes an acceptable basis for satisfying the requirements of Criterion 3i of the General Design Criteria.

5.3.3 Reactor Vessel Integrity

We have reviewed the factors contributing to the structural integrity of the reactor vessel, and we conclude there are no special considerations that make it necessary to consider potential vessel failure for Davis-Besse Units 2 and 3.

The basis for our conclusion is that the design, material, fabrication, inspection, and quality assurance requirements will conform to the rules of the ASME Boiler and Pressure Vessel Code, Section III, 1974 Edition, and applicable Code Cases.

The inservice inspection program will be in accordance with the revised rules in 10 CFR Part 50, Section 50.55a, paragraph (g).

5.4 Component and Subsystem Design

5.4.1 Reactor Coolant Pumps

The reactor coolant pumps will be sized to deliver flow at rates that equal or exceed the required flow rate under normal and transient operating conditions.

Sufficient pump rotational inertia will be provided, by the flywheel, to provide adequate flow coastdown following a loss of forced flow resulting from mechanical or power failures to the pumps. With such protection, the reactor neutron power can be reduced before departure from nucleate boiling limits are exceeded. The flywheel will be mounted on the reactor coolant pump motor shaft and will be designed to obtain a total moment of inertia of at least 70,000 pound-square feet for the pump rotating assembly.

The four reactor coolant pumps will be single stage, single suction, constant speed, vertical centrifugal pumps. Each pump will have a separate, single speed, top mounted electric drive motor connected to the pump by a removable shaft coupling. Each motor will be equipped with an anti-reverse device to prevent back-rotation of the pump.

Criterion 4 of the General Design Criteria requires that structures, systems, and components of nuclear power plants important to safety be protected against the effects of missiles that might result from equipment failures. Because flywheels have large masses and rotate at speeds of about 1,200 revolutions per minute during normal reactor operation, a loss of integrity could result in high energy missiles and excessive vibration of the reactor coolant pump assembly. The safety consequences could be significant because of possible damage to the reactor coolant system, the containment, or the engineered safety features.

The probability of a loss of pump flywheel integrity will be minimized by the use of suitable material, adequate design, and inservice inspection. The applicant has stated that the pump flywheels will be designed, fabricated, tested and inspected in conformance with the recommendations of Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity."

The methods of failure analysis will be based upon theories of fracture mechanics and Appendix G to Section III of the ASME Boiler and Pressure Vessel Code.

The applicant has stated that the integrity of the reactor coolant pump flywheel will be assured by designing it to 125 percent of the normal synchronous speed of the motor (i.e., 1500 revolutions per minute). The lowest design operating temperature is specified to be 120 degrees Fahrenheit. The combined primary stresses at operating speed will not exceed 33-1/3 percent of the materials yield strength, as measured in the weak direction at the normal operating speed. The shaft and bearings supporting the flywheel will be designed to remain operational under any combination of normal operating loads, anticipated transients, and safe shutdown earthquake. In addition, a 100 percent ultrasonic volumetric inspection of the flywheel will be performed using acceptance criteria of Section III of the ASME Boiler and Pressure Vessel Code for Class 1 components.

The potential for the pump flywheel to become a missile, in the event of a rupture in the pump suction or discharge sections of reactor coolant system piping, is under generic study by the staff. The Electrical Power Research Institute has contracted with Combustion Engineering to perform a 1/5-scale reactor coolant pump research program. The objective of the program, in part, is to obtain empirical data to substantiate or modify current mathematical models used to predict pump performance during a postulated loss-of-coolant accident. Results from the program are expected in 1978.

If the results of the generic investigation indicate that additional protective measures are warranted to prevent excessive pump overspeed or to limit potential consequences to safety-related equipment, we will determine what modifications, if any, are necessary to assure that an acceptable level of safety is maintained. Should additional protective measures be warranted, the applicant will be required to comply with the design modifications.

The probability of a loss of pump integrity will be minimized by the use of suitable material, adequate design, and preservice inspection. The selection of materials, fracture toughness tests, design procedures, and preservice overspeed spin testing program for reactor coolant pump flywheels, have been reviewed and found acceptable on the basis of conformance with Regulatory Guide 1.14. During the Final Safety Analysis Report stage, we will review the accessibility for inspection of the flywheels.

The use of suitable materials with adequate fracture toughness, conservative design procedures, preservice testing, and inservice inspection of flywheels for reactor coolant pump motors, provide reasonable assurance of the structural integrity of the flywheels in the event of design overspeed transients or postulated accidents. Conformance with the recommendations of Regulatory Guide 1.14 constitutes an acceptable basis for satisfying the applicable portions of Criterion 4 of the General Design Criteria.

5.4.2 Steam Generators

The two steam generators will be vertical straight-tube-and-shell heat exchangers, producing superheated steam at constant turbine throttle pressure over the operating power range. The reactor coolant will enter the steam generator upper hemispherical head and flow downward inside the tubes, giving up heat to generate steam on the shell side. The steam generators will provide a heat sink for the reactor coolant system, and they will be located at a higher elevation than the core to improve natural circulation for decay heat removal.

The components in the steam generator will be classified ASME Boiler and Pressure Vessel Code Class 1 and 2, depending upon their location in the primary and secondary coolant systems, respectively. The materials to be used in Class 1 and Class 2 components of the steam generators will be selected and fabricated according to codes, standards, and specifications acceptable to the staff. The onsite cleaning and cleanliness controls during fabrication will conform to the recommendations of Regulatory Guide 1.37, "Cleaning of Fluid Systems and Associated Components During the Construction Phase of Nuclear Power Plants." The controls to be placed upon secondary coolant chemistry are in agreement with our established technical positions. Conformance with applicable codes, standards, staff positions, and Regulatory Guides constitutes an acceptable basis for meeting the applicable requirements of Criteria 14, 15 and 31 of the General Design Criteria.

The inservice inspection of Code Class 2 and 3 components will be in accordance with the proper edition and addenda of the ASME Code, as required by 10 CFR Part 50, Section 50.55a, paragraph (g).

Conformance with the applicable codes, standards, staff positions, and Regulatory Guides constitutes an acceptable basis for meeting the applicable requirements of Criteria 1, 14, 15, 31 and 32 of the General Design Criteria.

5.4.3 Pressurizer

The pressurizer will maintain the reactor coolant system pressure during steady-state operation and will limit pressure changes during transients. It will contain a water volume sized to permit the reactor system to experience a reactor trip and not uncover the low level sensors in the bottom head, while also maintaining the pressure above the activation point for the high pressure injection system. The steam volume will be sized to provide the capability of the system to experience a turbine trip without filling the pressurizer to more than 90 percent of its total volume.

Electric heater bundles in the lower section of the pressurizer, and water spray nozzles in the top head, will maintain the steam and water at the saturation temperature that corresponds to the desired reactor coolant system pressure. During outsurges the system pressure decreases; some of the water will flash to steam, limiting the pressure decrease, and the electric heaters will act to restore the normal operating pressure. During insurges the system pressure increases; some steam will condense, limiting the pressure increase, while the automatic water spray will condense more steam to reduce the pressure to the normal operating level.

Two ASME Code, Section III, relief valves will be connected to the pressurizer to relieve system overpressure. Each valve will have one-half the required relieving capacity. An additional pilot-operated relief valve will be provided to limit the lifting frequency of the code relief valves. All three relief valves will discharge to the pressurizer quench tank within the containment.

We find the design of the pressurizer to be acceptable.

5.4.4 Decay Heat Removal System

In normal shutdown operations, the decay heat removal system will be used in conjunction with the main steam system, the feedwater system and the auxiliary feedwater system to cool down the reactor following power operation. The decay heat removal system will consist of a line, from a reactor coolant system hot leg, that feeds two parallel cooling circuits, each containing a pump and a heat exchanger. The cooled water will be returned to the reactor through the core

flooding nozzles. The heat load will be transferred to the component cooling water system.

The decay heat removal system circuits will each have the same flow and cooling capacity. Taken together, the decay heat removal system will be sized to cool the reactor coolant system from 280 to 140 degrees Fahrenheit in 14 hours, beginning 6 hours after shutdown. During the first six hours after reactor shutdown, heat removal will be through the secondary system. Failure of one decay heat removal train will only increase the time required to reach 140 degrees Fahrenheit. One circuit will be more than ample to remove decay heat with adequate margin relative to fuel design limits. This is in conformance with the single failure requirement of Criterion 34 of the General Design Criteria.

We have reviewed the piping and instrumentation drawings to evaluate the vulnerability of the two parallel circuits to a single failure, with and without offsite power available. The two decay heat removal pumps will be connected to separate emergency power sources, so that a single failure will not preclude starting at least one pump.

Isolation on the suction side of each decay heat removal circuit will be provided by two motor-operated valves inside of containment. These valves will be interlocked with the reactor coolant system pressure to prevent opening above a reactor coolant pressure of 280 pounds per square inch, and will be interlocked to automatically close at that pressure. These interlocks will assure isolation of the decay heat removal system from the high pressure reactor coolant system under normal operating conditions. The power supply arrangement to these isolation valves will be from diverse sources to assure that no single failure will prevent opening or isolating the letdown line.

The discharge side of each circuit will be isolated by two check valves in series. These valves will be leak tested, periodically, to assure their integrity. These high-low pressure isolation features comply with Branch Technical Position RSB 5-1, "Design Requirements of the Residual Heat Removal System," the requirements of Criterion 34, and the containment isolation requirements of Criteria 54, 55 and 56 of the General Design Criteria.

Each circuit will contain a relief valve set to relieve at 320 pounds per square inch gauge. Valve capacity will be sufficient to relieve the flow from the inadvertent actuation of the high pressure safety injection pumps. This event has been identified by the applicant as the design basis anticipated occurrence.

As an engineered safety features system, all elements will be designed to seismic Category I requirements and will be located within structures designed in conformance with Criteria 2 and 4 of the General Design Criteria. The electrical

controls and instrumentation will meet the requirements of IEEE Standard 279, as discussed in Section 7.0 of this report.

Under emergency conditions, the decay heat removal system will form the low pressure injection portion of the emergency core cooling system. In addition to the emergency core cooling system function, the decay heat removal system will function during refueling to maintain refueling temperature, and will provide a means for filling and draining the refueling cavity. It will provide initial reactor coolant system circulation prior to startup. It will provide cooled auxiliary spray to the pressurizer for complete depressurization after shutdown of the reactor coolant pumps. These different and separate functions do not compromise the capability of the system to provide adequate decay heat removal.

In our Report to the Advisory Committee on Reactor Safeguards in July 1977 concerning the BSAR-205 plant (Docket No. STN 50-561), we stated that we were considering, on a generic basis, whether the capability should be provided for transferring heat from the reactor to the environment, during the transition from normal reactor operating conditions to cold shutdown, using only safety-grade systems, and assuming (1) only offsite or onsite power is available, and (2) the most limiting single failure has occurred. We also stated that we might require the BSAR-205 design and the designs of the balance-of-plant portions of applications referencing BSAR-205 to be modified to provide such cold shutdown capability.

In the period following our Report to the Advisory Committee on Reactor Safeguards, further staff work led to our decision to require Davis-Besse Units 2 and 3 to have the capability to be taken to a cold shutdown condition in approximately 36 hours, using only safety-grade equipment, assuming a loss of onsite or offsite power and assuming a single failure.

The requirements applicable to the Davis-Besse Units 2 and 3 design are:

- (1) Provide safety-grade steam generator dump valves, operators, air and power supplies which meet the single failure criterion.
- (2) Provide the capability to cool down to cold shutdown in less than 36 hours, assuming the most limiting single failure and with only offsite or onsite power available, or show that manual actions inside or outside containment or return to hot standby until the manual actions or maintenance can be performed to correct the failure provides an acceptable alternative.
- (3) Provide the capability to depressurize the reactor coolant system with only safety-grade systems assuming a single failure and with only offsite or onsite power available, or show that manual actions inside or outside containment or remaining at hot standby until manual actions or repairs are complete provides an acceptable alternative.

- (4) Provide the capability for boration with only safety-grade systems assuming a single failure and with only offsite or onsite power available, or show that manual actions inside or outside containment or remaining at hot standby until manual action or repairs are completed provides an acceptable alternative.
- (5) Conduct or reference approved prototype qualification tests to study the mixing of the added borated water and the cooldown under natural circulation conditions with a worst-case single failure (i.e., a single failure of a steam generator atmospheric dump valve). These tests and analyses will be used to obtain information on cooldown times and the corresponding auxiliary feedwater requirements.
- (6) Provide specific procedures, at the operating license review stage, for cooling down using natural circulation, and submit a summary outline of these procedures during the construction permit review.
- (7) Provide or require a seismic Category I auxiliary feedwater supply for at least four hours at hot shutdown plus cooldown to the decay heat removal system cut-in based upon the longest time (for only onsite or offsite power and assuming the worst single failure), or show that an adequate alternate seismic Category I source will be available.

The implementation of these requirements is being pursued with the applicant and Babcock and Wilcox. We will report on this matter in a supplement to this report.

5.5 Loose Parts Monitor

The applicant has provided a commitment to install a loose parts monitoring system. We will require a more detailed description of equipment and procedures at the operating license stage of review.

Recently, prototype loose parts monitoring systems have been developed and are presently in operation or being installed at a number of plants. As a result of a study we completed on the installation of, and experience with, loose parts monitoring systems in operating plants, we have identified the following aspects for a loose parts monitoring system which we will use to assess the acceptability of the specific system to be provided for Davis-Besse Units 2 and 3 when we review the detailed information to be submitted in the Final Safety Analysis Report:

- (1) The description of the loose parts monitoring system shall include the location of all sensors and the method for monitoring them. A minimum of two sensors will be required at each natural collection region. For example, in a pressurized water reactor, two sensors should be included at the top and at the bottom of the reactor vessel and at each steam generator primary coolant inlet.

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- (2) The description of the monitoring equipment shall include the levels and the basis for the alarm settings. In addition, the manufacturer's sensitivity specifications for the equipment shall be provided. Anticipated major sources of internal and external noise shall be identified along with the plans for minimizing the effects of these sources on the ability of the monitoring equipment to perform its intended function.

- (3) The loose parts monitoring system shall be required to function after any seismic event for which plant shutdown is not required. The procedures of Regulatory Guide 1.100, "Seismic Qualification for Electric Equipment for Nuclear Power Plants," are acceptable for demonstrating the seismic qualification of this system. An exception of this seismic qualification is that recorders are not required to function within their specified accuracy during or after seismic events without maintenance. However, monitoring (alarm or indication) capability must remain available for that channel at all times during and after the seismic event. A description of the precautions to be taken to assure the operability of the system after an operating basis earthquake shall be provided.

Our experience has indicated that the detailed design information on loose parts monitoring systems is not required at the construction permit stage of our review. The applicant's commitment is acceptable at the construction permit stage of review, since the additional information required is of the type that can be submitted later in accordance with 10 CFR Part 50.35.

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6.0 ENGINEERED SAFETY FEATURES

6.1 Design Considerations

Systems and design features that will be provided to prevent and reduce the release of fission products are called engineered safety features. These engineered safety features are intended to function during or following postulated accidents. They will be designed to contain the fission products that might be released from the reactor fuel, to mitigate the damage to the fuel cladding and other fission product barriers, to provide protection for the station personnel, and to provide for fission product removal and cleanup within the plant structures. This section describes our review of the containment systems, the emergency core cooling system, and the control room habitability system.

Systems and components designated as engineered safety features will be designed to be capable of assuring safe shutdown of the reactor under the adverse conditions of the various postulated design basis accidents described in Section 15.0 of this report. They will be designed, therefore, to seismic Category I standards, and must function even with complete loss of offsite power. Components and systems will be provided in sufficient redundancy so that a single failure of any active component or system will not result in the loss of the capability to perform the safety function. The instrumentation systems and emergency power systems will be designed to the same seismic and redundancy requirements as the systems they serve. These systems are described in Sections 7.0 and 8.0 of this report.

6.2 Containment Systems

The containment systems will include the containment vessel, containment heat removal system, containment isolation system, containment combustible gas control system, and the provisions for containment leakage rate testing.

6.2.1 Containment Functional Design and Analysis

The containment will consist of a free standing steel vessel surrounded by a reinforced concrete shield building. The containment vessel will have a net free volume of 2.8 million cubic feet. The vessel will house the nuclear steam supply system, which includes the reactor vessel, reactor coolant piping, reactor coolant pumps, pressurizer, and steam generators, as well as certain components of the plant engineered safety feature systems. The containment vessel will be designed for an internal pressure of 39 pounds per square inch gage and a temperature of 264 degrees Fahrenheit.

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Containment Analysis

The applicant has analyzed the containment pressure response to postulated pipe break accidents in the following manner. Mass and energy release rates for postulated reactor coolant system pipe breaks were calculated using the CRAFT computer program for the blowdown and reflooding periods. These rates were then input to the COPATTA computer program, which calculates the containment pressure transient. The calculations were made conservative for containment analysis by maximizing the rate of heat transfer from the core and by adjusting the calculated mass and energy release rate from the break so that no quenching of steam by the safety injection fluid would be included. By this method, all steam generated by the primary system would be available for release to the containment.

We have determined that the heat transfer coefficient used in the steam generators to calculate steam production in the primary system is not conservative. During the latter period of blowdown and during the reflooding period, reverse heat flow from the secondary system will produce boiling in the primary system, which will provide an additional steam source to the containment. The applicant has based this calculation on forced convection heat transfer without boiling. We believe that nucleate boiling heat transfer should have been assumed. The applicant has calculated that the peak containment pressure would be increased by approximately one pound per square inch as a result of this assumption. We have concluded that the applicant's mass and energy release data is acceptable because, as noted below, the peak calculated pressure is increased by one pound per square inch.

The analytical model used for the containment pressure response analysis, including the assumptions made regarding the availability of heat removal systems and structural heat sinks, has been described in the Preliminary Safety Analysis Report. The applicant has analyzed reactor coolant system pipe break accidents for a spectrum of break locations and sizes. The postulated double-ended break of the hot leg piping of the reactor coolant system resulted in the highest calculated containment pressure of 38 pounds per square inch gage, including the one pound per square inch increase discussed above.

We have also analyzed the containment pressure response to a postulated double-ended break of the hot leg piping of the reactor coolant system, using the CONTEMP-LT MOD 26 computer code. Our analysis was based on the mass and energy release, and the containment structural heat sink and spray system performance data, provided by the applicant. We used conservative condensing heat transfer coefficients to the structures inside containment. The results of our analysis confirm the applicant's results. Although the margin between the peak calculated pressure and the containment design pressure is less than the 10 percent (one pound per square inch rather than 3.8) margin normally required for construction permit applications, we have accepted the containment design pressure because the containment for Units 2 and 3 will duplicate the "as built" containment for Unit 1 and the applicant has used the condition of the "as built" containment in its evaluation.

The applicant has also analyzed a spectrum of main steam line break accidents to determine the limiting pressure and temperature responses. The mass and energy release to the containment was calculated using the TRAP-2 code, which describes the secondary and primary systems. The applicant has stated that the version of the TRAP-2 code used for the analysis is the same as was approved by us for B-SAR-205 (Docket No. STN 50-561). The method used in B-SAR-205 maximizes the energy release to the containment by assuming that only steam flows through the break. The steam is superheated by the upper portion of the tube bundle before being released into the containment. We have concluded that this method conservatively maximizes the rate of mass and energy flow to the containment.

The applicant calculated a peak containment pressure of 32.5 pounds per square inch gauge for a postulated double-ended main steam line break, and a calculated peak containment temperature of 439 degrees Fahrenheit for a postulated 1.5 square-foot main steam line break. In determining the main steam line breaks that resulted in the highest containment temperature and pressure, the applicant assumed that the feedwater control valves and the main steam nonreturn valves would function to terminate the addition of feedwater and steam to the affected steam generator. However, both the feedwater control valves and the nonreturn valves are nonsafety grade; i.e., they are Quality Group D. Pending completion of the staff's generic study, "PWR Main Steam Line Break -- Core, Reactor Vessel, and Containment Building Response," we require that, if these valves are to function to mitigate the consequences of a main steam line break accident, they be Quality Group B or better and be operated by electrical instruments and controls that meet IEEE Standard 279-1971. As a result, we conclude that the main steam line break analysis for the peak containment pressure calculation is unacceptable.

The applicant committed, by letter dated March 7, 1978, to qualify safety-related equipment to the most severe environmental conditions that are predicted to exist for each particular piece of equipment. In addition, the applicant has submitted an analytical model for the thermal analysis of equipment to determine the acceptability of the qualification testing that may have been done and to identify the need to do additional qualification testing or protect the equipment.

We have concluded that the applicant's commitment, regarding the qualification of safety-related equipment, is acceptable for the construction permit stage of review. However, we have not reviewed the applicant's analytical model for calculating the thermal response of equipment. We are involved in a generic program to determine acceptable methods of component thermal analysis. We expect that, within approximately one year, the generic program will be complete, resulting in the establishment of consistent environmental qualification requirements for all plants, including Davis-Besse Units 2 and 3.

Containment Subcompartment Analysis

The applicant has analyzed the pressure response, of containment interior compartments, to postulated high energy line breaks. The compartments investigated include the reactor cavity, and the steam generator compartment that also houses the pressurizer.

The applicant has provided short-term mass and energy release data for the limited displacement breaks of the hot and cold legs. The CRAFT code was used in these calculations, and the break flow was calculated using the Moody critical flow correlation with a multiplier of 1.0. The flow from the vessel side of the break is calculated assuming no flow losses, which results from locating the break in the reactor vessel rather than in the coolant piping. This assumption increases the flow rate.

Comparison with experimental data has indicated that the Moody critical flow model may not be conservative when the upstream conditions are subcooled. This comparison is described in the modified Zaloudek correlation, as discussed in TREE-NUREG-1006, "A Study of Critical Flow Prediction for Semi-Scale MOD-1 Loss-of-Coolant Accident Experiments," December 1976. The modified Zaloudek correlation has been approved by us for subcooled flow in B-SAR-205, and we have found it to be conservative in comparison with experimental data for subcooled flow.

The applicant provided a comparison between the hot and cold leg flows predicted with the approved B-SAR-205 method and the break flows predicted with the Davis-Besse method with no flow losses. The B-SAR-205 method locates the break in the piping instead of in the reactor vessel. The Davis-Besse method was shown to be more conservative. We have, therefore, concluded that the mass and energy release data, predicted for postulated hot and cold leg breaks, are conservative for subcompartment analysis of Davis-Besse Units 2 and 3.

The applicant has committed to increase the calculated pressures by 40 percent for use in the design of the subcompartment structures. However, the applicant has not specifically identified the results to which the factor of 1.4 will be applied. Further, the nodalization sensitivity study, performed for the reactor cavity in the vicinity of the break, is inadequate to assure that the maximum pressure loads, acting on compartment walls and components, have been conservatively predicted.

Preliminary Safety Analysis Report Sections 6.2.1.3.2.6.2 and 6.2.1.3.2.6.3 discuss the nodalization sensitivity studies performed to determine the minimum number of volume nodes required to conservatively predict the maximum pressure load acting on the compartment walls and major component supports for the reactor cavity and steam generator compartments. The discussion is inadequate in that it does not clearly distinguish whether the sensitivity studies apply to the loads acting on the compartment walls or on the component supports. The applicant must clarify the discussion

in these two Preliminary Safety Analysis Report sections to (1) differentiate between the study that applies to the compartment walls and the study that applies to component supports, and (2) clearly demonstrate the conservatism of the model with respect to the loads acting on the compartment walls and the component supports.

Also, the nodalization sensitivity study did not determine the sensitivity of increasing the number of volume nodes in the vicinity of the pipe break, nor did it address (1) a postulated rupture, in the reactor cavity, of piping that connects the core flood tanks to the reactor coolant system, and (2) postulated ruptures, in the pressurizer compartment, of the surge line and the spray lines.

As a result of our concerns, the applicant submitted, by letter dated March 30, 1978, additional studies of subcompartment pressurization. We have not yet completed our review of this new information.

We will report on the resolution of the above stated matters for the containment subcompartment analysis, including the acceptability of the design pressures for the subcompartments and the component supports analysis, in a supplement to this Safety Evaluation Report.

Containment External Differential Pressure Analysis

The applicant has analyzed the consequences of inadvertent actuation of the containment spray system in the containment. The containment atmosphere was assumed to be initially at a pressure of 14.7 pounds per square inch gage, a temperature of 120 degrees Fahrenheit, and a relative humidity of 10 percent. The spray water was also conservatively assumed to be at a temperature of 50 degrees Fahrenheit.

In the analysis, the applicant assumed only one of the two containment spray trains would be actuated; the system design will preclude any single failure of single operator action that will cause operation of more than one train. The applicant further assumed that one of the two banks of vacuum relief valves, i.e., five of the ten vacuum relief valves, failed to open. These valves will be designed to open when a differential pressure of 0.15 pounds per square inch occurs across them.

The applicant calculated a maximum pressure differential of 0.24 pounds per square inch, which is less than one-half the containment vessel design external differential pressure of 0.5 pounds per square inch. Based on our review of the applicant's analysis, we conclude that the containment vessel design differential pressure of 0.5 pounds per square inch is acceptable.

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Secondary Containment Functional Design

The secondary containment, the shield building, will be a reinforced concrete structure surrounding the steel containment vessel. The annulus between these two structures contains a volume of 678,700 cubic feet. Potential leakage from the containment vessel to the shield building and adjoining penetration rooms will be collected and processed by the emergency ventilation system. Following a loss-of-coolant accident, the emergency ventilation system will maintain the annulus region at a negative pressure to assure the collection of leakage from the containment.

The applicant has committed to confirm the operability of the emergency ventilation system components and equipment as part of the preoperational and periodic inservice inspection and test programs. The test programs will assure that the emergency ventilation system is capable of maintaining the shield building annulus under a minimum negative pressure of 0.25 inch of water gage with one system train operating. At the operating license stage, we will review the parameters to be monitored to verify the operability of the emergency ventilation system performance.

The applicant has analyzed the pressure response of the shield building annulus following a postulated loss-of-coolant accident inside the containment. The analysis shows that a negative pressure of 0.25 inch of water gauge will be established by the emergency ventilation systems, in about 650 seconds assuming: (1) only one train is operable; (2) no outleakage during the positive pressure transient; and (3) inleakage during the negative pressure transient. Furthermore, in the offsite radiological dose calculations, the applicant conservatively assumed that the negative pressure of 0.25 inch water gauge will not be established until 780 seconds. We have reviewed the applicant's analysis of the annulus drawdown time and conclude it is acceptable.

The applicant has identified all the high energy lines that pass through the annulus space of the shield building. They include the main steam lines, main feedwater lines, and the letdown lines from the reactor coolant system. The applicant has committed to providing guard pipes on all high energy lines to preclude overpressurization of the annulus space in the event of a pipe break. We find this to be an acceptable commitment.

The applicant has identified potential leak paths from the containment vessel that would bypass the volume to be treated by the emergency ventilation system, and has specified that the total bypass leakage will be less than 0.03 weight percent of the containment atmosphere per day. The potential bypass leak paths were determined using the guidelines of Branch Technical Position CSB 6-3, "Determination of Bypass Leakage in Dual Containment Plants." We concur with the applicant's identification of the bypass leak paths.

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Except for the penetrations associated with the secondary side of the steam supply system, the bypass leakage paths will be leak tested in accordance with the requirements of 10 CFR Part 50, Appendix J, for the Type B and Type C tests. Rather than leak test these bypass leak paths, the applicant proposes to conservatively calculate the post-loss-of-coolant accident leakage through cracks that may exist in the steam generator tubes. The applicant's proposal is still under review. We will report on the resolution of this matter in a supplement to this Safety Evaluation Report.

Conclusions

We have evaluated the proposed containment functional design for conformance with the General Design Criteria; in particular, Criteria 16 and 50. The proposed containment internal design pressure of 39 pounds per square inch gage and design temperature of 269 degrees Fahrenheit are acceptable for the loss-of-coolant accident. The applicant's commitment for qualifying safety-related equipment inside containment is acceptable. The feedwater control valves and the steam nonreturn valves must be designed and fabricated to meet quality Group B requirements.

The applicant has submitted additional information with regard to the subcompartment analysis.

We will report further on these matters in a supplement to the Safety Evaluation Report.

6.2.2 Containment Heat Removal Systems

The containment heat removal system will consist of the containment spray system and the containment air cooling system. These systems will be designed to reduce the containment pressure and temperature following a postulated high-energy line break accident within the containment. The containment air cooling system will also be used during normal plant operation, whereas the containment spray system has no normal operating function.

The containment spray system will consist of two separate trains of equal capacity. All active components of the system will be located outside the containment vessel to facilitate maintenance operations. Missile protection will be provided by direct shielding or physical separation of equipment. The system is classified seismic Category I and Quality Group B. The containment spray pump recirculation intake, in each of the containment emergency sumps, will be enclosed by a screen assembly to prevent the entry of debris that could clog the spray nozzles. The sump design meets all of the recommendations of Regulatory Guide 1.82, "Sumps for Emergency Core Cooling and Containment Spray Systems," except that the water velocity at the inner screen is calculated to be 0.5 feet per second rather than less than

0.2 feet per second, and a three-inch curb will be provided around the sumps instead of sloping the floor away from them.

The applicant has provided an analysis that shows that, for the proposed sump design, debris will settle before reaching the screening. Also, due to space limitations, sloping the floor is impractical and would not be as effective a barrier against drawing debris into the sump as would be a three-inch curb. Based on our review of the applicant's justification for deviating from the recommendations of Regulatory Guide 1.82, we have concluded that the proposed sump design is acceptable for Davis-Besse Units 2 and 3.

A high-high containment pressure signal, in conjunction with either a high containment pressure signal or a low reactor coolant pressure signal from the safety feature actuation system, will automatically actuate the containment spray system. The system pumps and valves will also be manually operable from the control room. The spray pumps will take suction initially from the borated water storage tank. When the water in the tank reaches a low level, a switchover from the injection mode to the recirculation mode will be initiated.

The applicant has not finalized the design of the containment spray system recirculation piping nor has he procured the containment spray pumps. Therefore, the applicant has not provided an analysis that demonstrates that sufficient net positive suction head will be available to the spray pumps for both the injection and recirculation modes of operation. However, the applicant has committed to perform the suction head calculation in accordance with Regulatory Guide 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps," and report the results in the Final Safety Analysis Report. We have concluded that the applicant's commitment to meet Regulatory Guide 1.1 is acceptable for the construction permit stage of review.

The containment air cooling system will consist of three equal capacity air cooler units. The components and equipment required to remain operable following an accident will be located outside the secondary concrete shield for missile protection and at an elevation that precludes flooding. They will be designed to withstand the differential pressures resulting from a loss-of-coolant accident. The system will be classified seismic Category I. A high containment pressure signal or a low reactor coolant system pressure signal from the safety features actuation system will automatically actuate the containment air cooling system. The system will also be manually operable from the control room.

Based on our review of the containment heat removal system, we conclude that the system will be designed in accordance with the requirements of Criteria 38, 39 and 40 of the General Design Criteria, and is, therefore, acceptable.

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6.2.3 Containment Isolation System

The containment isolation system will be designed to automatically isolate the containment atmosphere from the outside environment following postulated accidents. Double barrier protection, in the form of closed systems and isolation valves, will be provided to assure that no single active failure will result in the loss of containment integrity.

The containment isolation provisions, including the isolation valving and piping that penetrates containment, will be seismic Category I.

The containment isolation provisions for the lines penetrating containment must conform to the requirements of General Design Criteria 54, 55, 56 or 57, as appropriate. As permitted by Criteria 55 and 56, there will be containment penetrations whose isolation provisions do not satisfy the explicit requirements of the General Design Criteria but that are acceptable on some other defined basis. These penetrations are discussed below.

Each containment vessel vacuum breaker line will have one motor-operated isolation valve and one check valve in series outside the containment vessel, between the vessel and the shield building. These two valves will provide double-barrier protection. The safety function of the vacuum breaker lines is to prevent the containment from being depressurized below its design limit. Locating both valves outside containment improves system reliability and facilitates surveillance testing. Therefore, we have concluded that the isolation provisions for these lines provide an acceptable "other defined" basis for satisfying the requirements of Criterion 56 regarding the location of the isolation valves.

The containment vessel leak test inlet lines each will have a locked closed manual isolation valve outside containment and a blind flange at each end of the piping. We have concluded that a blind flange is an appropriate substitute for an isolation valve and provides an acceptable "other defined" basis for satisfying the requirements of Criterion 56.

The fuel transfer tube will have one blind flange with a double O-ring seal located inside the containment. We have determined that, from a functional standpoint, the fuel transfer tube is not a piping system but is actually an equipment hatch. Therefore, General Design Criterion 56 is not applicable in establishing the isolation requirements for the fuel transfer tube, and we conclude that a single blind flange, with testable double O-ring seals, is acceptable.

The chemical cleaning lines will contain one blind flange inside and one outside the containment to provide a double barrier. We have concluded that the blind flanges are appropriate substitutes for isolation valves.

The containment vessel emergency sump recirculation lines each will contain one motor operated valve outside containment. The recirculation system forms a closed system outside containment. Double-barrier protection will be provided by the valve in each line and the closed system. Since these lines will have a post-accident safety function, we have concluded that system reliability is greater with only one isolation valve in each line and that the isolation provisions provide an acceptable "other defined" basis for satisfying the requirements of Criterion 56.

The decay heat pump suction lines each will contain a remote manual isolation valve and a safety relief valve inside containment. One remote manual valve and one manual locked open valve will be outside containment. Each of these lines will lead to a closed safety grade system outside the containment. The relief valve setpoint is 1.5 times the containment design pressure. Since these lines will have a post-accident safety function, we have concluded that remote manual isolation capability provides an acceptable "other defined" basis for satisfying the automatic isolation requirements of Criterion 56.

We had also concluded that the containment vessel hydrogen purge outlet line does not meet General Design Criterion 56. The applicant had proposed to have two motor operated isolation valves in series outside containment, on the basis that system reliability will be greater with this arrangement. However, the applicant's evaluation of the time after a loss-of-coolant accident at which the containment purge outlet line will be opened for purging indicates that these valves need not be opened until months after the accident. Therefore, in order to meet the provisions of Criterion 56, we required a commitment that the purge line have one isolation valve outside containment and one inside. The applicant provided such a commitment in a letter dated February 28, 1978.

Containment isolation, except for systems needed for operation of engineered safety features, will occur automatically upon receipt of containment high pressure signals or reactor coolant low pressure signals from the safety features actuation system.

The containment purge system lines will also isolate from redundant, safety-grade high radiation signals. All power-operated isolation valves will have position indication in the control room.

Our review of the containment isolation system has also included the functional capability of the proposed containment purge system, which will function to reduce airborne radioactivity in the containment, limit radiation exposure to operating personnel, and provide outside air to the containment during extended periods of occupancy.

The containment purge system will consist of a high capacity system and a low capacity system. However, the high capacity system will not be operated during

normal operation and, therefore, the isolation valves in this system will be closed during normal operations.

The low capacity system will provide the purging capability during normal plant operation. The system will have a single supply line and a single exhaust line with 18-inch valves.

The applicant has provided an analysis of the consequences of a loss-of-coolant accident occurring while the containment is being purged. The analysis uses the guidelines of Branch Technical Position CSB 6-4, "Containment Purging During Normal Plant Operations." We have reviewed the applicant's analysis and have concluded that the purge system design meets the recommendations of Branch Technical Position CSB 6-4.

Based on our review, we conclude that the design of the containment isolation system conform to Criteria 54, 55, 56 and 57 of the General Design Criteria and the recommendations of Regulatory Guide 1.11, "Instrument Lines Penetrating Primary Reactor Containment," and is, therefore, acceptable.

6.2.4 Combustible Gas Control System

Following a loss-of-coolant accident, hydrogen may accumulate inside the containment as a result of (1) a chemical reaction between the fuel rod cladding and the steam resulting from vaporization of emergency core cooling water, (2) corrosion of construction materials by the spray solution, and (3) radiolytic decomposition of the cooling water in the reactor core and the containment sump.

The combustible gas control system will be designed to control the concentration of hydrogen within the containment vessel following a loss-of-coolant accident. The applicant proposes a system that will consist of the containment hydrogen dilution system, hydrogen purge system, recirculation system, and gas analyzer system.

The proposed containment hydrogen dilution system will dilute the hydrogen concentration within the containment vessel by adding air to the containment. The addition of air to the containment will result in an increase in the containment pressure. Eventually the containment atmosphere will be purged to the environment.

The system will be seismic Category I and will consist of redundant trains. The system blowers will have a capacity of 100 standard cubic feet per minute. The maximum pressure to which the system blowers will be capable of repressurizing the containment vessel is 18 pounds per square inch gage.

By our letters dated January 17, 1977 and May 25, 1977, we informed the applicant that his proposed method for combustible gas control was not acceptable, based upon current staff requirements for combustible gas control systems for proposed plants

subsequent to 1970. The applicant appealed our position in a meeting held on September 29, 1977. As a result of this meeting, the applicant submitted new information in the Preliminary Safety Analysis Report, concerning hydrogen production and accumulation in the containment vessel following a loss-of-coolant accident, to justify their proposed method for combustible gas control. The applicant's analysis shows that a hydrogen concentration of 3.5 volume percent would be reached 41 days following a loss-of-coolant accident. At that time, the hydrogen dilution system would be used to repressurize the containment and delay the need for purging the containment until about one year after the accident.

We have performed a similar analysis of the production and accumulation of hydrogen in the containment following a loss-of-coolant accident using more conservative material corrosion rates. Our confirmatory analysis predicts that a hydrogen concentration of 3.5 volume percent would occur sooner; i.e., at about 17 days after the accident. Although a hydrogen dilution system may effectively control the hydrogen concentration, we see no reason at this time to change our policy regarding a repressurization or purge systems. Therefore, in keeping with our current policy that repressurization or purge systems are not acceptable for combustible gas control, we require that hydrogen recombiners be included in the Davis-Besse Units 2 and 3 design. The staff is considering a program to evaluate the benefits and effectiveness of repressurization systems in conjunction with a study of alternate safety features.

A backup hydrogen purge system, consisting of a single train, will be provided. The system will relieve the containment vessel through particulate and charcoal filters to the station vent. The system will be seismic Category I.

The containment recirculation system will be designed to draw air from the containment vessel dome and discharge it toward the containment air coolers, to assure a more uniform concentration of hydrogen in the containment. The system will be seismic Category I and will consist of redundant trains.

The gas analyzer system will be designed to monitor the hydrogen concentration within the containment vessel following a loss-of-coolant accident. The system will be seismic Category I and will consist of redundant trains. Samples can be drawn from four points in the containment vessel.

Based on our review of the systems provided for combustible gas control following a loss-of-coolant accident, we have concluded that, subject to the inclusion of a hydrogen recombiner in the design, the hydrogen purge system, containment recirculation system, and containment gas analyzer conform to the guidelines of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," and the requirements of General Design Criteria 41, 42, and 43, and are, therefore, acceptable.

6.2.5 Containment Leakage Testing Program

Appendix J to 10 CFR Part 50 specifies the containment leakage testing requirements. Adherence to these requirements provides adequate assurance that the containment leak-tight integrity can be verified throughout the plant service lifetime and that the leakage rates will be periodically checked on a timely basis to maintain such leakage within the specified limits. Maintaining containment leakage within such limits provides reasonable assurance that, in the event of any radioactivity release within the containment, the loss of the containment atmosphere through leak paths will not be in excess of the limits specified for the site.

The applicant has provided a detailed discussion of the containment integrated leak rate (Type A) test procedure and acceptance criteria. All systems penetrating containment are either identified as being vented and drained to the containment atmosphere, so that the accident differential pressure will exist across the containment isolation valves, or are identified as remaining filled with liquid for the Type A test. Justification was provided for each system that was not vented and drained to the containment atmosphere for the Type A test. The applicant has also committed to locally leak test all containment isolation valves in systems that will not be vented and drained. We find this approach acceptable.

The applicant has listed all the containment penetrations and has itemized all the local leak testing that will be performed. Schematic drawings of each piping system penetrating containment have been submitted, showing test, vent and drain connections and indicating the direction in which the containment isolation valves will be locally leak tested.

All containment isolation valves that will be locally (Type C) leak tested, with the pressure applied in the direction opposite to that which occurs when the valve performs its safety function, have been identified. Justification has been submitted that performing Type C tests with the pressure applied in the reverse direction will result in equivalent or conservative leak rates.

The Davis-Besse 2 and 3 containment leak testing program identifies an exception to the requirements of Appendix J. Paragraph III.D.2 of Appendix J requires that personnel air locks be locally (Type B) leak tested after every opening. This testing procedure, however, is not practical for intervals when the personnel air locks are under frequent usage. Therefore, the applicant has proposed that the space between the double seals be pressurized to a reduced pressure within 72 hours of being opened. This testing, which is in addition to the six-month full pressurization tests required by Appendix J, would show that the integrity of the door seals is being maintained. We have reviewed the applicant's proposed exception to III.D.2 of Appendix J and have concluded that it is acceptable.

We have reviewed the applicant's containment leak testing program as presented in Sections 6.2 and 16.5 of the Preliminary Safety Analysis Report. We conclude that the program complies with the requirements of Appendix J and constitutes an acceptable basis for satisfying the applicable requirements of General Design Criteria 52, 53 and 54.

6.2.6 Containment Air Purification and Cleanup Systems

In addition to its heat removal function, the containment spray system also serves to reduce the fission product concentrations in the containment atmosphere following a postulated loss-of-coolant accident or a steam line break accident.

Sodium hydroxide will be added to the spray solution to enhance the elemental iodine absorption effectiveness of the solution. An eductor system will be used to inject 30 weight-percent sodium hydroxide into the suction flow of each of the two containment spray pumps. The eductor will be sized so that the amount of sodium hydroxide added yields a spray solution with a pH of about 9.3 during the injection phase, and a range of 8.5 to 11 during the recirculation phase. Chemical addition will be terminated at some time in the recirculation phase, when the chemical additive tank is emptied.

The spray solution will be dispersed in the upper region of the containment by nozzles located on four headers (two headers per redundant spray train) in the containment dome. We have conservatively estimated spray coverage to be 2.392 million cubic feet, which leaves unsprayed approximately 15 percent of the containment free volume. We find the proposed arrangement to be sufficient, in conjunction with the dual containment design, to limit offsite radiation doses to within the guideline values of 10 CFR Part 100.

The applicant has provided a description of tests to be performed on the system. Such tests are adequate to assure the operability and function of the components and of the system itself.

We have evaluated the iodine removal effectiveness of the spray in the sprayed region of the containment, and find the system effective for removal of elemental and particulate iodine. For the sprayed volume, we calculated first order elemental iodine removal coefficients above 10 per hour and a particulate iodine removal coefficient of 0.37 per hour. We have limited the elemental iodine removal coefficients used in the accident offsite dose calculations to a maximum of 10 per hour in order to maintain compatibility with the assumptions of Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors." With the proposed additive system, we estimated a minimum sump pH of 8.5, which is sufficient to maintain an equilibrium decontamination factor of 100 for elemental iodine. We have conservatively assumed that the elemental iodine removal effectiveness of the

spray system will cease when the partition of elemental iodine corresponding to this decontamination factor has been achieved in the containment atmosphere.

6.3 Emergency Core Cooling System

The applicant references Criterion 35 of the General Design Criteria as the principal design basis of the emergency core cooling system for providing protection over the entire spectrum of break sizes. Very small breaks that do not actuate the engineered safety features mode of operation will be accommodated by the makeup and purification system as required by Criterion 33 of the General Design Criteria.

Postulated loss-of-coolant accident analyses are used to demonstrate that the functional performance of the emergency core cooling system results in plant conditions in conformance with the acceptable criteria of 10 CFR Part 50.46. The analysis was performed with a model found acceptable relative to requirements of 10 CFR Part 50, Appendix K.

6.3.1 Discussion of Proposed System

An emergency core cooling system will be provided to provide cooling of the core in the event of a rupture of the reactor coolant system. Accidents considered are those up to and including a full guillotine rupture of the largest reactor coolant system pipe. The emergency core cooling system will be provided with sufficient redundancy, diversity, and capacity to assure emergency cooling even with a single active failure. The applicant has stated that, for breaks too small to actuate the engineered safety features, the normal makeup system will provide sufficient flow to satisfy the requirements of General Design Criterion 33.

The emergency core cooling system will consist of core flooding tanks, and high-pressure and low-pressure injection pumping systems. Injection water will be supplied from the borated water storage tank. Combinations of these systems will provide core cooling protection against the entire range of postulated reactor coolant system piping breaks, including those at injection inlet connections to the reactor coolant system.

The core flooding tanks system will provide emergency cooling to the core for intermediate-to-low reactor coolant system pressures. The two core flooding tanks will be pressurized with nitrogen to 600 pounds per square inch gauge. Each tank will have a normal borated water volume of 1040 cubic feet, and will be designed in accordance with the ASME Code, Section III, Class 2. The core flooding system will actuate when the reactor coolant system pressure drops below the nitrogen pressure. Appropriate valving, procedures, and interlocks will be provided to prevent over-pressurization of the core flooding system by the reactor coolant system under normal operation; to assure delivery of flooding water when required; and to

prevent inadvertent flooding of the reactor during normal shutdown depressurization. The core flooding tank will be equipped with a relief valve that actuates at 700 pounds per square inch gauge to avoid tank overpressurization.

The high pressure injection system will be actuated upon receipt of an engineered safety feature actuation signal. The signal to start high pressure injection will result from either high containment pressure or low reactor coolant pressure. The high pressure injection system includes 2 high pressure pumps, each rated at 500 gallons per minute. These pumps inject borated water into the primary system cold legs. High pressure injection will be designed so that the capacity of one pump will be sufficient to provide core cooling for breaks that are not large enough to actuate low pressure injection.

System valves will be designed to open fully within 11.5 seconds of receipt of the actuation signal. During the injection mode of operation, the pumps take suction from the borated water storage tank. The pumps may be aligned to take suction from the low pressure injection pump discharge, during the recirculation mode, to accommodate small breaks, up to 0.1 square feet, that do not fully depressurize the primary system. The high pressure system will be isolated from the primary system by two check valves inside containment and a normally closed motor-operated valve outside containment, to prevent overpressurization of the injection lines.

The high pressure injection system will include miniflow lines that permit recirculation of borated water back to the borated water storage tank. These lines will be used for periodic testing of the high pressure pumps, and also will provide a flow path in the event of actuation of high pressure injection when the primary coolant pressure is above the shutoff head of the pumps. There will be an individual miniflow line from each pump discharge, and each line will contain a normally open motor-operated isolation valve. Downstream of the isolation valves, the lines connect to a single line that returns the flow to the storage tank. The single line returning to the tank contains an air-operated isolation valve that fails in the open position.

We were concerned about a spurious closure of the air-operated valve during the injection phase of operation. Such a single failure would potentially result in loss of both high pressure pumps in the event of a small loss-of-coolant accident during which the primary pressure stayed above the pump shutoff head. The applicant has committed to disconnect the air supply to the valve. This procedure is acceptable to us because it protects against a single failure causing inadvertent isolation of the line.

We were also concerned about the need to isolate the miniflow lines during the recirculation phase of operation. Because only a single isolation valve will be in each miniflow line, a failure of one of these valves to close could result in contaminated water being discharged into the storage tank during the recirculation

phase. The applicant has committed to establish a procedure to shut down the pump connected to the nonisolated miniflow line and, thereby, prevent discharge of radioactive water to the tank. We find this proposed procedure acceptable provided the applicant demonstrates in the Final Safety Analysis Report that the single isolation valve in each line is leak tight and that instrumentation is available to detect and alarm any incomplete isolation of both miniflow lines.

The low pressure injection system will consist of two redundant low pressure injection flow trains, each having one decay heat pump, rated at 3000 gallons per minute, and one heat exchanger. The capacity of either train alone will provide acceptable core cooling. A passive cavitating venturi network will connect the two trains to assure adequate flow to the primary system even with a single failure.

During the injection mode, the low pressure pumps will take suction from the borated water storage tank. The system configuration for the long-term cooling phase of operation, including methods to detect, identify, and isolate passive failures in the emergency core cooling system outside of containment, have been addressed in the Preliminary Safety Analysis Report, and are discussed in Section 6.3.3 of this report.

The borated water storage tank will be the water supply for the injection phase of emergency core cooling. It will have a total volume of 550,000 gallons, with a minimum inventory of 422,500 gallons. When the tank level drops to 42,500 gallons, pump suction will be automatically switched from the tank to the containment emergency sump, and the tank discharge valve will close. During the 60 to 90 minutes required to draw the tank level down to the switchover point, the operator will have sufficient time to assess whether the primary system pressure has stayed high enough to require using the low pressure pumps to provide adequate suction pressure to the high pressure pumps during recirculation.

The applicant's analyses indicate that the sump design will provide adequate net positive suction head to permit low pressure pump operation during recirculation and will provide adequate submergence to avoid vortexing.

6.3.2 Tests and Inspections

The applicant has described programs for preoperational testing, periodic testing, and in-service testing and inspection of all emergency core cooling system components to demonstrate operability and component design adequacy.

The applicant has committed to complying with Regulatory Guide 1.68, "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors," in preoperational testing. The testing will include hot and cold flow tests of the high pressure system, cold flow and recirculation tests of the low pressure system, core flooding tests, and isolation valve tests.

In Preliminary Safety Analysis Report Revision 16, the applicant has also committed to providing a sump design which would permit testing that would fulfill the recommendations of Regulatory Guide 1.79, "Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors." We find the sump design commitment acceptable, and will review the sump-test program at the operating license stage.

A program of periodic testing of emergency core cooling system components has been described to include testing of pumps, valves, and high pressure-to-low pressure crossover valves. Low pressure isolation check valves will be monitored for leakage and will be tested at each refueling.

Periodic tests and in-service inspections will be in accordance with the ASME Code, Section XI.

6.3.3 Emergency Core Cooling Loss-of-Coolant Accident Analyses

The applicant has performed analyses for Davis-Besse Units 2 and 3 to determine the consequences of a postulated loss-of-coolant accident and to assure adequacy of the emergency core cooling system. These analyses were performed with approved evaluation models that conform to the requirements of Appendix K of 10 CFR Part 50.

Analyses of small breaks for Davis-Besse Unit 1 were cited as applicable to Units 2 and 3. These analyses result in a calculated peak cladding temperature of 1673 degrees Fahrenheit, and indicate that small breaks are not limiting. Because of the similarity of plant parameters and reactor power, and because the core is predicted to not be uncovered, we consider the Davis-Besse Unit 1 small break analysis to be acceptable for demonstrating that small breaks will not be limiting for Units 2 and 3.

Babcock & Wilcox generic studies have shown that a reactor coolant pump discharge break is the limiting location for large breaks in Babcock & Wilcox 177 fuel assembly plants that have raised loop arrangements.

The analysis assumed a steady-state operating power level of 102 percent of 2772 megawatts thermal, with a peak linear power of 15.5 kilowatts per foot and an axial peaking factor of 1.7.

A spectrum of pump discharge breaks analyzed for Davis-Besse Units 2 and 3, using an approved model, identified the worst break to be an 8.55 square-foot double-ended guillotine with a discharge coefficient of 1.0. Results calculated for this break indicate a 2122 degrees Fahrenheit peak cladding temperature and 5.34 percent local metal-water oxidation. A whole-core oxidation of 0.67 percent was obtained by noting similarity in fuel geometry and transient history between Davis-Besse Units 2 and 3 fuel and the fuel analyzed for the 205 fuel assembly type plant. We find this

extrapolation of the core-wide oxidation acceptable at the construction permit stage, but we will reevaluate its applicability during the operating license stage. The reported results meet the limits specified in 10 CFR 50.46 for the peak cladding temperature, local oxidation, and core-wide oxidation of 2200 degrees Fahrenheit, 17 percent and 1.0 percent, respectively.

The applicant has shown that the operator will be able to realign the emergency core cooling system to preclude excessive buildup of boron concentration in the core. Although the realignment will be performed about 24 hours after the accident, it is not necessary for boron flushing for about seven days.

The applicant has described equipment and procedures to detect, alarm, identify, and isolate passive failures in the emergency core cooling system and supporting subsystems outside of containment during long-term cooling. The applicant has indicated that the operator will be alerted by an alarm on the sump pumps in the emergency core cooling equipment rooms when approximately 700 gallons of water have accumulated. This accumulation would take about 23 minutes for a 30-gallon per minute leak. The applicant has calculated that the emergency core cooling system capability would not be compromised for at least three days by such a small leak, even if no corrective action were taken. The analysis, equipment, and procedures meet our requirements and are acceptable.

The applicant has investigated the possibility of the flooding of safety-related equipment inside containment following a loss-of-coolant accident or steam line break. He has considered all safety-related equipment, including motors, operators, controls, indicators, and alarms, and initially reported that only the decay heat pump suction isolation valves (DH11A, DH12A, DH11B, and 112B) might become submerged. The applicant has committed to modifying the design to raise the operators for these valves so that they will be above the flood level to assure valve operability. We find this to be acceptable.

6.3.4 Containment Pressure Evaluation for Emergency Core Cooling

Appendix K to 10 CFR Part 50 requires that the effect of operation of all the installed containment pressure-reducing systems and processes be included in the emergency core cooling system evaluation. For this evaluation, it is conservative to minimize the containment pressure, because low pressure will increase the resistance to steam flow in the reactor coolant loops and reduce the reflood rate in the core. Following a loss-of-coolant accident, the pressure in the containment building will be increased by the addition of steam and water from the primary system into the containment atmosphere. After initial blowdown, heat flow from the core, primary metal structures, and steam generators to the emergency core cooling water will produce additional steam. This steam, together with any emergency core cooling water spilled from the primary system, will flow through the postulated

break into the containment. This energy will be released to the containment during the blowdown and during the reflood and post-reflood phases.

Energy removal occurs within the containment by several means. Steam condensation on the containment walls and internal structures becomes effective early in the blowdown transient. Subsequently, the operation of the containment heat removal systems, such as containment sprays and fan coolers, will remove energy from the containment atmosphere. When the energy removal rate exceeds the rate of energy addition from the primary system, the containment pressure will decrease from its maximum value.

The emergency core cooling containment pressure calculations for Davis-Besse Units 2 and 3 were done generically by Babcock & Wilcox, for reactors of this type, as described in BAW-10105, "ECCS Evaluation of B&W's 177-FA Raised Loop NSS." We concluded that the Babcock & Wilcox containment pressure model was acceptable for the emergency core cooling system evaluation. We required, however, that the plant-dependent input parameters used in the analysis be submitted for our review of each plant.

This information was submitted for our review and we have concluded that the plant-dependent information used for the analysis for Davis-Besse Units 2 and 3 is conservative and, therefore, the calculated containment pressures are in accordance with Appendix K to 10 CFR Part 50.

6.3.5 Conclusions

We have reviewed the descriptions, design criteria, and piping and instrumentation diagrams of the emergency core cooling system. Suitable redundancy has been provided for the pumps, piping arrangement, and power sources so that no single active failure during the injection phase will compromise the anticipated minimum system performance as required by General Design Criterion 35. Isolation provisions between the reactor coolant system and the emergency core cooling system are acceptable and meet containment isolation requirements and General Design Criterion 35. Startup and periodic tests are proposed that meet the requirements of General Design Criterion 37. The system layout will be adequate to allow visual inspection of the components as required by General Design Criterion 36. The loss-of-coolant accident analyses, which were performed with an approved evaluation model in conformance with Appendix K to 10 CFR Part 50, show that the calculated fuel performance parameters are within the limits of 10 CFR 50.46(b). The design includes provisions for long-term cooling and maintenance of coolable geometry, as required by 10 CFR 50.46. We conclude that the system will meet the applicable requirements and is, therefore, acceptable.

6.4 Habitability Systems

In this section we report the results of our evaluation of the emergency protective provisions for the control room, as related to the accidental release of radioactivity or of toxic gases. Relevant portions of the control room ventilation system are described here, but the total system is described and evaluated more fully in Section 9.4 of this report.

6.4.1 Radiation Protection Provisions

The applicant proposes to meet General Design Criterion 19 of Appendix A to 10 CFR Part 50, by use of concrete shielding and a 3300 cubic feet per minute charcoal filter to control radiation levels within the control room.

In the event of high radioactivity, after a loss-of-coolant accident, the normal control room air-conditioning system will be shut down automatically and the emergency ventilation system will be started manually. All outside air dampers will be closed to minimize the dose to operating personnel. For the first four days following the accident, the emergency system will be operated in a fully recirculating mode, in which 3300 cubic feet per minute of air is processed through the charcoal filter. On the fourth day, the system will be manually switched to admit 200 cubic feet per minute of filtered outside air to the control room, with 3100 cubic feet per minute being recirculated through the filter.

We have performed operator dose calculations assuming a design basis loss-of-coolant accident. The resultant doses are within the guidelines of General Design Criterion 19. We, therefore, conclude that the control room radiation protection is acceptable.

6.4.2 Toxic Gas Protection Provisions

Control room habitability, following a postulated toxic gas release, is required to assure that operators can continue to operate the plant. Chlorine has been identified as the only material stored on site that, if released, would pose a potential operator hazard. Quick-acting chlorine detectors and self-contained breathing apparatus will be provided to protect the operator against a chlorine release. We have reviewed these provisions against the guidelines of Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," and have found them to be adequate. We conclude that the proposed toxic gas protection is acceptable.

6.5 Engineered Safety Features Materials

The mechanical properties of materials selected for the engineered safety features satisfy Appendix I of Section III of the ASME Code, or Parts A, B and C of Section II

of the Code, and our position that the yield strength of cold worked stainless steels shall not exceed 90,000 pounds per square inch.

The controls on the pH of the reactor containment sprays and the emergency core cooling water are adequate to assure freedom from stress corrosion cracking of the austenitic stainless steel components and welds of the containment spray and emergency core cooling systems throughout the duration of the postulated accident to completion of cleanup. The controls on the use and fabrication of the austenitic stainless steel of these systems satisfy the requirements of the NRC Interim Position on Regulatory Guide 1.31, "Control of Stainless Steel Welding," and Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel." Fabrication and heat treatment practices, performed in accordance with these requirements, provide added assurance that stress corrosion cracking will not occur during the postulated accident time interval. The controls on the pH of the sprays and cooling water, in conjunction with controls on selection of containment materials, are in accordance with Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," and provide assurance that the sprays and cooling water will not give rise to excessive hydrogen gas evolution by corrosion of containment metal or cause serious deterioration of the containment. The controls placed on concentrations of leachable impurities in nonmetallic thermal insulation used on austenitic stainless steel components of the engineered safety features are in accordance with Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel."

Conformance with the codes and regulatory guides mentioned above and with our positions on the allowable maximum yield strength of cold worked austenitic stainless steel, and the minimum level of pH of containment sprays and emergency core cooling water, constitute an acceptable basis for meeting applicable requirements of General Design Criteria 35, 38, and 41.

7.0 INSTRUMENTATION AND CONTROL

7.1 General

The instrumentation and control systems have been reviewed utilizing, as bases for evaluating their adequacy, the Commission's General Design Criteria, the Institute of Electrical and Electronics Engineers (IEEE) Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," applicable Regulatory Guides, and Branch Technical Positions.

7.1.1 Identification of Safety Criteria

We have reviewed the information provided in Preliminary Safety Analysis Report Section 7.1 for the instrumentation and controls associated with the proposed design. We have concluded that the list of criteria, regulatory guides, and standards that were utilized in the design of the instrumentation and control systems is acceptable.

In addition, we have reviewed the information provided in Section 7.0, which identifies the differences between Davis-Besse Units 2 and 3 and Davis-Besse Unit 1. The applicant has used Unit 1 as a reference design. Based on this information, we have concentrated our review efforts on those areas of design for Units 2 and 3 in which there were changes from the design of Unit 1.

7.1.2 Independence of Redundant Safety-Related Systems

For Davis-Besse Units 2 and 3, the applicant has identified IEEE 384-1974 and Regulatory Guide 1.75, "Physical Independence of Electric systems," as the design bases for the separation of electric systems. We find this acceptable. However, in order to satisfy certain portions of the design bases, the applicant has proposed utilizing isolation devices. The details of these isolation devices have not been provided in the Preliminary Safety Analysis Report. Therefore, we required the applicant to provide, at least one year prior to submittal of the Final Safety Analysis Report, the requested information concerning the qualification program for Class IE instrumentation system isolation devices. This will include the identification of (1) how each of the above isolation devices has been, or will be, qualified; (2) the acceptance criteria; (3) test procedures; and (4) test results, if available, or a schedule for submittal of test results. In a letter dated March 7, 1978 the applicant has committed to this requirement, and we conclude that this is acceptable.

7.2 Reactor Protection System

7.2.1 Description

Subsequent to the original submittal of the Preliminary Safety Analysis Report for Davis-Besse Units 2 and 3, the applicant decided to incorporate the new Babcock & Wilcox reactor protection system designated RPS-II. This design has been proposed for the BSAR-205 Standard Plant, Washington Nuclear Project, Units 1 and 4; Pebble Springs Nuclear Plant, Units 1 and 2; Bellefonte, Units 1 and 2; and Greene County Nuclear Power Plant. RPS-II is a "hybrid" system configuration combining both analog function modules and programmable digital microcomputers in its design. Analog function modules and solid-state optically isolated distable units are used for eight of the trip functions. This portion of RPS-II is similar to the previous Babcock & Wilcox protection system design (RPS-I) except that solid state, optically isolated devices have replaced relays in the trip logic. The unique feature of RPS-II is the utilization of a programmable calculating module (digital micro-computer system) to calculate the reactor offset, low departure from nucleate boiling ratio, reactor coolant pump status, and power versus delta-T (startup) trips.

Babcock & Wilcox submitted topical report BAW-10085, "Reactor Protection System," in June 1974 on RPS-II. This report was superseded by revisions in March 1975, January 1977, and April 1977. Topical Report BAW-10085 and its revisions encompass only that portion of the reactor protection system which possesses sensor input signals, determines the need for protective action at the channel level, and initiates the protective action, when required, at the system level.

The applicant has presented information in Section 7.2 of the Preliminary Safety Analysis Report regarding the sensors, actuating devices and other remote inter-connecting devices that are not included in the scope of BAW-10085. We have reviewed this information for conformance to the applicable regulations, guides, technical positions and industry standards and conclude that they are acceptable.

In a letter to Babcock & Wilcox dated January 8, 1976 concerning the review of topical report BAW-10085, we stated that "the hybrid design of RPS-II represents an acceptable concept for application in a reactor protection system." Our review of the topical report is being conducted as a generic item and is incomplete at this time.

However, it is our intent to complete our review and evaluation of BAW-10085 prior to the receipt of the Final Safety Analysis Report for Davis-Besse Units 2 and 3. The applicant has committed to conform to the generic resolution of the BAW-10085 review. We consider this commitment to be acceptable for the issuance of a construction permit.

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7.2.2 Protection Systems Response Time Testing

The applicant has stated his intention to perform response time testing for the reactor protection system and engineered safety features actuation system. With regard to sensor response time, the applicant has agreed to participate in industry research programs (such as those being pursued by Electric Power Research Institute) in order to determine acceptable methods for determining sensor response times. In addition, the applicant has committed to utilizing the test procedures to be developed for complying with the surveillance requirements of the technical specifications to be developed at the operating license stage of review. We conclude that this is acceptable for a construction permit.

7.3 Engineered Safety Features Actuation System

7.3.1 General

The engineered safety features actuation system is the portion of the plant protection system that initiates action of engineered safety features systems to mitigate the consequences of design basis events. The basic design of the engineered safety features actuation system for Units 2 and 3 is the same as that for Unit 1. Since the Unit 1 design was reviewed and approved for construction, we have issued Regulatory Guide 1.75, "Physical Independence of Electric Systems," and Regulatory Guide 1.89, "Qualification of Class IE Equipment for Nuclear Power Plants." Conformance to the regulatory positions of these guides required the applicant to modify the Unit 1 design for Units 2 and 3. We have reviewed the modified system and, for the reasons stated below, have concluded that it is acceptable.

The initiating circuits of the actuation system are the sensors that monitor (1) containment radiation, (2) containment pressure, (3) reactor coolant pressure, and (4) borated water storage tank level.

Four separate, independent, redundant sensing channels will be provided for each of the above variables. Each sensing channel will include analog circuits that are composed of sensors and bistable trip modules with digital isolators.

The digital signals will feed two-out-of-four coincidence logic matrices in the system logic cabinets. Should two of the four channel bistables, monitoring a unit variable, cease to send output signals (i.e., trip), the two-out-of-four logic matrices would be enabled, and the corresponding normally energized terminating relays on all logic channels would trip. Tripping of these relays will initiate the actuation signal to the engineered safety features equipment to be actuated.

There will be two separate, independent and redundant actuation channels corresponding to the two divisions of engineered safety features equipment. These actuation channels will be derived from the four logic channels. Logic channels one and three must both be de-energized (tripped) to activate actuation channel

one, and logic channels two and four must be de-energized to activate actuation channel number two.

The actuation system will be a solid state, de-energize-to-trip system and, if power is lost to a channel, that channel will trip, thereby reducing the coincidence matrices from two-out-of-four to one-out-of-three.

The capability for "sensing channel" bypass will be provided. Bypass changes the coincidence matrices from two-out-of-four to two-out-of-three. These bypasses are for test, calibration, or maintenance of the analog circuits of the actuation system only, and provisions will be included to allow the operators to bypass only one channel of a variable at a time.

7.3.2 Auxiliary Feedwater System

The auxiliary feedwater system will consist of two full capacity, independent systems. One system will include a motor driven pump that will be supplied with motive power from an essential Class IE alternating current power source. The other system will include a steam turbine driven pump that will be able to use steam from either of the two steam generators.

During our review, we required the applicant to demonstrate that there will be sufficient power supply diversity and independence for the system, including the instrumentation and controls to allow the turbine driven system to function during a total loss of alternating current power.

The diversity of power supply for the instrumentation and control portion will be accomplished in the following manner:

- (1) The turbine control system will be powered from a Class IE direct current power source.
- (2) The valves associated with the operation of the turbine driven portion (both steam and water sides) will be either powered from the direct current source or locked open.
- (3) The valves associated with the operation of the motor driven portion will be either powered from the alternating current source or locked open.
- (4) The closed position of any of these motor operated valves will be monitored in the control room.

We have concluded that the auxiliary feedwater system controls will be designed in accordance with our requirements and, therefore, are acceptable.

7.3.3 Periodic Testing of the Engineered Safety Features

The applicant's design criteria provide for testability of individual channels, logic and final actuating devices, and satisfy the requirements of IEEE Standard 279-1971 and Regulatory Guide 1.22, "Periodic Testing of Protection System Actuation Functions." The applicant has stated that all isolation valves, except the main steam and feedwater line isolation valves, will be tested and periodically full-stroke exercised during plant operation. The steam and feedwater valves will be tested by partial-stroking during plant operation. With this commitment, we find the design criteria, and the program for periodic testing of the engineered safety features actuation system, to be acceptable.

7.3.4 Bypassed and Inoperable Status Indication

The applicant has stated that the systems important to safety will be designed such that, when a component within the system is bypassed or rendered inoperative, a status indication in the main control room will be initiated. The status indication will be designed in accordance with the recommendation of Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Safety Systems." We conclude that the design criteria are acceptable.

7.4 Systems Required for Safe Shutdown

7.4.1 Control Rod Drive Control System

The control rods will be inserted into the core upon receipt of the reactor protection system trip signals, which act to de-energize the control rod drive mechanisms. The control rod drive trip logic will be such that, when power is removed from the mechanism, the roller nuts disengage from the leadscrew and a free-fall, gravity trip of the control rod occurs. The trip devices will consist of four alternating current circuit breakers, two in each power supply circuit. Trips will be initiated when power is interrupted to the under-voltage coils of the main and secondary supply breakers to the control rod drive power supplies. Since parallel power feeds will be provided, both the main and secondary supplies must be interrupted in order to effect trip action. The control rod drive control system will be the same as that used in recently licensed nuclear plants, including Davis-Besse Unit 1. We, therefore, conclude there is reasonable assurance that the design, when implemented in accordance with Section 7.1 of this report, will satisfy our requirements, and is acceptable.

7.4.2 Steam and Feedwater Line Rupture Control System

The steam and feedwater line rupture control system is the portion of the plant protection system that will initiate the action of the engineered safety features that are required to mitigate the consequences of a main steam line or main

feedwater line rupture, or to supply emergency feedwater in the event of loss of the normal feedwater supply or the loss of all offsite power.

The applicant has stated that the control system will be essentially identical to that of Davis-Besse Unit 1 except for design modifications to Units 2 and 3 that are necessary to meet current requirements.

The Davis-Besse Unit 1 steam and feedwater line rupture control system is composed of two redundant and independent subsystems. Each subsystem consists of an alternating current-powered logic channel and a direct current powered logic channel. The alternating and direct current logic trains are identical, and are maintained separate and independent within the channel cabinet. The loss of power to the logic channel will trip the affected channel.

Each logic channel receives inputs from main steam line pressure, main feedwater/steam generator differential pressure, steam generator level, and reactor coolant pump status.

Operation of each subsystem requires the actuation of both the alternating current and the direct current logic channel in the subsystem to initiate a safety action. We have concluded that the applicant has identified the required design bases and criteria for the steam and feedwater line rupture control system and has presented an acceptable design approach. Therefore, we conclude that the system is acceptable for the construction permit stage of review.

7.4.3 Auxiliary Shutdown Panel

The applicant has stated that, in accordance with General Design Criterion 19, if temporary evacuation of the control room is required due to some abnormal unit condition, the plant will be able to be maintained in a safe hot shutdown condition through the use of an auxiliary shutdown panel located outside the control room. In addition, the ability will exist to bring the unit to cold shutdown from outside the control room, but it would require additional manpower to perform local control action.

We have concluded that the applicant has given necessary design consideration for safe shutdown of the reactor from outside control room in the event of evacuation of the control room.

7.5 Safety-Related Display Instrumentation

The applicant has identified the instrumentation required for maintaining the plant in a safe shutdown condition, and for performing the required safety functions after postulated incidents, such as a loss-of-coolant accident. In addition, the applicant has identified the design requirements of this instrumentation, including

redundant indication in the control room; i.e., at least one channel recorded; energized from onsite power supplies; designed in accordance with the applicable portions of IEEE Standard 279-1971; and seismically qualified. We conclude that the design, as presented in the Preliminary Safety Analysis Report, is in conformance with Branch Technical Position EICSB-23, "Qualification of Safety-Related Display Instrumentation for Post-Accident Condition Monitoring and Safe Shutdown," and is, therefore, acceptable.

A staff task force is developing a revised position on minimum requirements for safety-related display instrumentation. It is possible that a revised position may impose additional requirements for safety-related display instrumentation at some time in the future. In that event we will require the Davis-Besse Units 2 and 3 design to be modified accordingly.

7.6 All Other Systems Required for Safety

The applicant has identified the decay heat removal valve control system, the core flooding tank isolation valve control system, the containment spray pump anti-cavitation control system, and the emergency diesel generator service water control system as additional equipment required for the safe functioning of the plant.

We have concluded that the design of these systems meet our requirements as stated in Section 7.1 of this report and, therefore, are acceptable. Significant areas of our review of these systems are discussed below.

7.6.1 Decay Heat Removal Valve Control System

The decay heat removal valve control system will include controls on each of the high pressure, motor-operated valves in the suction lines from the reactor coolant system. These independent controls will be designed to close the valves automatically, or to prevent the opening of the valves, when the reactor coolant system pressure is above 280 pounds per square inch gauge. This prevents overpressurizing the decay heat removal system in the event the valves are inadvertently left open during heatup or if an operator prematurely tries to open the valves during cooldown.

The high-pressure motor-operated valves will be powered and controlled by a four-channel arrangement. The use of four-channel control and power assures that no single failure will prevent the decay heat removal system from performing its intended function or from performing high-pressure to low-pressure isolation.

Diversity will be provided by the use of pressure switches for two of the four valves and pressure transmitters/trip units for the other two valves.

7.6.2 Core Flooding Tank Isolation Valve Control System

This control system will be provided to open the core flooding tank isolation valves, when required, and prevent their closing when the reactor coolant system pressure exceeds a preset level.

We have reviewed the design of the core flooding tank isolation valve control system. The provisions of the design are in accordance with Branch Technical Position EICSB-4, "Requirements on Motor-Operated Valves in the ECCS Accumulator Lines." The isolation valves will receive an engineered safety feature actuation system signal to open. Visual indication of the position of the isolation valves (open or closed) will be provided in the control room. Switches on the valves will be used to actuate these indicators. The facility technical specifications will require that power to the valve operators be removed during operation.

7.6.3 Containment Spray Pump Anti-Cavitation Control System

When the two-out-of-four level sensors on the borated water storage tank sense low tank level, the storage tank discharge valves will close and the containment emergency sump valves will open. This instrumentation and control will be part of the engineered safety features actuation system.

Controls will be provided on the sump valves to automatically throttle the containment spray pump discharge isolation valves when the pumps take suction from the containment emergency sump. This action is intended to prevent cavitation of the pumps caused by the lower net positive suction head available from the containment emergency sump. We consider this to be acceptable.

7.6.4 Emergency Diesel Generator Service Water Control System

This control system will be provided to assure that service water will flow through each diesel generator cooler whenever the diesels are running. Service water flow is required to prevent the diesels from overheating during operation.

The service water outlet valves to the coolers will be opened automatically whenever the emergency diesel generator starting circuits are energized and the emergency diesel generator speed reaches 40 revolutions per minute. The valves will be actuated open and closed by the emergency diesel generator start and stop controls.

7.7 Control Systems Not Required for Safety

The following systems have been identified by the applicant as control systems not required for safety:

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- (1) Non-nuclear instrumentation control system.
- (2) Integrated control system.
- (3) Control rod drive control system, excluding the trip portion.
- (4) Turbine generator electro-hydraulic control system.

For each of these systems, the applicant has performed analyses for the transient and accident events described in Chapter 15 of the Preliminary Safety Analysis Report, assuming the normal operation of the control system, the failure of the control system, and the malfunction of the control system. For each case the applicant has demonstrated that the worst case assumptions regarding the functioning of the control system do not produce results worse than the results of the accident analyses. We have interpreted this to mean that, for each accident, the safety systems alone are capable of mitigating the consequences, assuming the worst case malfunctioning of the control system for the particular incident analyzed. However, we will pursue the details of these systems, with regard to plant safety, during the operating license review stage when design details will be available.

Based on our review of the above systems and of the operational transients described in Section 15.0 of this report, we conclude that failures or malfunctions of these control systems would not be expected to degrade the capabilities of the safety systems to any significant degree, nor to lead to plant conditions more severe than those for which the safety systems are designed. Therefore, we find these systems to be acceptable for the construction permit stage of review.

7.8 Other Instrumentation Systems

The following systems have been identified as other instrumentation systems:

- (1) Nuclear Instrumentation.
- (2) Incore Monitoring System.
- (3) Unit Computer System.
- (4) Unit Annunciator.

In each case, the applicant has stated that the information provided by these systems will not be required for safety, and that unit safety will not be compromised or prevented by their loss.

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There is one exception to the above statement, and this pertains to the portion of the nuclear instrumentation that will monitor the power range. The applicant considers this portion to be an integral part of the reactor protection system, and it will be designed and qualified to meet all the reactor protection system qualification requirements.

Based on the above information, we conclude that these systems will not compromise safety and are, therefore, acceptable.

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8.0 ELECTRIC POWER SYSTEMS

8.1 General Discussion

The Commission's Criteria 17 and 18 of the General Design Criteria; Regulatory Guide 1.6, "Independence Between Redundant Standby (Onsite) Power Sources and Between their Distribution Systems"; Regulatory Guide 1.9, "Selection of Diesel Generator Set Capacity for Standby Power Supplies"; Regulatory Guide 1.32, "Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants"; Regulatory Guide 1.75, "Physical Independence of Electric Systems"; and the Institute of Electrical and Electronics Engineers (IEEE) Standards, including IEEE Standard 308-1971; were utilized as the primary bases for evaluating the adequacy of the electric power systems for Davis-Besse Units 2 and 3.

8.2 Offsite Power Systems

8.2.1 General Description

The 345 kilovolt switchyard for Davis-Besse Units 2 and 3 will be an expansion of the Unit 1 switchyard. The switchyard will be expanded from the present ring bus arrangement to a full breaker-and-a-half configuration. There will be three sources of offsite power to the switchyard. These sources will be physically independent from each other, and will be designed and located so as to minimize the likelihood of their simultaneous failure under operating or postulated accident and environmental conditions.

In addition, each 345 kilovolt breaker will be provided with redundant tripping coils actuated by redundant sets of protection relays that are powered from redundant batteries.

During normal operation, individual auxiliary power transformers, connected to each unit generator isolated phase bus, will provide the normal source of electrical power for their respective unit auxiliaries.

Four startup transformers, each of the same approximate capacity as the auxiliary power transformers, will be supplied from the 345 kilovolt switchyard. These transformers will provide power for startup, shutdown, and post-shutdown requirements, and will serve as a complete preferred power source in the event of failure of the auxiliary transformer supplies. Each unit will be supplied by independent and separately routed circuits from two startup transformers.

Normally, each startup transformer will be the preferred power source for only one 13.8 kilovolt bus. However, if either transformer is out of service, the remaining

transformer will be available to back up, automatically, both 13.8 kilovolt buses, if the normal source (auxiliary transformer) should fail.

The 13.8 kilovolt bus feeds the plant's 4.16 kilovolt bus system, which in turn feeds the Class 1E 4.16 kilovolt buses. These buses are discussed in Section 8.3 of this report.

Because two of the startup transformers will be shared among the three units (transformer 2 will be shared by Units 1 and 2, and transformer 3 will be shared by Units 2 and 3), we required the applicant to demonstrate conformance to General Design Criterion 5, "Sharing of Structures, Systems, and Components." The applicant has determined that the sharing of a startup transformer between units will not impair the ability of the transformers to perform their safety functions, including an orderly shutdown and cooldown of one unit following an accident to the other unit.

8.2.2 Analysis of Electric Power System

During our review, we asked the applicant to provide additional information and analyses with respect to the electrical power system. In response to these requests, the applicant provided additional information as follows:

- (1) The grid system is operated under guidelines established by the East Central Area Reliability Agreement (ECAR) of which the applicant's system is a member. The maximum and minimum acceptable spread values of voltage for the Davis-Besse switchyard are from 98.3 per cent to 102.2 percent (345 kilovolt base), and the operating guidelines are utilized to maintain the frequency spread of 60.0 Hertz to 59.0 Hertz.
- (2) To assure satisfactory operability of all electrical equipment during all modes of operation, the design documents will refer to the values of item (1) above or will utilize accepted industry-wide standards that define the nominal value and acceptable maximum and minimum values of voltage and frequency.
- (3) The applicant has described the operating procedures that are used to maintain grid configuration and operation within the specified limits.
- (4) The indications that will be provided to inform and alert the operator of the availability of the Class 1E power systems have been identified. In addition, the design provisions have been described for assuring continued operability of safety equipment should the offsite power system characteristics exceed the limits identified above.
- (5) Methods, including testing, have been described for verifying, before reactor operation, the adequacy of the plant design with regard to the plant electrical power system.

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We have concluded that, with the addition of the above information, the applicant has provided reasonable assurance that the electric power system design will conform to the Commission's requirements. The design is, therefore, acceptable for the construction permit stage of review.

In addition, the staff has developed further generic requirements in the areas of sustained degraded voltage conditions at the offsite power source and interaction of the offsite and onsite emergency power systems. The applicant has been informed of these requirements, and has committed to satisfying all of them. We will review the detailed implementation of the design in this area at the operating license stage.

8.2.3 Reactor Coolant Pump Coastdown/Coolant Flow Analysis

We requested that the applicant provide additional information on grid stability and frequency decay rate. Specifically, we require that the results of Babcock & Wilcox's loss-of-reactor coolant flow analysis show that, under the most adverse grid frequency conditions, the coast down capability of the reactor coolant pumps will be maintained. The applicant has committed to demonstrate that, for preventing fuel damage, the worst case underfrequency and maximum frequency decay rate for the Davis-Besse Nuclear Power Station grid will be within the corresponding values established in Babcock & Wilcox's analysis, to maintain the reactor coolant pump coast down without breaker trip. We will review the analysis at the operating license stage; the applicant's commitment is acceptable at the construction permit stage.

8.2.4 Conclusion

We have reviewed the offsite power system with respect to the requirements set forth in General Design Criteria 5, 17, and 18, and conclude that the system meets these requirements and is, therefore, acceptable.

8.3 Onsite Power Systems

8.3.1 Alternating Current Power Systems

The onsite power system will consist of the nonessential and essential systems. Essential electrical equipment will be required to assure a safe plant shutdown or to mitigate accident effects. This essential power system will consist of two completely redundant and independent load groups. Each load group will be comprised of distribution equipment and essential loads. There will be no automatic transfer schemes between load groups for any safety-related equipment. Two 4160 volt buses will be provided, along with 480 volt load centers and 480 volt motor control centers. The essential direct current requirements will be provided by four independent 125 volt direct current batteries. Essential alternating current power for each unit will be provided by two 4160 volt standby diesel generators and

four 120 volt vital buses. Each of the four 120 volt vital buses will be fed from a separate battery backed inverter, and each will be provided with its own voltage-regulating transformer as an alternate power source.

The applicant has stated that the criteria used for sizing the diesel generators are consistent with Regulatory Guide 1.9, "Selection of Diesel Generating Set Capacity for Standby Power Supplies," whereby each diesel generator will be sufficient to meet the engineered safety features demand caused by the loss-of-coolant accident and simultaneous loss of offsite alternating current power. The standby generators will be located in separate, individual rooms of a seismic Category 1 structure that is designed to withstand tornado missiles.

During standby generator operation, under essential operating conditions initiated by either a loss of essential bus voltage or an engineered safety features actuation signal, the generator protective trips will be limited to standby generator differential fault and engine overspeed. Except for manual synchronizing during routine testing, the essential bus will be isolated only upon loss of voltage on the bus (or a bus fault), and the standby generator output breaker will be closed only on loss of voltage without a bus fault.

With regard to diesel generator qualification, the applicant has committed that, in the event the diesel generators utilized for Davis-Besse 2 and 3 have not been previously qualified, they will conform to the qualification test program specified in Branch Technical Position EICSB 2, "Diesel Generator Reliability Qualification Testing."

The applicant has stated that provisions will be made for the manual transfer of a third 600 horsepower service water pump and a third 400 horsepower component cooling water pump between redundant load groups. These pumps will provide 100 percent backup for either of the two other service water pumps and the two other component cooling water pumps that are normally assigned to the two load groups. Each of these third pumps will be connected to two 4160 kilovolt mechanically interlocked, manually controlled, transfer switching breakers. The mechanical interlocks will consist of two shafts, one in each breaker, which will prevent one breaker from closing while the other is closed. This arrangement thus permits only one breaker to be closed at any time. Each breaker of a motor pair will be connected to a separate essential 4160 volt bus. Additional interlocks will be provided to prevent more than one component cooling water pump and one service water pump from being automatically connected to either emergency diesel generator at the same time.

The applicant has stated that his design will conform to IEEE Standard 317-1976 and to Regulatory Guide 1.63, "Electric Penetration Assemblies in Containment Structures for Light-Water-Cooled Nuclear Power Plants," which endorses this standard. With respect to the fault current versus time protection, the applicant

will use circuit overload protective devices that will meet the requirements of IEEE-Standard 279. The backup overload devices will be completely independent of the primary device.

The applicant has stated that Davis-Besse Units 2 and 3 will not use thermal overload protective devices for motor-operated valves in safety systems. We have previously concluded that this design approach is an acceptable way to design against the loss of function of safety related motor-operated valves, due to malfunction or failure of equipment protective devices during accident conditions. Therefore, we conclude that this is acceptable for Davis Besse Units 2 & 3.

The applicant has committed to conformance with the recommendations of Regulatory Guide 1.6, which provides guidance regarding independence between redundant power supplies and distribution systems.

We conclude that the proposed design of the onsite alternating current power system satisfies the applicable criteria stated in Section 8.1 of this report and, therefore, is acceptable.

8.3.2 Direct Current Power Systems

Four independent Class IE direct current power systems will be provided for each unit. Each Class IE 125 volt direct current supply will consist of one 125 volt battery supply, one battery charger, one direct current motor control center, one inverter (with manual transfer switch), and one 125 volt direct current distribution panel.

The four Class IE batteries, chargers, direct current switchgear, and voltage regulators, will be located in separate rooms of the seismic Category I auxiliary building. Each inverter will provide power to an independent 120 volt alternating current essential instrumentation distribution panel that will supply alternating current to the respective channel of the reactor protection and engineered safety features systems.

Each battery will have sufficient capacity to independently supply the required loads for two hours after a total loss of alternating current power. Each charger will be sized based on the largest combined demand for all the steady-state loads and the charging current required to restore the battery from design minimum charge state to the fully charged state within 12 hours, irrespective of the status of the plant when these demands occur.

Testing of the direct current power system will be in accordance with IEEE Standard 450-1975.

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In addition to the Class IE direct current system, Davis-Besse Units 2 and 3 will have a non-Class IE direct current system. The non-Class IE system will consist of two 125 volt batteries, two battery chargers, one 250/125 volt motor control center, four 125 volt distribution panels, two 125 volt direct current/120 volt alternating current inverters with integral static transfer switches, and two uninterruptible 120 volt alternating current instrumentation distribution panels. The system will be arranged to form two independent 125 volt direct current channels (A, B). When the two 125 volt systems are interconnected, a 250 volt system is established. The 250 volt system will supply the nonessential 250 volt direct current loads.

We conclude that the proposed design of the direct current power system satisfies the applicable criteria stated in Section 8.1 of this report and is, therefore, acceptable.

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9.0 AUXILIARY SYSTEMS

We have reviewed the design bases for the auxiliary systems, including their safety-related objectives, and the manner in which these objectives are achieved.

The auxiliary systems necessary for safe reactor operation and shutdown include: the service water system; portions of the component cooling water system; the ultimate heat sink; portions of the chemical and volume control system; portions of heating, ventilation and air conditioning systems for the control room, the emergency diesel generator rooms, and the service water and ultimate heat sink pump houses; the emergency diesel generators; the diesel generator fuel oil storage and transfer system; the diesel generator auxiliary systems; and the auxiliary feedwater system.

The systems necessary to assure safe handling of fuel and adequate cooling of the spent fuel include: the new and spent fuel storage facilities; the fuel pool cooling and cleanup system; fuel handling facilities; and the fuel handling building ventilation system.

We have also reviewed the equipment and floor drainage system, whose failure would not prevent safe shutdown of the reactor but could indirectly be a potential source of a radiological release to the environment.

Other auxiliary systems, whose failure would neither prevent safe shutdown nor result in potential radioactive release, include the pressurizer quench tank system, the nonessential portions of the service water system, the demineralized water system, the condensate storage facilities, portions of the compressed air system, the ventilation systems for nonsafety-related areas, and the communication and lighting systems.

The acceptability of these systems was based on our review that determined that: (1) where the system interfaces or connects to seismic Category I systems or components, seismic Category I isolation valves will be provided to physically separate the nonessential portions from the essential system or component, and (2) the failure of nonseismic systems or portions of the systems will not prevent the operation of safety-related systems or components located nearby. We find that the above-listed systems meet our criteria and, therefore, are acceptable.

Where systems or portions of systems are to be shared by Units 2 and 3 or among all three units, the applicant has stated that such sharing will not impair the ability of such systems to perform their safety functions. We have reviewed those systems and components to be shared, and find that the design meets the requirements of

General Design Criterion 5, and is acceptable. Shared systems are discussed in more detail in Sections 9.2.1 and 9.2.3 of this report.

The fire protection system is being reviewed to verify that failures in the system could not affect plant shutdown and that the system design meets required codes. However, as a result of investigations presently being conducted by the staff on the fire protection systems, additional requirements may be imposed to further improve the capability of the fire protection system to prevent unacceptable damage that may result from a fire.

9.1 Fuel Storage and Handling

9.1.1 New Fuel Storage

A new fuel storage pit will provide dry storage for approximately 80 new fuel assemblies. Normally, only 59 assemblies (1/3 of a core) will be stored at one time. The storage pit and racks will be designed to seismic Category I requirements. The racks will have a spacing that is sufficient to maintain a maximum effective multiplication factor of less than 0.90, even if the storage area were to be flooded with unborated water.

As a result of our review, we conclude that the design for the new fuel storage facilities meets the requirements of Criterion 62 of the General Design Criteria, regarding prevention of criticality, and the recommendations of Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," including seismic design and missile protection guidelines. Therefore, we find the design of the facilities for the storage of new fuel to be acceptable.

9.1.2 Spent Fuel Storage

Spent fuel will be stored under water in the spent fuel storage pool. The seismic Category I spent fuel storage racks will be designed to prevent fuel assemblies from being placed in other than their prescribed locations. The spent fuel storage racks and the spent fuel pool will be designed to accommodate one and one-third cores (236 fuel assemblies) plus 24 spares, including storage for failed fuel containers.

The fuel pool will be of reinforced concrete construction with a stainless steel liner, and will be designed to seismic Category I requirements. The spent fuel storage racks will be designed to withstand the maximum uplift forces of the spent fuel pool bridge hoist. The facility will be designed to prevent the cask handling crane from traveling over, or in the vicinity of, the pool, thereby precluding damage to the stored spent fuel in the event of a dropped cask (see Section 9.1.4 of this report). The racks will have a center-to-center spacing that is sufficient to maintain a maximum effective multiplication factor of 0.90 or less even if the pool were inadvertently filled with unborated water.

Based on our review, we conclude that the design criteria and bases for the spent fuel storage facilities are in conformance with the requirements of General Design Criterion 62 and the recommendations of Regulatory Guides 1.13, "Spent Fuel Storage Facility Design Basis," and 1.29, "Seismic Design Classification," including the recommendations on seismic design, missile protection design, and design compatibility with the handling of the fuel cask in the fuel pool areas, and are, therefore, acceptable.

9.1.3 Spent Fuel Pool Cooling and Cleanup System

The spent fuel pool cooling and cleanup system will be designed to remove the decay heat generated by stored spent fuel elements. A secondary function will be to clarify and purify the water in the spent fuel pool, in the transfer canal, and in the refueling cavities and borated water storage tank. The spent fuel pool cooling system will be designed to seismic Category I requirements. The piping system will be arranged so that loss of piping integrity or operator error will not result in inadvertent draining of the spent fuel pool water below a minimum depth.

The system will consist of two half-capacity trains that will dissipate heat to the component cooling water system. With both trains in service and 1/3 core in the pool, water temperature will not exceed 100 degrees Fahrenheit. With only one train in service, the pool water temperature may reach 150 degrees Fahrenheit. These temperature limits are acceptable.

Interconnections will be provided between the two cooling trains to assure uninterrupted cooling in the event of a pump or heat exchanger failure. The system will also be cross-tied with the decay heat removal system, which can be used as a backup when the reactor vessel is unloaded. One decay heat exchanger will be able to maintain the spent fuel pool water at lower than 135 degrees Fahrenheit when 1-1/3 cores are in the pool. Demineralized water and primary water will be provided as sources of makeup to compensate for losses during normal operation. Redundant seismic Category I makeup supplies will be provided from the borated water storage tank and service water system.

We reviewed the adequacy of the applicant's proposed design criteria and design bases for the spent fuel pool cooling and cleanup system necessary for continuous cooling during normal, abnormal, and accident conditions. We have concluded that the design criteria and design bases are in conformance with Criterion 61 of the General Design Criteria, and with the recommendations of Regulatory Guides 1.13 and 1.29, including the recommendations on seismic design, missile protection, and availability of assured makeup water systems. Therefore, we find the design of the fuel pool cooling and purification systems to be acceptable.

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9.1.4 Fuel Handling System

The fuel handling system will provide the means for transporting and handling fuel from the time it reaches the plant in an unirradiated condition to the time it leaves the plant after post-irradiation cooling. The fuel handling system will also provide for the safe disassembly, handling and reassembly of the reactor vessel head during refueling operations. The system consists of the fuel transfer canal, the manipulator crane, spent fuel pool handling bridges, fuel cask handling building crane, handling equipment, and the fuel transfer system.

Unacceptable damage due to a spent fuel cask drop will be prevented by limiting the travel of the spent fuel cask to an area that will contain no safety-related equipment or stored fuel. Travel of the cask bridge crane will be limited by mechanical stops, and by limit switches that will be electrically interlocked to prevent the crane from traveling over the spent fuel pool.

The cask pit will be separated from the fuel pool. In the unlikely event of a cask drop into the cask pit, the spent fuel pool will not be adversely affected. The arrangement of the cask loading pit and cask washdown area, relative to the spent fuel pool, will be such that, in the event of a cask drop, the cask cannot fall or tip into the spent fuel pool.

We have reviewed the adequacy of the applicant's proposed design criteria and design bases necessary for safe operation of the fuel handling system during normal, abnormal and accident conditions. We have concluded that the design criteria and design bases are in conformance with the positions of Regulatory Guide 1.13, including the recommendations regarding protection of the spent fuel storage facility from the impact of unacceptably heavy loads carried by overhead cranes. We, therefore, find the design of the system to be acceptable.

9.2 Water Systems

9.2.1 Service Water System

The service water system will provide cooling water to remove the heat from the safety-related plant auxiliary components during normal shutdown and loss of offsite power, and following the design basis loss-of-coolant accident. The system will provide water to the component cooling water heat exchangers, the control room emergency air conditioning system, the containment air coolers, and the emergency core cooling system room coils. It will also provide makeup water to the spent fuel pool system, and will serve as a backup water supply to the auxiliary feed-water system.

Service water will be taken from the existing intake forebay presently used by Unit 1. A portion of the forebay has been designed as a seismic Category I structure up to the intake canal structure. The intake canal structure and intake pipe are of nonseismic design. In the event of a postulated collapse of the canal

structure or intake pipe, the seismic Category I portion of the forebay would become the ultimate heat sink reservoir for Unit 1 and part of the ultimate heat sink for Units 2 and 3. The ultimate heat sink complex is discussed further in Sections 2.4.5 and 9.2.3 of this report.

At the station end of the forebay, two seismic Category I pump structures will be constructed next to the service water pump structure for Unit 1. Each structure, one for Unit 2 and one for Unit 3, will be designed to withstand tornado winds, tornado missiles, and floods. Each structure will house three 100 percent capacity service water pumps.

The service water pumps will be headered and cross connected, and will serve independent trains of each unit for proper plant operational flexibility. Under normal plant operation, two pumps will be in operation and one will be on standby. Under this condition, the two pumps supply service water for plant operations and main condenser cooling tower makeup. Under emergency operation, through automatic valve sequencing, one service water pump can provide the necessary service water for its unit. The other two pumps will be on standby service. Service water for Units 2 and 3, under emergency operation, will be discharged directly to the ultimate heat sink, which will be shared by Units 2 and 3.

To prevent freezing of the ultimate heat sink during the winter, a portion of the heated main condenser circulating water will be recirculated through the ultimate heat sink.

There will be no cross connections or sharing of the service water system between Units 2 and 3.

Based on our review, we conclude that the service water system design criteria and bases meet the requirements of General Design Criterion 44, regarding the capability of the system to transfer heat from the systems and components important to safety, and the requirements of General Design Criteria 45 and 46, regarding periodic inspection and testing. We conclude that the proposed service water system for Units 2 and 3 is acceptable.

9.2.2 Component Cooling Water System

The component cooling water system will be a closed system designed to provide cooling water to selected nuclear auxiliary components during normal plant operation and to safety-related systems following postulated accidents. The component cooling water system will be designed to remove decay and sensible heat from the reactor coolant system via the decay heat removal system during shutdown; cool the letdown flow to the chemical and volume control system during power operation; cool the spent fuel pool water; and provide cooling to dissipate the waste heat from various components during normal operation and postulated accident conditions.

The system will be designed to seismic Category I requirements, and will be designed to meet the single failure criterion by consisting of two redundant trains. Under accident conditions, both trains will be automatically aligned to cool only essential components. Further system operational flexibility will be provided by inter-connecting the pump suction, and the discharge headers, with remote and manually operated valves. Radiation monitors will be installed at selected points in each train to detect radioactive leakage from the primary coolant system.

Normal operation will be accomplished with one pump and one heat exchanger in service. This arrangement will also be capable of cooling the shut down unit following an accident.

Normal system makeup will be supplied to the surge tank from the demineralized water tank and the primary water storage tank. A seismic Category I makeup water supply will be provided from the service water system.

Cooling water will be supplied from the component cooling water pump header through one line that serves both seismic Category I spent fuel pool cooling heat exchangers. The proposed one-line system does not provide adequate redundancy for cooling the safety-related spent fuel pool cooling heat exchangers, in the event of a single failure. The applicant had committed to prevent boiling of the spent fuel pool, as follows. The applicant proposed to either redesign the component cooling water supply to the spent fuel heat exchangers by providing redundant headers or, in the event of a moderate energy line crack, to effect repairs before the pool reaches boiling. The applicant's proposal was unacceptable. We required and the applicant agreed to provide completely redundant component cooling water systems for the spent fuel pool heat exchangers.

The design of the component cooling water system will provide single supply and return lines for the four reactor coolant pumps. These lines will contain at least one valve for containment isolation, and will be designed to seismic Category I requirements up to the containment isolation valve. Inadvertent closure of any one of the valves would terminate the coolant flow to all of the pumps, which potentially may cause failure of more than one pump, resulting in unacceptable fuel damage. To prevent this condition, the applicant has committed to providing Quality Group C component cooling water lines to the four reactor coolant pumps, and safety grade instrumentation to sense cooling water return flow from each pump. In the event of a moderate energy line crack or inadvertent valve closure, the safety grade instrumentation will sense loss of cooling water flow and will automatically shut down the affected pumps. We find this meets our requirements and is, therefore, acceptable.

Based on our review we conclude that the component cooling water system design criteria and bases are in conformance with the requirements of General Design Criterion 44, regarding the ability to transfer heat from safety-related components

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to the ultimate heat sink under normal and accident condition, and the requirements of General Design Criteria 45 and 46 regarding inspections and tests, including functional testing and confirmation of heat transfer capabilities, and are, therefore, acceptable.

9.2.3 Ultimate Heat Sink

The source of water for the ultimate heat sink, under normal conditions, will be Lake Erie. Lake water will be supplied to the three service water pump houses through an intake system, which consists of an intake crib located about 3000 feet out into Lake Erie and piping that leads from the crib to a canal that connects to the forebay. In the event of an earthquake, the nonseismic portion of the canal may collapse. Under this condition; the seismic Category I portion of the intake canal (forebay) would become the ultimate heat sink reservoir for Unit 1 and part of the ultimate heat sink for Units 2 and 3. An onsite quarry forms the rest of the ultimate heat sink for Units 2 and 3. Our evaluation of the thermal capabilities of the ultimate sink is given in Section 2.4.5 of this report, and our evaluation of the structural integrity is given in Sections 2.6.4 and 2.6.5.

Redundant seismic Category I pumps and piping systems will connect the quarry, the forebay, and the service water system. Normal makeup to the quarry will be natural groundwater and water that is blown down from the cooling towers. Excess water in the quarry will be pumped to a collection box from which it will flow by gravity back to Lake Erie.

Based on our review, we conclude that the proposed design of the ultimate heat sink is in conformance with General Design Criterion 5, regarding sharing of structures and systems, and in conformance with the recommendations of Regulatory Guide 1.26, "Quality Group Classifications and Standards," and Regulatory Guide 1.29, "Seismic Design Classification," regarding quality group and seismic classification, and with Regulatory Guide 1.27, "Ultimate Heat Sink Design." We, therefore, conclude the proposed design of the ultimate heat sink for Units 2 and 3 is acceptable.

9.2.4 Condensate Storage Facilities

The condensate storage facility will provide deaerated feedwater for normal plant operation. Makeup water to the condensate storage will be delivered from the demineralized water storage tank. This facility also will provide sufficient water storage for emergency shutdown decay heat removal by the auxiliary feedwater system.

Each unit will be provided with two 250,000 gallon storage tanks located in a building adjacent to the turbine building. The tanks will not be designed to meet seismic Category I requirements. In the event the condensate storage tanks rupture, condensate level in the tank room will build up sufficient pressure to blow out the two doors into the turbine room, and the condensate will flow into the

condenser pit through floor gratings. This event could result in a unit shutdown; however, the safety of the plant will not be compromised.

Under normal operating or shutdown conditions, the proposed condensate storage capacity will be sufficient to remove reactor decay heat for 13 hours, plus a subsequent reactor cooldown to 280 degrees Fahrenheit. Further reactor cooldown will be accomplished using the decay heat removal system. Failures of either of the two condensate storage tanks will not preclude a safe reactor shutdown, since backup supply to the auxiliary feedwater pumps will be supplied by the seismic Category I service water system. The fire water system can also be used, if necessary.

As result of our review of the condensate storage system design criteria and design bases, we conclude the proposed system design is acceptable.

9.3 Process Auxiliaries

9.3.1 Station and Instrument Air System

The station and instrument air systems are nonsafety-related. However, certain air operated valves, that require actuation for safe shutdown of the reactor, will be provided with air accumulators. The accumulators, piping and valves affecting safety-related valve closure will be designed to seismic Category I requirements up to and including the compressed air supply isolation valves. The accumulator stored air pressure and volume will be sized to permit positioning of the valves to accomplish their necessary safety functions. The remaining air operated valves necessary for plant shutdown will be designed to fail in the safe position. Thus, the station and instrument air systems will not be needed during the process of plant shutdown.

As the result of our review of the design criteria and design bases, we conclude the compressed air system will perform its intended function and is, therefore, acceptable.

9.3.2 Equipment and Floor Drainage System

The equipment and floor drainage system will collect normal drainage from the auxiliary building, including the fuel handling area, and from containment. The system will collect water from potentially radioactive sources for processing in the liquid radwaste system. It will also collect nonradioactive drains, and will discharge them to the station external waste water system for disposal. Oil interceptors will be provided for drains in the emergency diesel generator and diesel fuel day tank rooms. Acid neutralizing tanks will be provided for all battery room drains. Turbine building equipment and floor drains will be collected in low point sumps, and will be pumped through oil separators before being discharged into the station external drain system.

Equipment necessary for the safe shutdown of each unit will be enclosed in separate watertight compartments or will be protected by walls, curbs, or pressure doors. Leakage will be sensed by level switches in sumps and tanks, by flow sensors in piping, and by pressure or temperature sensors in sump pump discharge lines. The alarms will be annunciated in the control room. Drainage systems will include anti-backflow check valves where necessary.

Based on our review, we conclude that the equipment and floor drainage system design criteria and design bases are adequate to protect safety-related areas and components from flooding and to prevent the inadvertent release of radioactive liquids to the environment due to piping or tank failure and are, therefore, acceptable.

9.3.3 Chemical and Volume Control System

The chemical and volume control system will consist of the makeup and purification system and the chemical addition and boron recovery system.

The chemical and volume control system will be designed to control and maintain reactor coolant inventory, to control the boron concentration, maintain the purity of the reactor coolant during normal operation, and provide high-pressure injection of borated water for emergency core cooling in the event of an accident. The makeup and purification system will purify the letdown fluid by remineralization. The chemical addition system will be designed to remove boric acid from the reactor coolant for reuse and to prepare and transfer chemicals and chemical solutions to the makeup and purification system. The makeup and purification system will also be designed to provide seal-water injection flow to the reactor coolant pumps, control the primary water chemistry and activity level by ion exchange and chemical addition, and process the reactor coolant to recover the boron and makeup water.

The makeup and purification system is safety related and will be designed to meet seismic Category I requirements. A single active failure will not prevent boration, because several alternate flow paths from redundant sources of boric acid will be available to add boron to the reactor coolant system.

The portion of the makeup and purification system used for high-pressure injection will also have redundancy to meet the single failure criterion.

Based on our review, we conclude that the makeup and purification system and the chemical addition and boron recovery system will be designed to meet their intended safety function and, therefore, we conclude that the systems are acceptable.

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9.4 Air Conditioning, Heating, Cooling and Ventilation Systems
9.4.1 Control Room

The control room ventilation system will be designed to maintain the control room within the thermal and air quality limits required for operation of plant controls and for uninterrupted safe occupancy of manned areas during normal operation, shutdown and post-accident conditions.

The system will consist of two subsystems, normal and emergency, each with redundant heating and cooling air handling units. The normal system will be non-seismically designed, and will be normally in operation to maintain control room environment. One of the normal air handling units will have sufficient capacity to maintain the desired environment in the control room areas while the second unit can be manually actuated by the control room operator in the event of failure of the operational unit. The emergency system will be a seismic Category I heating and cooling redundant system similar to the normal system but smaller in capacity. Each emergency system air handling package will contain a roughing filter, a high efficiency filter, and a charcoal filter. The emergency system will be normally held on standby, and will be used only during emergency or accident conditions to maintain the desired environment in the control room and other selected areas.

Redundant radiation and chlorine gas detectors will monitor the outside air, and will alarm in the control room. Initiation of these alarms will automatically isolate the outside air supply to the control room and actuate the recirculation of the control room air through the emergency air conditioning and filtration system.

Smoke detectors installed in the normal system ducts, and in the cable spreading room and heating and ventilation room ducts, will stop the air supply and exhaust fans for the respective areas, and will annunciate alarms in the control room. Fire dampers will be provided in all ductwork penetrating through fire walls.

The outside air intake and discharge ducts for the normal and emergency air conditioning system will employ redundant dampers to assure isolation of the control room from the outside environment when emergency operation is required. Outside air intakes or louvers will be tornado- and missile-protected. The control room heating and ventilation systems will be designed to maintain the control room under positive pressure with respect to all surrounding areas.

The applicant originally proposed to provide a single nonseismic supply fan unit and exhaust fan unit for the cable spreading room, to be used during normal and shutdown operation. The applicant revised the proposed system to include seismic Category I supports for the portions of the cable spreading room ventilation system whose collapse could result in damage to cables or loss of function of any safety-related system. This design change meets the applicable portions of Regulatory Guide 1.29, "Seismic Design Classification," and is acceptable.

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We have reviewed the design criteria and bases for the control room ventilation system, and conclude that the design criteria and design bases meet the requirements set forth in General Design Criterion 19, regarding the capability to operate the plant from the control room during normal and accident conditions. The system is, therefore, acceptable.

9.4.2 Auxiliary Building
Nonradwaste Area

The heating and ventilation systems for the nonradwaste areas of the auxiliary building of each Unit will be designed to provide a suitable environment for equipment and personnel. The ventilation and heating system for this area will serve the switchgear rooms, battery charger rooms, battery rooms, and other nonradwaste areas. It will be of nonseismic design.

Each battery room and switchgear room also will be provided with a seismic Category I, tornado missile-protected, louvered wall fan to maintain the required environment in the event of failure of the normal nonradwaste area ventilation system or during an emergency condition. The operation of this emergency ventilation system will be automatically controlled by a temperature control system, and can be remote manually started from the corresponding unit control room.

Smoke detectors will be provided in the return air ducts and main air supply ducts in compliance with National Fire Protection Association requirements. Fire dampers will be provided at all fire wall penetrations.

As a result of our review of the nonradwaste area heating and ventilation system design criteria and design bases, we conclude that the proposed system will provide protection under normal and postulated accident conditions and, therefore, the system is acceptable.

Radwaste Area

The radwaste area ventilation system will be independent of any other system employed in the auxiliary building. It will be designed for once-through flow to direct all potentially contaminated air to the station vent stack through a roughing filter and high efficiency particulate filters. The air from the radwaste area will be monitored before it is discharged to the station vent stack.

In the event radioactivity levels in the radwaste area exceed predetermined limits, the supply and exhaust fan units will be stopped, and the area will be exhausted by the auxiliary building seismic Category I emergency ventilation system. The seismic Category I portions of the radwaste ventilation system exhaust ducts will be automatically connected to the emergency ventilation system by ductwork

bypasses, dampers, and controls. The radiation monitors will sound an alarm in the control room to alert the control room operator to manually start the emergency ventilation system fans to exhaust the air through filters to the unit vent stack. The connections between the emergency ventilation system and the radwaste normal ventilation system will be automatically closed either by the safety features actuation system for post-loss-of-coolant accident operation, or by a high radiation signal from a fuel handling accident.

On the basis of our review and evaluation of the system design criteria and design bases, we conclude that the radwaste area ventilation system will be in accordance with General Design Criteria 61 and 64 and, therefore, is acceptable.

Fuel Handling Area

The fuel handling area heating, cooling, and ventilation system will be an independent system for each unit. The exhaust ductwork of the system will be connected to a seismic Category I emergency ventilation system. The emergency system will include prefilters, particulate filters and charcoal filters. The emergency system will contain redundant units to withstand a single failure without loss of function.

The normal system will be designed for once-through flow to control and direct all potentially contaminated air to the station vent stack via a filtering system. Exhaust air from this system will be monitored for radiation before being discharged. A monitor alarm will sound in the control room. The fuel handling area air exhaust duct, including the isolation dampers between the normal and emergency systems, will be designed to seismic Category I requirements. The remainder of the air exhaust system will be nonseismic design. Normal operation will be with the air supply fan and one exhaust fan operating. The other exhaust fan can be remote-manually actuated in the event of loss of the operating fan. The normal air exhaust system will be designed so that loss of both exhaust fans will stop the air supply fan. On loss of the air supply fan, the fuel handling area negative pressure will be maintained by the main station exhaust fans.

The connections between the normal and emergency ventilation systems will be automatically closed by the safety features actuation system for post-loss-of-coolant accident operations.

In the event of a fuel handling accident that results in the leakage of radioactivity, the fuel handling area air supply and air exhaust systems will be stopped, and the exhaust from the fuel handling area will be automatically transferred to the emergency exhaust system.

We have reviewed the system design criteria and bases for the fuel handling area ventilation system and conclude that they are in accord with General Criteria 61 and 64 and, therefore, are acceptable.

Emergency Core Cooling System Pump Room Cooling

The emergency core cooling system pump room cooling systems will be designed to seismic Category I requirements and to maintain a suitable environment to assure the operability of the pumps and motors.

There will be two pump rooms for each of Units 2 and 3. Each pump room cooling unit will consist of two 50 percent capacity fans and cooling coils, capable of being powered from the emergency diesel generators.

Based on our review of the design criteria and design bases, we conclude that there will be adequate redundancy to assure proper cooling and ventilation for the emergency core cooling pumps, and the ventilation system is, therefore, acceptable.

Emergency Diesel Generator Rooms

Each diesel generator room will be ventilated by two-half capacity fans to assure the required environment when the diesel generators are operating. The air supply will be 100 percent outside air and will be discharged directly to the atmosphere through the roof. The ventilation system, fans, ductwork, and control system will be designed to seismic Category I requirements. The fans will be interlocked with the diesel generators so that the fan will operate only when the diesel generators are in operation. Each diesel compartment will be provided with a unit heating system to maintain the compartment at required standby operating temperature.

The diesel generator building will be designed to include seismic Category I intake and exhaust provisions. The air intake for each diesel generator will be located outside the building. The exhaust structures will be located above the diesel generator roof and will be part of the roof structure. The physical separation between the inlets and exhausts for the diesel generators, and the location of the building, will be such as to preclude the possibility of fire extinguishing agents and other noxious gases from being drawn into the air intakes, and to preclude exhaust gas recirculation into the air intakes.

Based on our review, we conclude that the proposed diesel building ventilation system design will provide adequate redundancy to assure proper ventilation and air temperature conditions for the diesel generators. The proposed system is, therefore, acceptable.

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Service Water Pump Rooms

Each of the three service water pump rooms will have its own independent ventilation system. Each independent system will include one 100 percent capacity supply fan and one 100 percent capacity exhaust fan. The system will be designed to maintain the maximum room temperature at 100 degrees Fahrenheit during normal operation and 115 degrees Fahrenheit following a loss-of-coolant accident. These temperatures are within the limits for which safety-related equipment in these rooms will be qualified. The ventilation system fans, cooling coil, duct-work and controls will be designed to seismic Category I requirements. All air intake and exhaust openings will be provided with missile protection. System power supply will be taken from the independent essential Class IE source. The system meets single failure criteria since only one pump is required for safe plant shutdown.

Based on our review, we conclude that the above design criteria and bases provide adequate assurance that the service water pump rooms ventilation system design will provide sufficient ventilation for the equipment within the pump rooms to perform their safety-related function and are, therefore, acceptable.

9.5 Other Auxiliary Systems

9.5.1 Diesel Generator Fuel Oil System

The fuel oil system will be designed to provide fuel oil storage and transfer capability to allow operation of each standby diesel generator for at least seven days.

The fuel oil system will consist of two separate and independent trains, one for each diesel generator. Each system will include a day tank that will hold a one-hour supply of fuel oil for each standby diesel. The fuel oil system will be designed to seismic Category I requirements. The fuel oil storage tanks will be buried underground, and the transfer pumps will be located inside the fuel oil storage tank. The pumps will be powered from separate emergency buses. Based on our independent evaluation, we have determined that the design of the fuel oil system meets our single failure criteria.

Based on our review of the diesel generator fuel oil system design criteria and bases, we conclude the system will have adequate capacity and will be able to perform its designated safety functions, and is, therefore, acceptable.

9.5.2 Diesel Generator Auxiliary Systems

The diesel generator auxiliary systems will include the diesel generator cooling water system, the diesel generator air starting system, and the diesel generator lubrication system.

The diesel generator cooling water system will be an integral part of the diesel generator and will be designed to maintain the temperature of the diesel engine within a safe operating range. The system will be a closed cooling system, and the heat will be rejected to the component cooling water system. When the engine is idle, the engine water will be heated by electric heaters to keep the engine warm and ready to accept loads within the prescribed time interval. The system will be designed to seismic Category I requirements.

Each of the standby diesel generators will be provided with an independent compressed air starting system consisting of one air compressor and two storage tanks. The starting air systems will be designed to seismic Category I requirements. Each system will be capable of five cold starts without recharging.

Each diesel engine will be provided with a lubrication system, designed to supply lubricating oil to the diesel generators. The system, which is an integral part of the diesel generator, will circulate lube oil through the engine for heating when the engine is idle and for cooling when the engine is operating. The lube oil will be cooled by the component cooling water system and will be heated by an electric heater. The system will be designed to seismic Category I requirements.

Based on our review, we conclude that the diesel generator auxiliary systems design criteria and bases assure that these systems will meet their designated safety functions, and will have the needed capacity. They are, therefore, acceptable.

9.6 Fire Protection System

During our review of the fire protection system, we requested that the applicant conduct a reevaluation of the proposed fire protection provisions and that a detailed comparison be made with the guidelines in Appendix A to Branch Technical Position APCSB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants." The applicant provided an initial response to our request for information in June 1977, and we are reviewing that response.

The design, as presently proposed, meets the fire protection requirements of Criterion 3 of the General Design Criteria and of applicable guidelines in effect prior to issuance of Branch Technical Position APCSB 9.5-1. For the construction permit stage of the review, we find it to be acceptable. Final approval of the system will depend on our review of the applicant's submittal. Our review may or may not be completed prior to a decision on issuance of a construction permit. However, based upon our current review of the facility, sufficient flexibility exists in the design to allow implementation of any design changes that may be necessary to assure compliance with Appendix A to Branch Technical Position APCSB 9.5-1.

10.0 STEAM AND POWER CONVERSION SYSTEM

10.1 Summary

The steam and power conversion system will be of conventional design, similar to those of previously approved pressurized water reactor plants using the Babcock & Wilcox nuclear steam supply system. The system will be designed to remove heat from the reactor coolant system by generating steam in two once-through steam generators, then to convert the steam energy to electrical energy in the turbine generator. A condenser will transfer reject heat in the secondary cycle to the condenser cooling water. Reject heat will then be transferred to the atmosphere through a cooling tower. The entire system will be designed for the maximum design heat generation from the nuclear steam supply system.

Upon loss of full load, the system will be capable of dissipating the energy in the reactor coolant in the steam generators either through bypass valves to the condenser, or through power operated relief valves, dump valves and safety valves to the atmosphere.

10.2 Turbine Generator

The turbine-generator will consist of a tandem arrangement of one double-flow high-pressure turbine and three double-flow low pressure turbines driving a direct-coupled generator at 1800 revolutions per minute. The turbine will be equipped with an electro-hydraulic control system. The speed of the turbine will be controlled by modulating the turbine inlet steam control valves.

The control system will be designed to automatically trip the turbine under the following conditions: generator faults; main and auxiliary transformer faults; low condenser vacuum; excessive thrust bearing wear; high vibration; high exhaust hood temperature; low bearing oil pressure; turbine overspeed; turbine protection for generator motoring, reactor trip; manual trip (from control room or locally); low hydraulic oil pressure; prolonged loss of generator stator coolant; loss of direct current trip voltage; and backup overspeed.

The turbine generator will use electro-hydraulic controls for speed regulation employing two independent electrical inputs and one mechanical speed input. The mechanical overspeed trip will close the main and intermediate stop valves by spring force and hydraulic pressure, and the control and intercept valves by electro-hydraulic means. The station piping, feedwater heaters, and hydraulically actuated nonreturn valve systems will be designed to assure that entrained steam

cannot overspeed the unit beyond safe limits. The overspeed system will be designed so that loss of hydraulic fluid pressure leads to valve closing and consequent turbine shutdown. The mechanical trip valve will actuate at 110 percent of rated speed and the backup electro-hydraulic device will actuate at 112 percent of rated speed.

As a result of our review, we conclude that the turbine-generator overspeed protection design criteria and bases provide suitable redundant and diverse controls for preventing turbine-generator overspeed and are, therefore, acceptable.

10.3 Main Steam Supply System

The main steam supply system for each unit will utilize two once-through steam generators. The two main steam outlets from each steam generator will be headered inside containment, and will penetrate the containment and continue to the turbine as one steam line. Each main steam line will contain a set of ASME Code safety valves, atmospheric relief valve, one main steam isolation valve, and one main steam nonreturn valve.

The main steam isolation valves will be designed to seismic Category I, Quality Group B requirements, and will provide isolation for forward steam flow. The main steam isolation valve will be a pneumatically operated, balanced disc type valve, designed to close within 10 seconds after a steam line break. The valve will fail close on loss of air pressure. The valves will be designed to withstand dynamic forces resulting from valve closure, and will be missile protected. The main steam isolation valves will close automatically on a low reactor coolant pressure signal or a high containment building pressure signal. The valves will also be remote manually operated from the control room. Nonseismic Category I pneumatically operated nonreturn valves, located downstream of the isolation valves, will prevent reverse flow. These nonreturn valves will close automatically upon closure of the main steam isolation valves, and they will also be remote manually operated from the main control room.

As a result of our review, we conclude that the main steam supply system design criteria and bases, except as noted in Section 6.2.1.1 of this report concerning the nonreturn valves, are in conformance with the single failure criterion and to the recommendations of Regulatory Guide 1.29, "Seismic Design Classification," related to seismic design and valve closure time requirements. Therefore, we find these design criteria and bases to be acceptable, subject to the upgrading of the nonreturn valves to meet our requirements as stated in Section 6.2.1.1 of this report.

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10.4 Other Features of Steam and Power Conversion System

10.4.1 Main Condenser Evacuation System

The main condenser evacuation system will be designed to establish and maintain main condenser vacuum by transferring noncondensable gases from the condenser through a charcoal filter to the unit vent. The components of the system will be designed to Quality Group D and to a nonseismic design classification.

The scope of our review included the system capability to transfer radioactive gases to the ventilation systems, and the design provisions incorporated to monitor and control releases of radioactive materials in gaseous effluents in accordance with General Design Criteria 60 and 64. Based on our evaluation, we find the proposed main condenser evacuation system to be acceptable. The basis for our acceptance has been conformance of the applicant's designs, design criteria, and design bases for the main condenser evacuation system to the General Design Criteria 60 and 64.

10.4.2 Turbine Gland Sealing System

The turbine gland sealing system will be designed to control radioactive steam leakage from, and air leakage into, the turbine and large steam valve shaft seal glands. The components of the system will be designed to Quality Group D and to a nonseismic design classification. The turbine gland sealing system will consist of a steam seal header, steam seal regulator, and a gland seal condenser. Steam will be supplied to the shaft seals from the auxiliary boiler during startup, and from the high-pressure turbine packing during load operations. The gland seal condenser will condense water vapor and will exhaust the noncondensable gases to the atmosphere.

Our review included the source of sealing steam and the provisions incorporated to monitor and control releases of radioactive material in gaseous effluents in accordance with General Design Criteria 60 and 64.

Based on our evaluation, we find the proposed turbine gland sealing system to be acceptable. The basis for acceptance in our review has been conformance of the applicant's design, design criteria, and design bases for the turbine gland sealing system to General Design Criteria 60 and 64.

10.4.3 Circulating Water System

The circulating water system will be designed to remove the heat from the main condenser and turbine plant service water and dissipate that heat to the atmosphere by means of the cooling tower. The condenser will be connected to the circulating water piping with expansion joints located between the condenser and the motor operated isolation valves, one on each side of the condenser. All safety-related

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equipment required for safe shutdown, or required to limit the consequences of an accident, will be located in areas that would not be affected by potential flooding caused by rupture of nonseismic Category I piping and components of the circulating water system.

We reviewed the adequacy of the applicant's proposed design criteria and design bases necessary for safe operation of the circulating water system during normal, abnormal, and accident conditions. We conclude that the design criteria and design bases of the circulating water system are acceptable.

10.4.4 Auxiliary Feedwater System

The auxiliary feedwater system will be designed to supply an independent source of water to the steam generators, and to remove reactor decay heat when the main condensate and feedwater systems are not available. The auxiliary feedwater system will be designed to function automatically in the event of malfunctions, such as loss of power and breaks in main steam or feedwater lines. The auxiliary feedwater system will be designed to seismic Category I requirements and will be located in a tornado- and missile-protected structure. A separate startup feed pump system will be provided for normal startup and shutdown of the reactor.

The system pump redundancy will be provided by one 1050 gallons per minute (100 percent capacity) motor-driven pump and one 1050 gallons per minute turbine-driven pump. The motor-driven pump will be aligned to an essential bus powered from a diesel generator, and the turbine-driven pump will be powered by steam from the two steam generators through two direct current-powered normally closed isolation valves. The steam turbine can also be powered from the station auxiliary steam system. Each pump will be connected to a feedwater line feeding one steam generator through appropriate check and isolation valves. Piping crossover lines, with normally closed motor operated valves, will be provided to enable either auxiliary feedwater pump to feed one or both steam generators. All the valves required for operating the motor driven pump will be manually locked open, or will be motor-operated, powered from the alternating current emergency power source. Similarly, all the valves required for operating the turbine driven pump will be manually locked open, or will be motor-operated, powered from the direct current power source. The proposed design, therefore, satisfies the power diversity requirements of our Branch Technical Position APCS 10-1, "Design Guidelines for Auxiliary Feedwater System Pump Drive and Power Supply Diversity for Pressurized Water Reactor Plants."

Normally, the pumps will take suction from the condensate storage tank, which is not designed to withstand the safe shutdown earthquake but which will contain enough water to keep the reactor coolant temperature down to 280 degrees Fahrenheit for 13 hours. Backup water sources will be provided from the nonseismic fire protection system and from the seismic Category I service water system.

We have reviewed the adequacy of the applicant's proposed design criteria and bases of the auxiliary feedwater system necessary for safe operation of the plant during normal, abnormal, and accident conditions. We conclude that the proposed design conforms with our technical positions regarding diversity of power sources, system flexibility, and redundancy, including the capability of the system to withstand the combination of single active and high energy line failures, in accordance with Criteria 44 and 45 of the General Design Criteria, and is, therefore, acceptable.

10.5 Steam and Feedwater System Materials

The mechanical properties of materials to be selected for Class 2 and Class 3 components of the steam and feedwater systems will satisfy Appendix I of Section III of the ASME Boiler and Pressure Vessel Code, and Parts B and C of Section II of the Code. The fracture toughness properties of the ferritic materials will satisfy the requirements of Articles NC-2300 and ND-2300 of Section III of the ASME Code.

The controls to be imposed upon austenitic stainless steel comply with the requirements of NRC Interim Position MTEB 5-1 on Regulatory Guide 1.31, "Control of Stainless Steel Welding," and Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel." Fabrication and heat treatment practices that will be performed in accordance with these requirements provide reasonable assurance that stress corrosion cracking will not occur during the design life of the plant. The controls to be placed upon concentrations of leachable impurities in nonmetallic thermal insulation, used on austenitic stainless steel components of the steam and feedwater systems, are in accordance with Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel."

The welding procedures to be used in limited access areas satisfy the recommendations of Regulatory Guide 1.71, "Welder Qualification for Areas of Limited Accessibility." The onsite cleaning and cleanliness controls during fabrication satisfy the recommendations given in Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," and the requirements of ANSI Standard N45.2-1973, "Cleaning of Fluid Systems and Associated Components for Nuclear Power Plants." The precautions to be taken in controlling and monitoring the preheat and interpass temperatures, during welding of carbon and low alloy steel components, meet the recommendations given in Regulatory Guide 1.50, "Control of Preheat Temperature for Welding Low-Alloy Steel."

Conformance with the codes, standards, regulatory guides, and NRC staff positions mentioned constitutes an acceptable basis for assuring the integrity of steam and feedwater systems, and for meeting the applicable requirements of General Design Criterion 1.

10.6

Water Hammer

We are currently evaluating design and operating conditions that could result in damage to feedwater system piping as a consequence of feedwater flow instability occurrences such as occurred at Indian Point 2 (Docket Number 50-247) on November 13, 1973. The results of our generic investigation will be reported at the operating license stage of review.

11.0 RADIOACTIVE WASTE MANAGEMENT

11.1 Summary

Each reactor unit will have separate radioactive waste management systems designed to provide for controlled handling and treatment of liquid, gaseous, and solid wastes. The liquid waste system will process wastes from equipment and floor drains, decontamination and laboratory wastes, condensate demineralizer backwash wastes, and laundry and shower wastes. The gaseous waste system will provide holdup capacity to allow decay of short-lived noble gases stripped from the primary coolant, and treatment of ventilation exhausts through high efficiency particulate air filters and charcoal adsorbers to reduce releases of radioactive materials to "as low as is reasonably achievable" levels in accordance with 10 CFR Part 20 and 10 CFR Part 50.34a. The solid waste system will provide for the solidification, packaging and storage of radioactive wastes, generated during station operation, prior to shipment offsite for burial at a licensed facility.

In our evaluation of the liquid and gaseous radwaste systems, we have considered: (1) the capability of the systems for keeping the levels of radioactivity in effluents "as low as is reasonably achievable" based on expected radwaste inputs over the life of the plant, (2) the capability of the systems to maintain releases below the limits in 10 CFR Part 20, Appendix B, Table II, Columns 1 and 2, during periods of fission product leakage at design levels from the fuel, (3) the capability of the systems to meet the processing demands of the station during anticipated operational occurrences, (4) the quality group and seismic group design classification applied to the system design, (5) the design features that will be incorporated to control the releases of radioactive materials in accordance with General Design Criterion 60, and (6) the potential for gaseous release due to hydrogen explosions in the gaseous radwaste system

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

In our evaluation of the process and effluent radiological monitoring and sampling systems, we have considered the system's capability: (1) to monitor all normal and potential pathways for release of radioactive materials to the environment, (2) to control the release of radioactive materials to the environment, and (3) to monitor

the performance of process equipment and to detect radioactive material leakage between systems.

In the Final Environmental Statement for Davis-Besse Units 2 and 3, we indicated that we had not completed our review of the radwaste systems to meet the requirements of Appendix I of 10 CFR Part 50, issued May 5, 1975, because the assumptions and models for calculating radioactive effluent releases were being reassessed. We have completed the reassessment of our models and assumptions, and the applicant has chosen to comply with the September 4, 1975 amendment to Appendix I rather than submit a cost-benefit analysis as required by Paragraph II.D. On this basis, we have reassessed the radwaste systems, using source terms calculated with the revised models and methods described in NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWRs)," April 1976. The source terms for Units 1, 2, and 3 are given in Appendix A of this report. In estimating the offsite doses, we have considered the source term from Unit 1 along with the source term from Units 2 and 3.

Based on our reassessment of the liquid radioactive waste management systems, we estimate that the quantity of radioactive materials to be released in liquid effluent, excluding tritium and dissolved noble gases, will be less than five Curies per year per reactor and that the total calculated quantity of radioactive materials released in liquid effluents from Units 1, 2, and 3 will not result in an annual dose or dose commitment exceeding five millirem to the total body or to any organ of an individual, in an unrestricted area, from all pathways of exposure. Based on our reassessment of the gaseous radioactive waste management systems, we estimate that the total quantity of radioactive materials to be released in gaseous effluents from all three units will not result in a calculated annual gamma air dose in excess of ten millirads or a beta air dose in excess of 20 millirads at any location near ground level, at or beyond the site boundary, which could be occupied by individuals. We estimate that the annual total quantity of iodine-131 to be released in gaseous effluents will not exceed one Curie per year per reactor, and that the calculated total quantity of radioiodine and radioactive particulates to be released in gaseous effluents from the three units will not result in an annual dose or dose commitment in excess of 15 millirem to any organ of an individual, in an unrestricted area, from all pathways of exposure.

Our evaluation of the proposed liquid and gaseous radioactive waste management systems for Units 2 and 3 shows these systems to be capable of meeting the criteria given in Appendix I of 10 CFR Part 50 for keeping releases of radioactive materials to the environment "as low as is reasonably achievable," and, therefore, we find the proposed systems to be acceptable.

Based on our evaluation, as described below, we find the liquid, gaseous, and solid radwaste and associated process and effluent radiological monitoring systems to be acceptable.

11.2 System Description and Evaluation

11.2.1 Liquid Radioactive Waste Treatment System

The liquid radioactive waste treatment system will consist of process equipment and instrumentation necessary to collect, process, monitor, and recycle or dispose of radioactive liquid wastes. All potential radioactive liquids generated in the plant, except for the turbine building floor drain liquids, will be collected and processed through the radioactive liquid waste treatment system prior to release to the cooling tower discharge. The turbine building floor drains will normally be monitored and released without treatment, but can be transferred to the liquid waste treatment system. The liquid radioactive waste will be processed on a batch basis to permit optimum control of releases. Prior to being released, wastes will be analyzed to determine the types and amounts of radioactivity present. Based on the results of the analyses, the waste will be retained for further processing or will be released under controlled conditions.

The liquid waste treatment system will be designed to collect and process wastes based on the chemical purity, relative to the primary coolant, as determined by the origin of the waste in the plant.

The liquid waste treatment system will consist of two subsystems: (1) the clean liquid radioactive waste system, and (2) the miscellaneous liquid radioactive waste system.

The clean liquid radioactive waste system will process shim bleed and equipment drain waste. This system will consist of two 93 cubic foot mixed bed primary demineralizers, two 15 gallon per minute capacity boric acid evaporators, and two 14 cubic foot polishing demineralizers. Two 103,000 gallon receiver tanks upstream of the evaporator will provide surge capacity to allow batchwise operation, and two 23,000 gallon monitor tanks will allow sampling and monitoring of the evaporator condensate. We estimate the system input flow to be approximately 3,000 gallons per day and the system design capacity to be 21,700 gallons per day based on the evaporator flow rate.

The miscellaneous liquid radioactive waste system will process miscellaneous low purity wastes collected in floor drains and building sumps, laboratory and sample drains, detergent tanks, and the condensate polishing demineralizer holdup tanks. The holdup tanks will collect wastes from the backwashing of the condensate polishing demineralizers, which are of the powdered resin type. The decant from these tanks will be slurried to the solid radwaste system. The nondetergent waste portion of the miscellaneous liquid radioactive waste system will consist of a 13,400 gallon miscellaneous waste drain tank, a 15 gallon per minute waste evaporator, a 14 cubic foot mixed bed polishing demineralizer, and an 8700 gallon waste monitoring tank. We estimate that the nondetergent waste portion of the system input flow to be approximately 2900 gallons per day and the system design capacity to be 21,600 gallons per day based on the waste evaporator flow rate.

The detergent waste portion of the system will be collected in a separate 7300 gallon tank, from which it will be sampled and discharged. We calculate that these wastes will be approximately 450 gallons per day, and the system design will permit use of the waste evaporator should additional treatment be required.

The turbine building floor drain wastes will normally be released, without treatment after monitoring for radioactivity. If the radioactivity in these releases exceeds a predetermined level, the wastes will be transferred to the liquid waste treatment system.

For the above two subsystems of the liquid waste treatment system, the difference between the expected flows and the design capacities will provide adequate reserve for processing surge flows. We consider the system capacity and system design to be adequate for meeting the demands of the station during anticipated operational occurrences.

The liquid waste treatment system will be located in the auxiliary building, which will be designed to be seismic Category I. The design parameters of principal components considered in the liquid radwaste evaluation are listed in Table 11.1. We find the applicant's proposed liquid waste treatment system design to be acceptable in accordance with Branch Technical Position ETSB 11-1, Revision 1, "Design Guidance for Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Power Plants." The system will also be designed to control the release of radioactive materials, due to overflows from tanks outside containment, by providing level instrumentation that will alarm in the control room. Spillage from these tanks will be collected in the equipment drain tank for reclamation. All liquid radwaste storage tanks that might contain significant quantities of radioactivity will be housed within the reactor compartment, and we calculate that no tank will contain more than 10 Curies. We consider these provisions to be capable of preventing the uncontrolled release of radioactive materials to the environment.

We have determined that, during normal operation, the proposed liquid radwaste treatment systems will be capable of reducing the release of radioactive materials in liquid effluents to approximately 0.25 Curie per year per reactor, excluding tritium and dissolved gases, and 550 Curies per year per reactor for tritium.

11.2.2 Gaseous Radioactive Waste Treatment Systems

The gaseous radwaste treatment system will be designed on the basis of the origin of the wastes in the plant and their expected activity levels. The gaseous waste treatment system will process gases stripped from the primary coolant and miscellaneous tank cover gases through a 30 standard cubic feet per minute compressor and moisture separator, and through three decay tanks. Each decay tank will hold 103 cubic feet and will be designed for a pressure of 150 pounds per square inch gauge.

TABLE 11.1

DESIGN PARAMETERS OF PRINCIPAL COMPONENTS CONSIDERED
IN THE LIQUID RADWASTE EVALUATION

<u>COMPONENT</u>	<u>NUMBER</u>	<u>CAPACITY EACH</u>
<u>Clean Liquid Radioactive Waste</u>		
Receiver Tanks	2	103,000 gallons
Primary Demineralizers	2	140 gallons per minute (93 cubic feet of resin)
Polishing Demineralizers	2	40 gallons per minute (14 cubic feet of resin)
Boric Acid Evaporators	2	15 gallons per minute
<u>Miscellaneous Liquid Radioactive Waste System^a</u>		
Waste Drain Tank	1	13,400 gallons
Waste Monitor Tank	1	8,700 gallons
Waste Polishing Demineralizer	1	40 gallons per minute (14 cubic feet of resin)
Waste Evaporator	1	15 gallons per minute
Detergent Waste Drain Tank	1	7,300 gallons
<u>Miscellaneous^a</u>		
Spent Resin Storage Tank	1	580 cubic feet
Concentrate Storage Tank	1	780 gallons
Evaporator Storage Tank	1	780 gallons

^aQuality Group and Seismic Design in accordance with Branch Technical Position ETSB 11-1 Revision 1.

Redundant compressors and moisture separators will be provided to allow operation during periods of equipment downtime. The principal components in the gaseous radioactive waste treatment system, along with their principal design criteria, are listed in Table 11.2.

We consider the system capacity and the system design to be adequate for meeting the demands of the plant during anticipated operational occurrences.

The gaseous waste treatment system will be located in a seismic Category I structure, and will be designed to quality group and seismic design standards compatible with Branch Technical Position ETSB 11-1 Revision 1.

The gas decay tanks will receive inputs from the waste gas surge tank portion of the waste gas system and from the cover gas line portion. The system design will include dual oxygen analyzers on both portions of the system. The analyzers will initiate an alarm if oxygen concentrations rise above the design concentration limits. The system design will also include an automatic control function that will close the system inlets if oxygen concentrations reach a predetermined level. Inclusion of these features will reduce the potential for explosive hydrogen/oxygen mixtures. We find the system quality group and seismic design criteria, and the design provisions incorporated to reduce the potential of hydrogen explosions, to be adequate.

Gaseous wastes from the main condenser will be released to the unit vent without treatment. If the activity exceeds a predetermined level, the release will be through a charcoal adsorber. The system releases will be proportional to the rate of primary-to-secondary system leakage and the primary coolant activity. In the event of excessive primary-to-secondary leakage, the affected steam generator will be isolated before radioactive material concentrations in main condenser offgas releases exceed the applicable limits of 10 CFR Part 20.

The nonradioactive areas of the auxiliary building will be ventilated with a once through system, and the ventilation air will be exhausted to the environment without treatment. Ventilation air from potentially radioactively contaminated areas of the auxiliary building will be exhausted through high efficiency particulate air filters, and through charcoal adsorbers if activity levels are above a predetermined value.

The containment building ventilation system will consist of a high-capacity containment vessel purge system and a low-capacity containment vessel purge system. During reactor outages, the high-capacity system will purge the containment through high efficiency particulate filters. The low-capacity system will purge the containment, during normal power operation, through high efficiency particulate filters and charcoal adsorbers. The turbine building ventilation exhaust will be released through the turbine building roof vent without treatment. The charcoal

TABLE 11.2

DESIGN PARAMETERS OF PRINCIPAL COMPONENTS^a
CONSIDERED IN THE GASEOUS RADWASTE EVALUATION

<u>COMPONENT</u>	<u>NUMBER</u>	<u>CAPACITY EACH</u>
<u>Gaseous Waste Processing Systems</u>		
Waste Gas Surge Tank	1	1030 cubic feet
Compressors	2	30 standard cubic feet per minute
Decay Tanks	3	1013 cubic feet

^aQuality Group and Seismic Design in accordance with Branch Technical Position ETSB 11-1
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adsorbers used in the ventilation systems will provide a decontamination factor of 10 for radioiodine.

The plant ventilation systems will be designed to induce air flows from potentially less radioactively contaminated areas to areas having a greater potential for radioactive contamination. Potentially contaminated building areas will be maintained at a slightly negative pressure with respect to the exterior pressure. This will reduce exfiltration, and will promote collection of radioactive materials by the ventilation system for dispersion through roof and plant vent exhausts. The ventilation system will have adequate capacity to limit radioactive material concentrations, in areas within the plant that are accessible during operation, to below the limits in 10 CFR Part 20.

We have determined that the proposed gaseous radwaste treatment systems and plant ventilation system will be capable of reducing the release of radioactive materials in gaseous effluents, from each unit, to approximately 5900 Curies per year of noble gases, 0.13 Curie per year of iodine-131, 570 Curies per year of tritium, eight Curies per year of carbon-14, and 0.004 Curie per year of particulates from each unit.

11.2.3 Solid Radioactive Waste Treatment System

The solid radwaste treatment system will be designed to collect and process wastes based on their physical form and on the need for solidification prior to packaging. "Wet" solid wastes, consisting of spent demineralizer resins, evaporator bottoms, spent filter cartridges, and spent powdered resins from the condensate washing demineralizers, will be combined with a solidification agent and catalyst mixture to form a solid matrix, and will then be sealed in 50 cubic foot casks. Dry solid wastes, consisting of ventilation air filters, contaminated clothing and paper, and miscellaneous items such as tools and glassware, will be compacted into 55-gallon drums, using an industrial baling machine.

"Dry" wastes will be compacted using an industrial hydraulic baler. During compaction, drums will be enclosed in a dust shroud that will be vented to the plant vent to preclude releasing dusts to the operating area.

Casks will be filled by pumps that bring together radwaste and a liquid solidification agent. Waste transfer piping and solid radwaste system components will be designed to the guidelines given in Branch Technical Position ETSB 11-1, Revision 1.

We determined that the expected solid waste volumes and activities shipped annually offsite will be approximately 12000 cubic feet of "wet" solid waste containing approximately 1700 Curies total, and approximately 4100 cubic feet of "dry" solid waste containing less than five Curies total. Storage facilities to accommodate approximately five drums will be provided within the auxiliary building.

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Based on our estimate of expected solid waste volumes, we find the storage capacity adequate for meeting the demands of the plant. Wastes will be packaged in accordance with the requirements of 10 CFR Part 20, 10 CFR Part 71 and 49 CFR Parts 170-178, and will be shipped, in accordance with NRC and Department of Transportation regulations, to a licensed burial site.

11.3 Process and Effluent Radiological Monitoring System

The process and effluent radiological monitoring system will be designed to provide information concerning radioactivity levels in systems throughout the plant, indicate radioactive leakage between systems, monitor equipment performance, and monitor and control radioactivity levels in plant discharges to the environs.

Liquid and gaseous streams will be monitored. Table 11.3 indicates the proposed locations of continuous monitors. Monitors on certain effluent release lines will automatically terminate discharges, should radiation levels exceed a predetermined value. Systems that are not amenable to continuous monitoring, or for which detailed isotopic analyses are required, will be periodically sampled and analyzed in the plant laboratory.

We have reviewed the locations and types of effluent and process monitoring provided. Based on the plant design and on the continuous monitoring locations and intermittent sampling locations, we have concluded that all normal and potential release pathways will be monitored. We have also determined that the sampling and monitoring provisions will be adequate for detecting radioactive material leakage to normally uncontaminated systems and for monitoring plant processes that affect radioactivity release. On this basis we conclude that the monitoring and sampling provisions meet the requirements of General Design Criteria 60, 63 and 64 and the guidelines of Regulatory Guide 1.21, "Measuring and Reporting of Effluents from Nuclear Power Plants."

11.4 Conclusions

Our review of the radwaste systems included (1) system capabilities to process the types and volumes of wastes expected during normal operations and anticipated operational occurrences, in accordance with General Design Criterion 60, (2) design provisions incorporated in accordance with General Design Criterion 60 to preclude uncontrolled release of radioactive material due to leakage or overflows and (3) the quality group classification and seismic design criteria for conformance with staff technical positions. We have reviewed the applicant's system descriptions, process flow diagrams, piping and instrumentation diagrams, and design criteria for the components of the radwaste treatment systems, and for those auxiliary supporting systems that are essential to the operation of the radwaste treatment systems. We have performed an independent calculation of the releases of radioactive materials in liquid and gaseous effluents.

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TABLE 11.3

PROCESS AND EFFLUENT MONITORING LOCATIONS

STREAM MONITORED

Liquid*

Component Cooling Water
Liquid Waste Releases**
Service Water Discharge
Reactor Coolant Letdown
Station Effluent

Gaseous*

Containment Purge**
Condenser Air Ejector
Gaseous Waste Discharge**
Radwaste Area Exhaust
Fuel Handling Area

*All liquid and gas streams will be monitored in accordance with the guidelines of Regulatory Guide 1.21.

**These monitors will alarm and automatically terminate the release when the radiation level exceeds a predetermined level.

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Our review of the process and effluent radiological monitoring and sampling systems included the provisions proposed for (1) sampling and monitoring all station effluent, in accordance with General Design Criterion 64, (2) automatic termination of effluent releases and assuring control over discharges in accordance with General Design Criterion 60 and Regulatory Guide 1.21, (3) sampling and monitoring of plant waste process streams for process control in accordance with General Design Criterion 63, (4) conducting sampling and analytical programs in accordance with the guidelines of Regulatory Guide 1.21, and (5) monitoring process and effluent streams during postulated accidents. The review included piping and instrument diagrams and process flow diagrams for the liquid, gaseous, and solid radwaste systems and ventilation systems, and the location of monitoring points relative to effluent release points on the site plot diagram.

Based on the foregoing evaluation, we conclude that the above aspects of the radwaste treatment and monitoring systems are acceptable. The basis for acceptance has been conformance of the applicant's designs, design criteria, and design bases for the radioactive waste treatment and monitoring system to the applicable regulations and guides referenced above, as well as to staff technical positions and industry standards.

12.0 RADIATION PROTECTION

The radiation protection program for Davis-Besse Units 2 and 3 is discussed in Chapter 12 of the Preliminary Safety Analysis Report. In this chapter the applicant has described how he plans to manage the radiation protection program so as to keep radiation exposures within the limits of 10 CFR Part 20 and to maintain exposures as low as is reasonably achievable. The Preliminary Safety Analysis Report includes discussion of design features, such as shielding and layout of facilities, the radiation monitoring systems, the ventilation system for providing a suitable radiological environment, and the health physics program, to assure that exposures will be as low as is reasonably achievable.

The acceptability of the applicant's program is based on the criterion that doses to personnel will be maintained within the established limits of 10 CFR Part 20, "Standards for Protection Against Radiation," and on the consistency of the radiation protection design and program features with the guidelines of Regulatory Guide 8.8, "Information Relevant to Maintaining Occupational Radiation Exposures As Low As Practicable." In response to our requests for additional information, the applicant has added extensive material to Preliminary Safety Analysis Report Chapter 12 concerning the implementation of design features for assuring that occupational radiation exposures will be as low as is reasonably achievable. On the basis of our review, we conclude that implementation of the applicant's radiation protection program will provide reasonable assurance that personnel doses will be maintained as low as is reasonably achievable and below the limits established by 10 CFR Part 20. Further, we believe that the applicant's radiation protection design features are consistent with the guidelines of Regulatory Guide 8.8.

12.1 Shielding

Radiation shielding at Davis-Besse Units 2 and 3 will be designed to assure that (1) the criteria of 10 CFR Part 20 and 10 CFR Part 50 are met during normal operations and anticipated operational occurrences; (2) operating personnel will be adequately protected in the event of a reactor accident; and (3) activation of components will be minimized so as to reduce personnel exposure during refueling, maintenance, and inspection operations. These design objectives have been chosen by the applicant to assure that occupational radiation exposures will be as low as is reasonably achievable. This is consistent with the guidance given in Section C.3 of Regulatory Guide 8.8 and is, therefore, acceptable.

All plant areas will be divided into radiation zones. There will be five zone classifications, all based on the limiting of personnel occupation time in radiation

areas and, thereby, maintaining occupational radiation exposure as low as is reasonably achievable and within 10 CFR Part 20 limits. The dose rate criterion for each zone, based on the radiation sources in each compartment within the zone, will be used as the basis for the radiation shielding design. Since the source terms used in the shielding calculations have conservatively assumed one percent failed fuel, the dose rate in the vicinity of tanks, filters, demineralizers, degasifiers, evaporators, coolers, and other process equipment is expected to be lower than the zone designations indicate.

The applicant's shield design is based on plant operation at maximum design power, including the release of fission products from failed fuel, or on TID 14844 ("Calculation of Distance Factors for Power and Test Reactor Sites," TID 14844, USAEC) accident releases where applicable. The applicant evaluated each room, valve station, sample station, and pipeway for potential radiation sources during normal full power operation, including anticipated operational occurrences, for maintenance occupancy requirements and for general access requirements. The applicant used finite cylindrical volume sources as geometric models for evaluation of shielding for tanks, heat exchangers, filters, demineralizers, evaporators, and the containment. Pipes were modeled as infinite cylinders. The applicant will use shielding models based on acceptable formulations presented by T. Rockwell ("Reactor Shielding Design Manual," D. Van Nostrand Company, New York, 1956), and by R. G. Jaeger ("Engineering Compendium on Radiation Shielding, Shielding Fundamentals and Methods," International Atomic Energy Agency, 1966), and will use acceptable standard point kernel codes, such as the QAD-P5A code, the Grace I and II codes, and the SDC code, and an equivalent line source code. We find these shielding models and computer codes acceptable.

Radioactive components and piping will be located in separate shielded cubicles in order to minimize radiation exposures to personnel during maintenance and inspection activities. Valve and instrumentation stations, and motor-operated or diaphragm-operated valves, will be used whenever feasible. Manually operated valves will be operated remotely, using reach rods, from shielded corridors. All valves, instrumentation and other components in high radiation areas will have a design life of 40 years. Shielding will be provided for pipe chases and penetrations, and will include offsets to minimize streaming. Field-run process piping will be routed to minimize radiation exposure to personnel. Some additional measures to be taken to reduce exposures are decontaminatable walls in radioactive equipment cubicles, adequate and rapidly serviceable lighting in cubicles, floor drains with properly sloping floors, display and control instrumentation located in low radiation areas, and provisions for components, located in high radiation areas, to be removed for repair work. Large radius pipe bends, sloped transfer lines, butt-welded valves and connections, and valves with few crevices will be used in resin and sludge treatment systems to minimize crud traps. These design features are consistent with the guidelines of Regulatory Guide 8.8.

There will be 44 area radiation monitors located throughout the plant, each having gamma dose rate readouts both locally and in the control room. Both audible and visual alarms will be provided at the local readout unit and in the control room. Area monitors will be source checked periodically and will be calibrated, when required, using a portable calibration unit. In general, area radiation monitors will be located where access is unrestricted or partially restricted and where radiation levels could possibly increase due to postulated occurrences. In addition to the use of fixed area monitors, area radiation surveys will be conducted on a routine weekly basis. Thus, operating personnel will have a continuous knowledge of areas that may contain changing levels of radioactivity. The objectives and location criteria of the applicant's area radiation monitoring system are in conformance with 10 CFR Parts 50 and 70 and the guidelines of Regulatory Guide 8.8 and are acceptable.

By keeping radiation exposures as low as is reasonably achievable, the applicant hopes to minimize annual man-rem at Davis-Besse Units 2 and 3. His health physics procedures will be based on procedures and work experience at the US Department of Energy Savannah River Plant, the Connecticut Yankee Atomic Power Company Haddam Neck Plant, and the Rochester Gas & Electric Company Robert E. Ginna Station. He will also have available his own operating experience with Davis-Besse Unit 1. By using techniques described in Regulatory Guide 8.8, the applicant plans to limit the annual personnel exposure at Davis-Besse Units 2 and 3 to 200 man-rem per unit per year for operating and maintenance crews. The method that the applicant used to make this estimation is acceptable to us. Although this estimate is less than the average collective dose, in man-rem per year, to personnel associated with present-day pressurized water reactor plants, it does not take into account unexpected major equipment outages, operations that take place less frequently than once a year, or doses to contractor personnel. We expect that the extra exposure contribution from these activities will result in an average annual collective dose to personnel at the Davis-Besse plant of approximately 400-500 man-rem per year per reactor. As modified by the expected dose contributions attributed to unexpected repairs to major equipment, to contractor personnel, and to infrequent operations, we find the applicant's exposure estimates to be reasonable and consistent with the acceptance criteria in our Standard Review Plan.

12.2 Ventilation

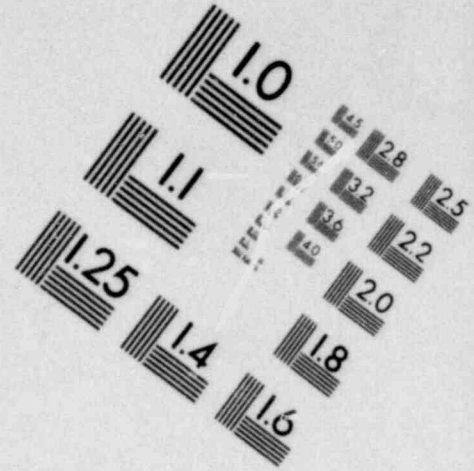
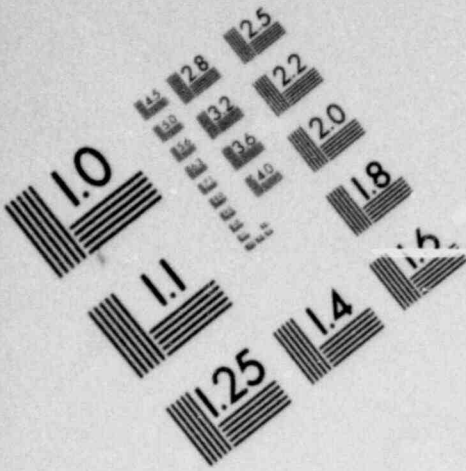
The ventilation system will be designed to protect personnel and equipment from extreme thermal environmental conditions and to assure that personnel are not inadvertently exposed to airborne concentrations exceeding the limits given in 10 CFR Part 20. The applicant intends to meet these objectives, and maintain personnel exposures as low as is reasonably achievable, by (1) maintaining air flow from areas of lesser potential airborne contamination to areas of higher contamination; (2) assuring negative or positive pressures to prevent exfiltration or infiltration of potential contaminants; (3) using once-through exhaust systems,

with prefilter and high efficiency particulate filter banks, in the fueling handling and radwaste areas; (4) providing sufficient airflows in all areas to keep airborne radioactivity levels as far below 10 CFR Part 20 limits as is reasonably achievable. These design criteria are in accordance with those given in Regulatory Guide 8.8 and are acceptable. The air filtration in the control room will be designed to limit radiation exposure to control-room personnel in accordance with General Design Criterion 19. Nonradioactive areas will be served by a separate ventilation system. Part of the ventilation air for the auxiliary building will be used to sweep the entire spent fuel storage pool surface to minimize buildup of airborne radioactivity in the fuel handling building.

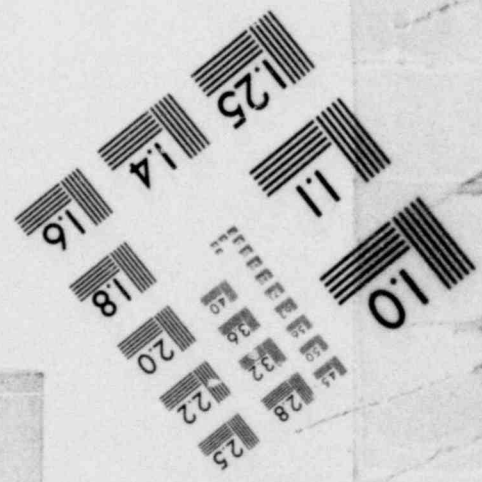
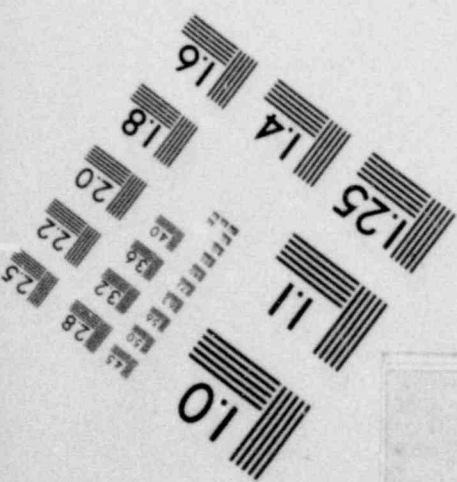
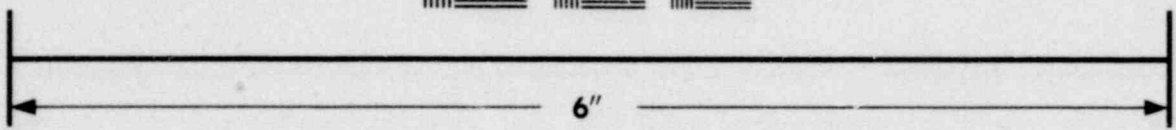
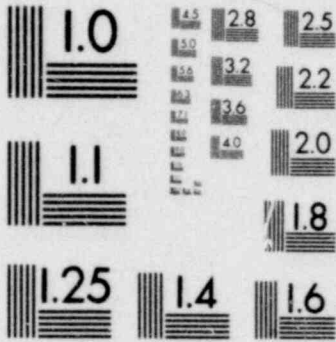
The objective of the airborne radioactivity monitoring system is to measure the levels of airborne radioactivity at various locations in the plant and to assist in maintaining radiation levels within the requirements of 10 CFR Part 20. The airborne radioactivity monitoring system will (1) provide continuous indication and record of airborne radiation levels for the fuel handling and radwaste areas; (2) include local indication and high radiation alarms at each monitor location; and (3) assure that abnormal levels of radioactivity will be detected and routed through the emergency ventilation system. The radiation monitors will be located upstream of the filters in the fuel handling area, radwaste area and penetration rooms. In addition to the airborne radioactivity monitors, the applicant will perform in-plant airborne radioactivity level surveys on a routine monthly basis. If these surveys show that airborne activity levels are higher than expected for normal conditions, or if operating conditions indicate that higher than normal levels could be expected, the frequency of the surveys will be increased. The applicant's airborne radioactivity monitoring system satisfies its design objectives, and conforms to the regulatory positions of Regulatory Guide 8.8 and of Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Engineered-Safety Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants."

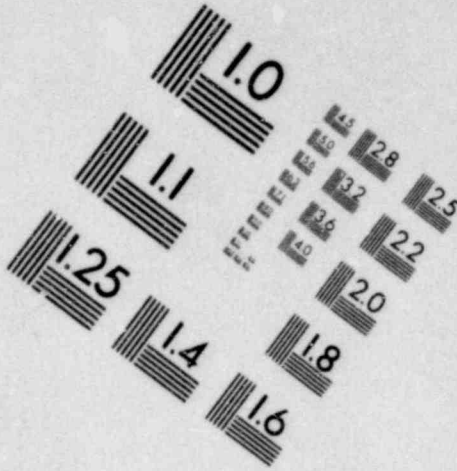
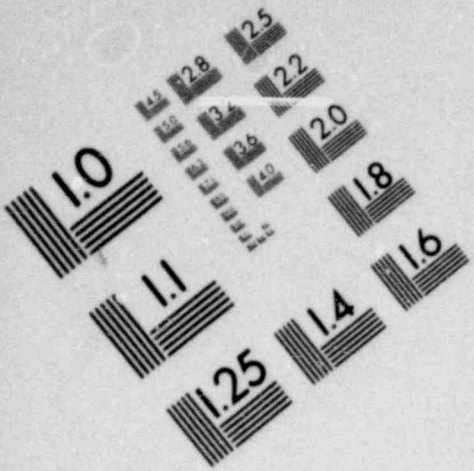
Onsite inhalation exposures will be kept as low as is reasonably achievable during normal operations and maintenance, by personnel training, area surveillance, contamination control, and proper work procedures. Besides mandatory health physics training classes given to all unit personnel, all personnel who use protective clothing will be instructed in its use and application. Portable continuous air monitor and smear surveys will be used to detect contaminated areas. All permanently assigned personnel, and all those suspected of having been exposed to airborne radioactive materials, will be given whole body counts. These practices are consistent with Regulatory Guide 8.8 and are acceptable.

Estimates of inhalation dose and peak airborne radioactivity concentrations for each building are based on operating experience from similar reactors. The applicant predicts that the inhalation doses will be a very small fraction of 10 CFR Part 20 limits. Based on the applicant's assumptions, we find these estimates to be acceptable.

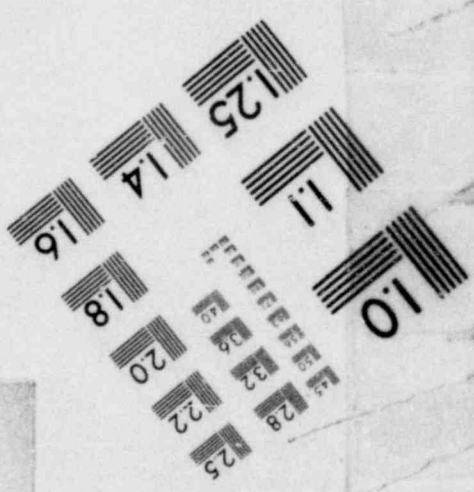
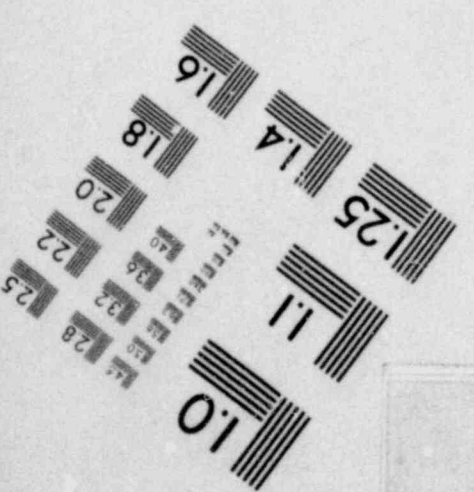
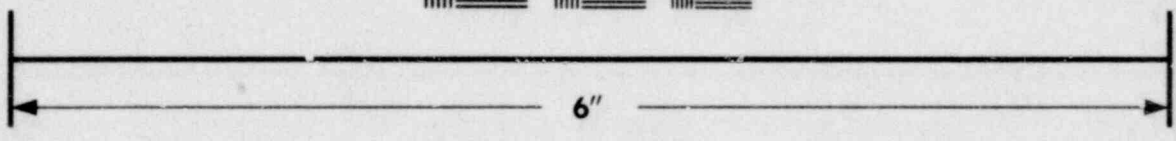
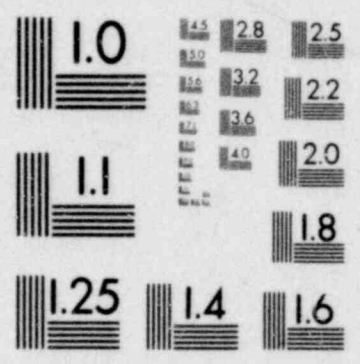


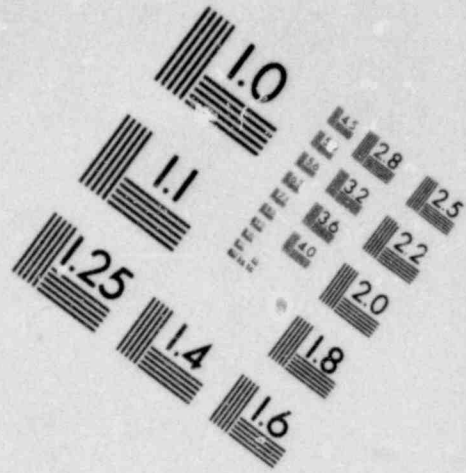
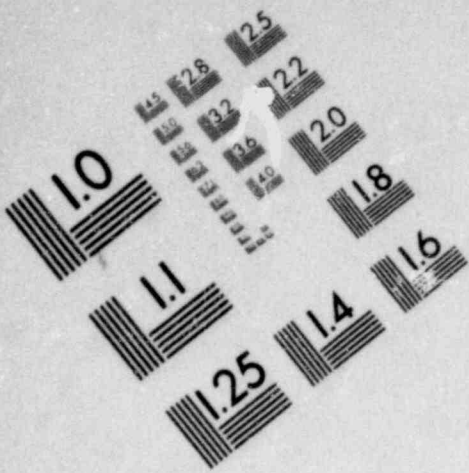
**IMAGE EVALUATION
TEST TARGET (MT-3)**



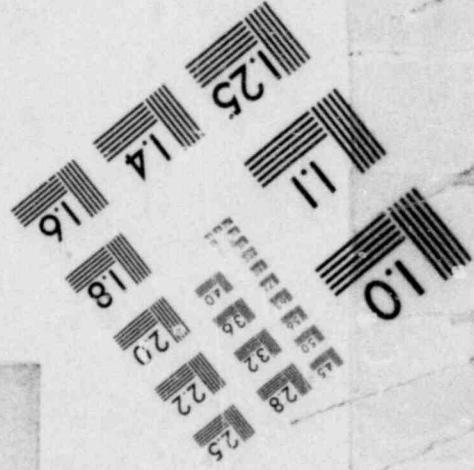
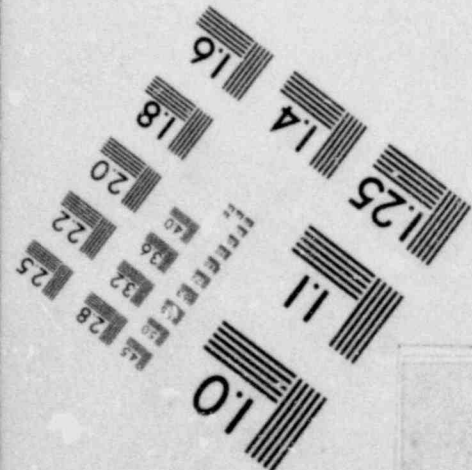
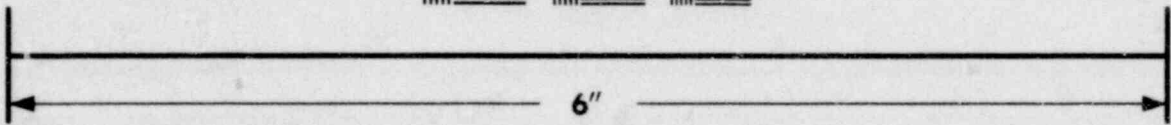
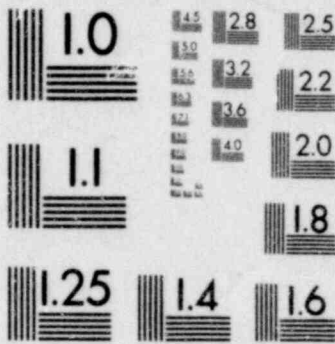


**IMAGE EVALUATION
TEST TARGET (MT-3)**





**IMAGE EVALUATION
TEST TARGET (MT-3)**



Health Physics

The objective of the applicant's health physics program is to provide protection equipment and administrative controls that will assure that radiation exposures will be as low as is reasonably achievable and within 10 CFR Part 20 limits. The station chemist and health physicist will be in charge of this program. He will report directly to the station superintendent on matters concerning any phase of radiological protection. Duties of the chemistry and health physics section will include monitoring and controlling the radiation exposure of personnel, continuously evaluating and reviewing the radiological status of the station, making recommendations for control or elimination of radiation hazards, training personnel in radiation safety, and protecting the health and safety of the public, both onsite and in the surrounding area. The objective of the health physics program and the ways in which it will be implemented are in accordance with Regulatory Guide 8.8 and are acceptable.

The radiation protection facilities will include a controlled access checkpoint, change room, protective clothing robing and disrobing area, decontamination facilities, radiochemical laboratory, counting room, instrument calibration room, health physics office, and a laundry. We consider that these facilities will be sufficient as part of the overall program, to maintain occupational exposures as low as is reasonably achievable, and their inclusion is consistent with Regulatory Guide 8.8.

Equipment to be used for radiation protection purposes will include portable radiation measuring instruments, personnel dosimetry instruments, calibration sources, area monitors, airborne activity monitors, laboratory equipment, air sampling equipment, respiratory equipment, and protective clothing. The counting room will contain instrumentation such as sodium iodide and germanium (lithium) detectors for gamma spectrometry, a liquid scintillation counter for tritium, and Geiger-Mueller and proportional counters for gross alpha, beta, and gamma counting. We consider that use of this equipment will contribute to the applicant's program to maintain occupational exposures as low as is reasonably achievable.

All personnel will be assigned a thermoluminescent dosimeter to monitor external beta-gamma radiation. Additional thermoluminescent dosimeters will be attached to extremities when the extremity dose could be higher than the whole body dose. For neutron dosimetry, film or an equivalent will be used. Self-reading dosimeters will be issued to those individuals whose work conditions make day-to-day indication of exposures desirable, and these will be maintained by the health physics staff for recording daily exposures. Dosimeter records will furnish the exposure data necessary for administration of the control of radiation exposures. A bioassay program, consisting of whole body counting, will be conducted annually on individuals selected on the basis of quarterly whole body exposure or of their work history in airborne radioactivity areas. All radiation exposure information will be processed and recorded in accordance with 10 CFR Part 20.

Based on the information presented in the Preliminary Safety Analysis Report and the applicant's responses to our requests for additional information, we conclude that the applicant intends to implement a radiation protection program that will maintain inplant exposures within the applicable limits and will keep radiation exposures as low as is reasonably achievable.

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13.0 CONDUCT OF OPERATIONS

13.1 Organizational Structure of Applicant

The Toledo Edison Company has responsibility for the design, construction, quality assurance and operation of the Davis-Besse Nuclear Power Station, Units 2 and 3. Babcock and Wilcox Company will be responsible for the nuclear steam supply system, and the Bechtel Power Corporation will act as the architect-engineer for the project. By letter of intent, United Engineers and Constructors, Inc., will construct Units 2 and 3. The Davis-Besse project is under the direction of Toledo Edison's Vice President, Facilities Development, who is responsible for carrying out Toledo Edison's responsibility for the design and construction of the project. The operational responsibilities will be under the direction of the Vice President Energy Supply. Quality assurance aspects of the project are discussed in Section 17.0 of this report.

The proposed station staff for the operation of the station (Units 1, 2 & 3) will consist of a technical staff of approximately 300 people under the direction of the Station Superintendent and Assistant Superintendent. Reporting to the Station Superintendent will be five technical sections. The Operations Section, under the supervision of the Station Operations Engineer, will be responsible for the day-to-day operation of the station. Reporting to the Station Operations Engineer will be a Unit Operations Engineer for each unit, responsible for the day-to-day operation of his unit. The shift crew for each unit (which reports to its respective Unit Operations Engineer) will consist of five people, one of whom will be a licensed senior reactor operator and two of whom will be licensed reactor operators. The Technical Section, under the supervision of the Station Technical Engineer, will be responsible for reactor engineering, station performance, and instrument and control maintenance. Reporting to the Station Technical Engineer will be a Unit Technical Engineer for each unit. The Chemistry and Health Physics Section, under the supervision of the Station Chemist and Health Physicist, will be responsible for station radiological protection and chemistry. Reporting to the Station Chemist and Health Physicist will be a Health Physicist, a Chemical Engineer, and, for each unit, a Chemical and Health Physics Foreman. The Maintenance Section, under the supervision of the Station Maintenance Engineer, will be responsible for the mechanical and electrical maintenance at the station, except for instrument and control maintenance. Reporting to the Station Maintenance Engineer will be a Unit Maintenance Coordinator or Supervisor for each unit. The Reliability Section, under the direction of the Reliability Engineer, will be responsible for outage planning, reliability, and inservice inspection coordination.

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The applicant has described his proposed minimum qualification requirements for the plant staff. We find these meet the qualification requirements described in ANSI N18.1-1971, "Selection and Training of Nuclear Power Plant Personnel," and Revision 1 to Regulatory Guide 1.8, "Personnel Selection and Training."

Technical support for the station staff will be provided primarily by the Power Engineering and Construction Group, that reports to the Vice President Facilities Development. Additional support will be provided by the Transmission and Substation Engineering Division that reports to the General Superintendent Transmission and Substation.

Based on (1) our review of the applicant's corporate and technical organization; (2) the technical resources as embodied in the numbers and technical experience of personnel assigned and available to the project; (3) the applicant's participation in the Davis-Besse Unit 1 project; (4) the Quality Assurance Program discussed in Section 17.0 of this report; and (5) the exchange of technical information experienced in our meetings and correspondence during the course of this review; we conclude that the applicant is technically qualified to design and construct Davis-Besse Units 2 and 3.

We further conclude that the proposed plant organization, the proposed qualifications of personnel, and the proposed plans for offsite technical support, are acceptable for the construction permit stage of review.

13.2 Training Program

Responsibility for the conduct and administration of the overall training program rests with the Training Supervisor. He reports to the Station Operations Engineer for license training and the Assistant Station Superintendent for all other training.

The objective of the training program is to provide the necessary formal training, observation training and actual work experience to all assigned individuals so that each person has a balanced period of experience in operations of a generating station and is fully qualified to carry out his assigned duties. The personnel presented for initial licensing on Unit 2 or Unit 3 will hold an operating license on Unit 1 or Unit 2, respectively. Specialized training will be provided, as necessary, in those areas where Unit 2 or Unit 3 may differ in design or operating characteristics from Unit 1. All formal training for the plant staff should be completed well in advance of fuel loading. This should allow the plant staff sufficient time to aid in the preparation of operating and startup procedures and checkout of systems.

The training program is to be developed by the Toledo Edison Company, with assistance from the General Physics Corporation. Various formal training courses will

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be given. The attendance at these courses will be a function of the needs of each man, depending on his background, previous training and job assignments. For personnel to be licensed, the courses will include: Academic Training for Nuclear Power Plant Personnel; PWR Observation Training; PWR Technology; and PWR Operation. The Babcock & Wilcox simulator may be used as a training supplement. Maintenance and technical staff personnel will receive specialized training in their particular fields. All station personnel will receive training in the station quality assurance program, health physics and emergency procedures.

The information submitted relative to the training program is acceptable, at the construction permit stage of review, to give reasonable assurance that qualified individuals will be available for the preoperational test program, for operator licensing, and for fuel loading.

13.3 Emergency Planning

The applicant has prepared an emergency plan for the Davis-Besse Nuclear Power Station. This plan, which we reviewed and approved during the operating license stage for Unit 1, will encompass all three units.

We have evaluated the applicant's plans for coping with emergencies, as presented in the Preliminary Safety Analysis Report, against the requirements of paragraphs I and II of Appendix E to 10 CFR Part 50, using NUREG-75/087, Standard Review Plan, Section 13.3, "Emergency Planning," for interpretive guidance. The emergency situations listed include fire, vehicular or transportation accidents, natural disasters, medical injury or illness, radiation and contamination accidents, civil disturbance, and reactor accident. The Station Superintendent will be responsible for the preparation and maintenance of the emergency plans. The normal operating crew, headed by the on-duty shift foreman, will perform the actions necessary to institute immediate protective measures and to implement their plans. An emergency control center will be activated, in the event of an offsite emergency, and will be staffed by qualified members of the station staff. Communications will be established with the operating crew and with offsite support groups. Radiation monitoring teams will perform onsite and offsite surveys under the direction of an Emergency Duty Officer. In addition to backup technical support supplied by Toledo Edison Company, arrangements have been made with several offsite support groups for specialized assistance.

The plans propose to categorize radiological emergencies in three classifications, local, site, and offsite. For each classification, the plans generally describe in-plant actions to be taken under each class. The applicant notes that emergency response arrangements have already been established for the notification and participation of local, State and Federal agencies for Unit 1 at the Davis Besse site and that these will be expanded to include Units 2 and 3. These agencies and other organizations include:

U. S. Energy Research & Development Administration, Argonne, Illinois*
Ohio Department of Agriculture, Columbus, Ohio
Radiation Management Corporation, Philadelphia, Pennsylvania
Magruder Memorial Hospital, Port Clinton, Ohio
Ottawa County District Board of Health, Port Clinton, Ohio
Ottawa County Sheriff, Port Clinton, Ohio
Oak Harbor Fire Department, Oak Harbor, Ohio

The agreement with the Ottawa County Sheriff's Department provides assurance that prompt emergency action will be initiated in the environs of the plant upon notification by the Station Superintendent or the Emergency Duty Officer that protective measures for the public may be necessary.

A first aid station is already located on the site, and provides for personnel monitoring, decontamination, and emergency medical treatment. Appropriate equipment, survey instruments and medical supplies are available at the station. The services of a company physician are available, as required. Arrangements have been made with H. B. Magruder Memorial Hospital and the Hospital of the University of Pennsylvania for the treatment of contaminated and injured personnel. Transportation will be provided by the Robinson Funeral Home Ambulance Service.

The Emergency Plan will be reviewed annually by the Station Review Board, which will recommend the updating of the implementing procedures as necessary. Training on the contents of the Emergency Plan will be provided for all personnel on the station operating staff. Drills will be conducted to help develop and maintain the competence of operating personnel in handling all types of emergencies. Simulated drills involving offsite agencies are also planned.

The plant will be designed, and will incorporate features, to assure the capability of plant evacuation, and of re-entry to mitigate the consequences of an accident, including radiation emergency alarms, communications systems, and evacuation routes. The plant control rooms will be designed for continuous occupancy during and following the most severe accidents, as analyzed in Chapter 15 of the Preliminary Safety Analysis Report.

We have reviewed the applicant's preliminary plans for coping with emergencies. We conclude that they meet the applicable requirements of 10 CFR Part 50, Appendix E; are consistent with facility design features, analyses of postulated accidents, and characteristics of the proposed site location; and provide reasonable assurance that appropriate protective measures can be taken within and beyond the site boundary in the event of a serious accident. They are, therefore, acceptable for the construction permit review stage.

*Now the U. S. Department of Energy

13.4 Review and Audit

Provisions for the review and audit of design and construction activities, prior to the issuance of an operating license, are described in Preliminary Safety Analysis Report Chapter 17 and in our evaluation of the Quality Assurance Program given in Section 17.0 of this report.

The applicant has committed to a review and audit program, for the review and audit of plant operations, that will meet Section 4 of ANSI N18.7-1972, "Administrative Controls for Nuclear Power Plants." This program meets the staff position described in Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)." A detailed review of the applicant's program will be performed during the operating license review.

13.5 Plant Procedures

All safety-related operations will be conducted with written and approved procedures. The following categories of procedures will be used:

- Systems Procedures
- Unit Procedures
- Emergency Procedures
- Instrument Calibration and Test Procedures
- Maintenance Procedures
- Health Physics Procedures
- Radiochemistry Procedures
- Chemistry Procedures
- Test Procedures
- Alarm Procedures
- Surveillance Tests
- Periodic Tests
- Administrative Procedures
- Miscellaneous Procedures

Administrative and Operating Procedures will be written and approved at least three months prior to fuel loading.

The information submitted relative to these subjects is satisfactory for the construction permit stage of review.

13.6 Plant Records

The applicant has stated that plant records will be maintained in accordance with Section XVII of Appendix B of 10 CFR Part 50, and ANSI N18.7, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants." We conclude that these record keeping provisions are acceptable.

13.7 Industrial Security

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

On February 24, 1977 the Commission published new requirements for the physical protection of nuclear power plants against acts of sabotage (10 CFR 73.55). This new rule does not require applicants for construction permits to demonstrate compliance at this stage, but does require such demonstration at the operating license stage. As a result of our review of the applicant's preliminary plans for physical security, we conclude that the applicant has described a satisfactory planning base upon which a complete security program can be developed to demonstrate compliance with the new regulations and to provide an acceptable level of physical protection to this site at the appropriate time. We will continue to work with and provide guidance to the applicant to this end.

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14.0 INITIAL TESTS AND OPERATIONS

We have completed our review of the information provided in the Preliminary Safety Analysis Report on the initial test program. The review included:

- (1) Evaluation of the scope of the applicant's test program including the responsibilities and qualifications of participating organizations, the general testing objectives, the divisions between major phases of the test program, the administrative controls governing the test program, and the extent to which the test program would verify the functional adequacy of the facility.
- (2) Evaluation of the testing proposed for unique or first-of-a-kind design features of the facility.
- (3) Evaluation of the applicant's plans to Regulatory Guide 1.68, "Initial Test Programs for Water-Cooled Reactor Power Plants," and other Regulatory Guides applicable to testing.
- (4) Evaluation of the applicant's plans to use operating experiences from other reactors in developing his test program.
- (5) Evaluation of the applicant's test program schedule to establish that sufficient time for testing is planned and that the schedule is compatible with the schedules for the hiring and training of plant personnel.
- (6) Evaluation of the applicant's plans to utilize plant operating and emergency procedures during preoperational testing.
- (7) Evaluation of the applicant's plans to augment the station staff to perform the testing.

On the basis of this review, we have concluded that assurance has been provided that the applicant has established acceptable advanced plans for the initial test program. We have also concluded that the information provided satisfies the acceptance criteria in Section 14.1 of the Standard Review Plan. We will conduct an additional review of the initial test program during the operating license review stage.

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15.0 ACCIDENT ANALYSIS

15.1 Introduction and Classification of Events

The applicant has submitted, in Preliminary Safety Analysis Report Chapter 15, a series of safety analyses that evaluate the capability of the facility to withstand normal and abnormal operational transients and a broad spectrum of postulated accidents without undue risk to the health and safety of the public. The postulated events have been taken from the list of representative types of events to be analyzed given in the "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 1, issued October 1972. The applicant has categorized the events into different classes as follows:

<u>Class</u>	<u>Definition</u>
1	Events leading to no radioactive release at exclusion area boundary. Class 1 events are abnormal operational transients.
2	Events leading to small-to-moderate radioactive releases at exclusion area boundary. Class 2 events are off-design operational transients or accidents.
3	Design basis accidents. Accidents of very low probability.

- ✓ The events analyzed by the applicant are listed by category in Table 15.1 of this report.

15.2 Input Parameters for Transients and Accident Analysis

As part of our review of the facility's transient and accident analyses, we reviewed the assumptions and input parameters used by the applicant in these analyses. A discussion of the more significant assumptions and input parameters follows in this section.

The parameters used by the applicant in these transient and accident analyses are listed in Preliminary Safety Analysis Report Table 15.1-2. The analyses presented in the Preliminary Safety Analysis Report have been based on initiation of the event at 102 percent (2828 megawatts thermal) of the rated core power level except that events that produced more severe consequences at a lower power were assumed to be initiated at the lower power.

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TABLE 15.1

ANALYZED INCIDENTS AND FAULTS BY CATEGORY

Class 1

Uncontrolled control rod group withdrawal from subcriticality.
Uncontrolled control rod group withdrawal at power.
Control rod misoperation (stuck-out, stuck-in, or dropped control rod).
Makeup and purification system malfunction.
Loss of forced reactor coolant flow.
Startup of an inactive reactor coolant loop.
Loss of external electrical load and/or turbine trip.
Loss of normal feedwater.
Loss of all alternating current power to the plant auxiliaries.
Heat removal greater than heat generation.
Failure of regulating instrumentation.
Internal and external events (fires, earthquakes, etc.).
Control room uninhabitability.
Failure of low pressure portion of decay heat removal system.
Loss of condenser vacuum.
Turbine trip with failure of generator breaker to open.
Turbine trip with coincident failure of turbine bypass valves to open.
Loss of service water system.
Loss of one direct current system.
Inadvertent operation of emergency core cooling system during power operation.
Loss of instrument air system.
Loss of turbine gland sealing system.

Class 2

Inadvertent loading of a fuel assembly into an improper position.
Small spills or leaks of radioactive material outside containment.

Class 3

Fuel cladding failure combined with steam generator leak.
Loss-of-coolant accident.
Steam line break.
Waste gas decay tank rupture.
Steam generator tube rupture.
Rod ejection accident.
Break in instrument line or lines from primary system that penetrate containment.
Fuel handling accident.

The applicant states that no operator actions or nonsafety-related control system actions will be required for reactor protection. No actions of the nonsafety-related control system are assumed unless such actions would produce more serious consequences. The nonsafety-related control systems are discussed in Section 7.7 of this report.

The uncertainties resulting from allowable operating bands and measurement uncertainties are reflected in the assumed initial conditions and the assumed trip setpoints. Added conservatism is included in the analysis by use of high-level or low-level (whichever is more conservative) steam generator inventory, and minimum or maximum tank volumes. The delays, associated with the reactor protection system trips and control rod drop time, are given in the Preliminary Safety Analysis Report. The control rod drop times are based on the use of silver-indium-cadmium control rods.

Except for the emergency core cooling system calculations described in Section 6.3 of this report, the analyses are based on undensified fuel. Penalties due to fuel densification will be considered at the final design review stage, if appropriate.

15.3

Abnormal Operational Transients

A number of plant transients can be expected to occur, as a result of equipment malfunction or operator error in the course of refueling and power operation. These events will, at worst, result in a reactor trip, with the plant being capable of a return to normal operation.

The transients have been analyzed to be sure that they will not violate the following criteria:

- (1) Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design pressures.
- (2) Clad integrity shall be maintained by assuring that the minimum departure from nucleate boiling ratio does not fall below 1.32. For certain transients, this criterion is confirmed by showing that core thermal power does not exceed 112 percent, with the reactor coolant flow, power peaking, reactor coolant system pressure and core outlet temperature remaining near their normal operating values or varying in a direction to increase the margin in the departure from nucleate boiling ratio.

The applicant has analyzed these transients by the use of several different computer codes. Our review of the CADD, RADAR, and POWERTRAIN computer codes has progressed to the point that there is reasonable assurance that analytic results will not be appreciably altered by any revision to the codes that we may require. These codes are acceptable for this application, and any revisions that may result from our completed review will be implemented at the operating license stage.

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Moreover, we will require, during the power ascension testing, that the applicant verify that the dynamic response of the plant for these and other transients can be predicted by an approved analytical method. Therefore, we conclude that the analytical methods used for these events is acceptable for the construction permit stage of review.

The applicant has submitted analyses for these moderate frequency events to show that these events do not result in a breach of the above criteria. We have reviewed the Class 1 events given in the Preliminary Safety Analysis Report and the following describes the bounding incidents and our evaluations.

15.3.1 Uncontrolled Control Rod Group Withdrawal

Uncontrolled control rod group withdrawal presupposes either an operator error or equipment malfunction. Such events have been reviewed as occurring both at power and from a subcritical or low power startup condition. The scope of our review has included consideration of initial conditions, control rod reactivity worths, and the course of the transient, including instrument and safety system response. Initial conditions and assumptions for the analyses were determined to be conservative.

We have reviewed the range of parameters assumed in these analyses and the results of the calculations. The transients are terminated by the negative Doppler coefficient, the high pressure trip, or the nuclear overpower trip.

We conclude that the calculations contain sufficient conservatism, with respect to both input assumptions and models, to assure that neither fuel damage nor overpressure will occur as a result of a control rod group withdrawal transient, and that the plant design is acceptable in this regard.

15.3.2 Control Rod Misoperation

The case of control rod misoperation (rod stuck or dropped) has been examined. The limiting case, that of a rod dropped into the core, has been analyzed. The integrated control system acts to prohibit rod withdrawal and to initiate a power runback to 60 percent power whenever a rod is more than nine inches removed from the average position of its group. No credit is taken from this action in the analysis. A further conservatism consists of using the combination of the largest peaking factor increase, due to a dropped rod, and the largest worth rod dropped, even if these two factors do not coincide.

Analyses have been performed for both beginning-of-life and end-of-life conditions. At beginning-of-life, the core power drops to about 60 percent of full power and remains there. The reactor coolant system pressure and temperature fall because heat is being removed at a rate greater than that at which it is being produced.

The integrated control system, a nonsafety-grade system, is assumed not to operate. The reactor finally trips on low pressure or low temperature. The departure from nucleate boiling ratio during this transient has been analyzed and it does not become less than 1.32.

At end-of-life, the moderator and Doppler coefficients are more negative, and the core power at first drops rapidly but then returns to approximately full power where it remains. The moderator pressure and average temperature decrease, and the reactor stabilizes in this new condition. Trip does not occur. In the new stabilized state, peaking factors may be higher than normal. However, tests in the Rancho Seco Unit 1 reactor, a 177 fuel assembly plant, have shown that no thermal limits are exceeded at full power when the rod causing the largest peaking factor change is dropped into the core.

On the basis of our review, we conclude that the plant design for the rod drop transient is acceptable.

15.4 Design Basis Accidents

Accidents are limiting design basis events that are not expected to occur, but are postulated because their consequences include a potential for the release of significant amounts of radioactive material. Accidents are used as a basis for evaluating the various barriers and other protective features included in the facility design.

15.4.1 Reactor Coolant Pump Rotor Seizure

Seizure of a reactor coolant pump rotor results in a sharp reduction of reactor coolant flow. This degradation in core heat transfer conditions could result in fuel damage.

Initial conditions assumed were 102 percent power with four pumps in operation. No credit was taken for integrated control system functions. The reactor protection system is assumed to trip the reactor on high power/flow ratio. The calculated minimum departure from nucleate boiling ratio is 1.72, and the reactor coolant pressure increased 45 pounds per square inch.

Our review considered the analytical model, the extent to which normally operating plant instrumentation and controls were assumed to function, and the extent to which operator actions are required.

We conclude that the analysis was performed with conservative assumptions in that no credit was taken for the integrated control system, and no operator actions were required in the near term to mitigate the consequences.

We conclude that the plant design is acceptable, in terms of fuel damage and pressure limits, with regard to possible seizure of a reactor coolant pump rotor.

15.4.2 Rod Ejection Accident

This design basis accident is assumed to be caused by the physical failure of a pressure barrier component in a control rod drive mechanism, resulting in the rapid ejection of a control rod assembly. The maximum worth rod that may be ejected will be limited by technical specifications to one percent reactivity at zero power and 0.65 percent reactivity at full power.

We have reviewed the rod ejection analysis presented in the Preliminary Safety Analysis Report, and have concluded that it conforms to the recommendations of Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors." Analyses were performed for both zero power and full power conditions at beginning of life and at end of life. The limiting case is that at full power at beginning of life. The environmental consequences of the postulated accident are shown to be acceptable.

Three-dimensional effects are treated in the analysis by assuming a larger than normal radial peaking factor (to account for the effect of the ejected rod) with the design axial peaking factor. The peaking factor is assumed to be unchanged during the transient. Actually, the power would be depressed prior to initiation of the transient, and would peak during the transient. To compensate for the neglect of the change in peaking factor, the reactivity feedback during the transient is based on the average rod in the core (i.e., a peaking factor of 1.0). Calculations in two dimensions (using the TWIGL code) have shown that this procedure is conservative in the expected range of ejected rod worths.

Birkhofer, et al., have reported the need for three-dimensional, time-dependent calculations to predict correctly the peak flux and temperature distributions for super-prompt-critical reactivity excursions. Babcock and Wilcox has submitted comments on this article, observing that the Birkhofer article addressed a boiling water reactor, which has several core features that tend to make three-dimensional effects more important than in a pressurized water reactor. Among these are the magnitudes of the feedback coefficients, the geometrical design of the control rod, and the reactivity control scheme. Babcock and Wilcox further observes that the rod worth for the case analyzed was 1.67 percent reactivity which is more than a factor of two larger than the technical specification limit for Babcock and Wilcox reactors. It is to be expected that three-dimensional effects would increase in importance rapidly as a function of rod worth. Also, the two-dimensional problem analyzed by Birkhofer had an axial peaking factor of one as opposed to a value of 1.7 used in the Babcock and Wilcox analyses. Application of this factor to the two-dimensional results will bring them into line with those of the three-dimensional calculations.

We agree with the Babcock and Wilcox comments regarding the weaknesses of the Birkhofer article. Until full three-dimensional time-dependent calculations are performed, however, we are unable to ascertain the magnitude of the uncertainty involved in the synthetic three-dimensional treatment performed by Babcock and Wilcox. Meanwhile, we accept the analysis of the rod ejection accident on the basis that the computed maximum enthalpy of the hottest fuel rod is of the order of 180 calories per gram, which is far below our maximum acceptance criterion of 280 calories per gram. It is very unlikely that residual three-dimensional effects could cause the 280 calories per gram value to be exceeded.

15.4.3 Steam and Feedwater Line Breaks

Several cases of steamline breaks were analyzed, including breaks inside containment, outside containment, and upstream and downstream of the main steam isolation valves. For steamline breaks larger than about 1.65 square feet in area, the reactor trips on low reactor coolant pressure. For smaller breaks, the reactor trips on high neutron flux. Steam generator isolation and auxiliary feedwater actuation are automatically initiated. The departure from nucleate boiling ratio, for all main steam line breaks, remained above 1.32, and the reactor coolant pressure did not approach the allowable limit of 110 percent of design pressure. Safety injection is initiated when reactor coolant pressure drops to 1600 pounds per square inch.

Several feedwater line rupture cases were analyzed. For feedwater line break transients, the safety systems will be actuated by high differential pressure between the steam and feedwater pressures. The reactor trips on high reactor coolant pressure. The reactor coolant safety valves actuate during the transient and prevent further rise in reactor coolant pressure. The 110 percent ASME Code pressure limit was not approached. For all cases, the departure from nucleate boiling ratio remained above 1.32, and no fuel damage was predicted. We conclude that the plant design is acceptable in terms of fuel damage and pressure limits, with regard to potential accidents involving steam and feedwater line breaks.

15.4.4 Anticipated Transients Without Scram

A number of plant transients can be affected by a failure of the scram system to function. For a pressurized water reactor, the most important transients include loss of normal feedwater, loss of electrical load, inadvertent control rod withdrawal, and loss of normal electric power.

In September 1973, we issued WASH-1270, "Technical Report on Anticipated Transients Without Scram for Water-Cooled Power Reactors," establishing acceptance criteria for anticipated transients without scram. Babcock & Wilcox analyses for such transients are discussed in BAW-10099, "Babcock & Wilcox Anticipated Transients Without Scram Analysis," Revision 0.

On December 9, 1975, we issued our staff status report, which identified guidelines for further analyses, and, in a staff letter of April 7, 1976, we required Babcock & Wilcox to provide additional analyses by June 30, 1976 and to identify any design changes needed to meet our requirements concerning anticipated transients without scram. Subsequently, Babcock & Wilcox requested a delay for submittal of these analyses. In December 1976, Babcock & Wilcox provided partial analyses.

We are continuing a generic review of this area of concern, and the staff evaluation of the Babcock & Wilcox analyses is expected to be published this year. We will require that any design changes needed, as a result of approved Babcock & Wilcox analyses, shall be incorporated into the design in a timely manner.

15.5 Radiological Consequences of Accidents

The postulated accidents analyzed by the applicant and those analyzed by us, to determine the offsite radiological consequences, are the same as those analyzed for previously licensed pressurized water reactor plants. These accidents include a design basis loss-of-coolant accident, a fuel handling accident, and a rupture of a radioactive gas storage tank. The offsite doses that we calculated for these accidents and the assumptions used in the analyses, are given in the following sections of this report.

On the basis of our experience with the evaluation of the steam line break and the steam generator tube rupture accidents for pressurized water reactor plants of similar design, we have concluded that the consequences of these accidents can be controlled by limiting the permissible radioactivity concentrations in the reactor coolant system and the secondary coolant system. At the operating license stage of review, we will include limits in the technical specifications for the reactor coolant system and secondary coolant system activity concentrations, such that the potential two-hour doses at the exclusion radius, as calculated by us for these accidents, will be small fractions of the guideline values of 10 CFR Part 100. Similarly, we will calculate the consequences of a rod ejection accident, prior to a decision to issue an operating license, and will limit by technical specification the permissible primary to secondary leak rate such that the consequences we calculate are within 10 CFR Part 100 guidelines.

The radioactive waste gas decay tanks will be designed to seismic Category I requirements. Therefore, the total failure of these tanks is sufficiently improbable that 10 CFR Part 100 guideline doses are applicable. Our calculations (see Table 15.2) indicate that doses for failure of these tanks would be well within 10 CFR Part 100 guidelines. Appropriate technical specifications will be placed on the maximum activity that can be stored in any one tank at any time such that single failure of active components, including the lifting or sticking of a safety or relief valve, will not result in radiological consequences that exceed small fractions of 10 CFR Part 100 guideline doses.

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15.5.1 Loss-of-Coolant Accident

The reactors will each be surrounded by a double containment structure that consists of a low leakage steel containment vessel and an outer reinforced concrete shield building. The applicant has specified a design leak rate for the primary containment of 0.5 percent of the containment volume per day for the first day following the loss-of-coolant accident and 0.25 percent per day from one to thirty days following the accident.

Leakage from the containment vessel into the annulus between the containment vessel and the concrete shield building will be collected and discharged to the atmosphere through particulate filters and iodine adsorbers. A solution of sodium hydroxide in water will be sprayed into the containment vessel to remove iodine from the containment. All of these engineered safety features will be included in order to minimize the offsite radiological consequences of design basis accidents.

For our dose evaluation purposes, radioactive material that leaks from the primary containment can take the following pathways to the environment:

- (1) Direct outleakage during the period in which the annulus pressure is positive.
- (2) Leakage to the shield building annulus, thence through the shield building ventilation system to atmosphere.
- (3) Direct through-line bypass leakage, which will not be treated for fission product removal.

In modeling the releases through the shield building annulus pathway, we conservatively assumed direct, unfiltered leakage from the annulus until the shield building ventilation system reduced the pressure in the annulus to less than atmospheric pressure.

The applicant has estimated that direct bypass leakage will be 0.03 percent of containment volume per day. We have used this bypass leakage percentage in our calculations of the loss-of-coolant accident doses.

For control of combustible gases that could be produced as a result of a loss-of-coolant accident, the applicant has proposed a system that would dilute and purge the containment atmosphere. This is unacceptable to us, as discussed in Section 6.2.4 of this report. Hydrogen recombiners are included in all light-water-cooled nuclear power plants for which construction permits have been issued recently. Therefore, our dose calculations are based on the assumption that hydrogen recombiners will be used for combustible gas control.

As part of our evaluation of the loss-of-coolant accident, we have evaluated the consequences of leakage of containment sump water, which is circulated by the emergency core cooling system outside the containment after the postulated accident.

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We have assumed that the sump water contains a mixture of iodine fission products consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident." At the time of the recirculation mode of operation, about 40 minutes after the accident, the sump water will be circulated outside of the containment to the reactor auxiliary building, to be cooled. If a source of leakage should develop, such as from a pump seal, a portion of the iodine could become gaseous and escape to the outside atmosphere. The applicant has estimated a low level leakage rate of about 5700 cubic centimeters per hour from components of the emergency core cooling system. Our calculation of the dose that results from assumed leakage of that amount is small and, when added to the calculated loss-of-coolant accident dose at the low population zone, is still within 10 CFR Part 100 guidelines. The equipment that circulates containment sump water will be located in areas of the auxiliary building from which ventilation will be treated by iodine adsorbers.

The results of our calculations are shown in Table 15.2, and the assumptions used in the analysis are listed in Table 15.3. The doses we calculated for the loss-of-coolant accident are within the 150 rem thyroid guideline dose value given in Revision 2 of Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors."

15.5.2 Fuel Handling Accidents

For the analysis of a fuel handling accident occurring in the fuel handling building, we have assumed that a fuel assembly is dropped in the fuel pool and that all of the fuel rods in the assembly were damaged, thereby releasing the volatile fission product gases from the gap between fuel pellets and cladding. The radioactive material that escapes from the fuel pool is assumed to be released to the environment over a two-hour time period, with the iodine activity being reduced by filtration through the fuel building exhaust system. The dose model and dose conversion are in conformance with 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 calculated dose results are shown in Table 15.2, and the assumptions and parameters used in the analysis are shown in Table 15.4. Calculated doses for this postulated accident are well within the guidelines of 10 CFR Part 100.

We have also independently evaluated the consequences of a postulated fuel handling accident inside containment.

TABLE 15.2

RADIOLOGICAL CONSEQUENCES OF DESIGN BASIS ACCIDENTS

<u>Accident</u>	<u>Exclusion Area</u> <u>2-Hour Dose, Rem</u>		<u>Low Population Zone</u> <u>30-Day Dose, Rem</u>	
	<u>Thyroid</u>	<u>Whole Body</u>	<u>Thyroid</u>	<u>Whole Body</u>
Loss-of-Coolant	112	6	8.2	<1
Fuel Handling				
Spent Fuel Pool	3	<1	<1	<1
Containment Building	9	<1	<1	<1
Gas Decay Tank Failure	-	<1	-	<1

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TABLE 15.3

LOSS-OF-COOLANT ACCIDENT DOSES
ASSUMPTIONS AND INPUT PARAMETERS

Power level	2772 megawatts thermal
Operating time	3 years
Fraction of core inventory available for leakage	
Iodines	25 percent
Noble Gases	100 percent
Initial iodine composition in containment	
Elemental	91 percent
Organic	4 percent
Particulate	5 percent
Primary containment volumes	
Sprayed	2.392×10^6 cubic feet
Unsprayed	4.165×10^5 cubic feet
Primary containment leak rate	
0-24 hours after accident	0.5 percent per day
>24 hours after accident	0.25 percent per day
Direct outleakage (no filtration)	13 minutes
Bypass leakage direct to atmosphere	6 percent of primary containment leakage
Iodine filter efficiencies	
Elemental iodine	95 percent
Organic iodine	95 percent
Particulate iodine	95 percent
Containment spray system effectiveness	
Decontamination factor, elemental iodine	100
Elemental iodine removal coefficient	10 inverse hours
Organic iodine removal coefficient	0 inverse hours
Particulate iodine removal coefficient	0.368 inverse hours
Atmospheric Diffusion Factors (seconds per cubic meter)	
0-2 hours (exclusion area boundary, distance 635 m)	2.1×10^{-4}
0-8 hrs (low population zone, distance 3200 m)	8.2×10^{-6}

TABLE 15.3 (Continued)

8-24 hrs (low population zone)	5.7×10^{-6}
1-4 days (low population zone)	2.6×10^{-6}
4-30 days (low population zone)	8.0×10^{-7}

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TABLE 15.4

FUEL HANDLING ACCIDENT
ASSUMPTIONS AND INPUT PARAMETERS

Power level	2772 thermal megawatts
Number of fuel rods damaged	264
Total number of fuel rods in core	46,728
Power peaking factor of damaged fuel	1.65
Shutdown time	72 hours
Inventory released from damaged rods (iodines and noble gases)	10 percent
Pool decontamination factors	
Iodines	100
Noble gases	1
Iodine fractions released from pool	
Elemental	75 percent
Organic	25 percent
Filter efficiency for iodine removal	95 percent
Atmospheric diffusion factors;	2.1×10^{-4} seconds per cubic meter
(exclusion boundary, 2 hours)	8.2×10^{-6} seconds per cubic meter
(low population zone, 2 hours)	

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The containment ventilation system, which will be in operation during refueling operations, will consist of two fan coolers, having a total capacity of 234,000 cubic feet per minute. The fan coolers will be in the upper portions of the containment. The fan coolers will blow cooled air to the lower portions of the containment, from which one-fifth (about 46,000 cubic feet per minute) will be exhausted past the containment purge isolation valves to the atmosphere. The remainder will be returned to the fan coolers for recirculation. A separate fan cooler will circulate 20,800 cubic feet per minute of cooled air into the steam generator and reactor compartments. Two safety grade radiation monitors in the containment will actuate closure of the purge valves, isolating the containment in 15 seconds.

Because the contaminated air is likely to be significantly diluted and containment isolation will be relatively prompt, we believe the dose consequences are likely to be low. It is possible, however, that the initial release of activity might be directed downward, away from the upper radiation monitors. We have, therefore, conservatively assumed that the entire activity release is directed initially into the lower portions of the containment and that one-fifth is released to the environment before containment isolation can occur. Our other assumptions are given in Table 15.4 and the dose consequences are listed in Table 15.2. The doses are well within the guideline values of 10 CFR Part 100.

Our independent assessment of a postulated fuel handling accident inside containment has conservatively assumed the operation of existing plant systems. We conclude that these systems will effectively mitigate the consequences of such an event, and that the calculated doses are well within the guideline values of 10 CFR Part 100.

15.5.3 Radioactive Liquid Waste Tank Failure

The consequences of component failures, which could result in release to the environs of liquids containing radioactive materials, were evaluated for components located outside the reactor containment. Considered in our evaluation were (1) the radionuclide inventory in each component, assuming a one percent fission product source term; (2) a component liquid inventory equal to 80 percent of its design capacity; (3) the mitigating effects of plant design, including overflow lines and the location of storage tanks in curbed areas designed to retain spillage; and (4) the effects of site geology and hydrology.

The applicant has incorporated provisions in the design to retain releases from liquid overflows, as discussed in Section 11.2.1 of this report. The site is adjacent to Lake Erie. In the event of a spill resulting in radionuclides entering the ground water, the ground water flow will move the spillage toward Lake Erie.

Based on our evaluation, the potential tank failure that would result in the greatest quantity of activity released to the environment, is failure of one of the

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clean waste receiver tanks. The tank is assumed to contain radionuclides at 50 percent of primary coolant activity levels for the design basis fission product inventory stated above. In our evaluation, we have determined the liquid transit time for the leakage to the surface of Lake Erie to be 72 years. See Section 2.4.6 of this report. Considering the leakage transit time, the calculated radionuclide concentrations in Lake Erie would be small fractions of the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, for unrestricted areas. Based on the foregoing evaluation, we conclude that the provisions incorporated in the applicant's design to mitigate the effects of component failures involving contaminated liquids are acceptable.

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16.0 TECHNICAL SPECIFICATIONS

The technical specifications in an operating license define certain features, characteristics, and conditions governing operation of a facility that cannot be changed without prior approval of the Nuclear Regulatory Commission. Final technical specifications will be developed and evaluated at the operating license stage. However, in accordance with Section 50.34 of 10 CFR Part 50, an application for a construction permit is required to include preliminary technical specifications. The regulations require an identification and justification for the selection of those variables, conditions or other items that are determined, as a result of the preliminary safety analysis and evaluation, to be probable subjects of technical specifications for the facility, with special attention given to those that may significantly influence the design.

We have reviewed the proposed technical specifications presented in Section 16.0 of the Preliminary Safety Analysis Report with the objective of identifying those items that would require special attention at the construction permit stage in order to preclude the necessity for any significant change in design to support the final technical specifications. The proposed technical specifications are similar to those being developed or in use for plants of design similar to the proposed facility. We have not identified any items that require special attention at this stage of our review.

On this basis, we have concluded that the proposed technical specifications are acceptable.

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17.0 QUALITY ASSURANCE

17.1 General

The quality assurance (QA) program for Davis-Besse Units 2 and 3 is described in Chapter 17 of the Preliminary Safety Analysis Report. Section 17.1.1 describes the QA program of the Toledo Edison Company (Toledo Edison), the applicant; Section 17.1.2 references the QA program of the Bechtel Power Corporation (Bechtel) responsible for architect/engineering services; Section 17.1.3 references the QA program for the constructor, United Engineers and Constructors (UE&C); and Section 17.1.4 references the QA program for the Babcock & Wilcox Company (Babcock & Wilcox), responsible for designing and supplying the nuclear steam supply system.

The QA program descriptions for Bechtel, UE&C, and Babcock & Wilcox are contained in topical reports, which are incorporated in the Preliminary Safety Analysis Report by reference. Toledo Edison is responsible for the total QA program and is responsible for controlling and verifying the QA programs of its principal contractors (Bechtel, UE&C, and B&W).

17.2 Toledo Edison

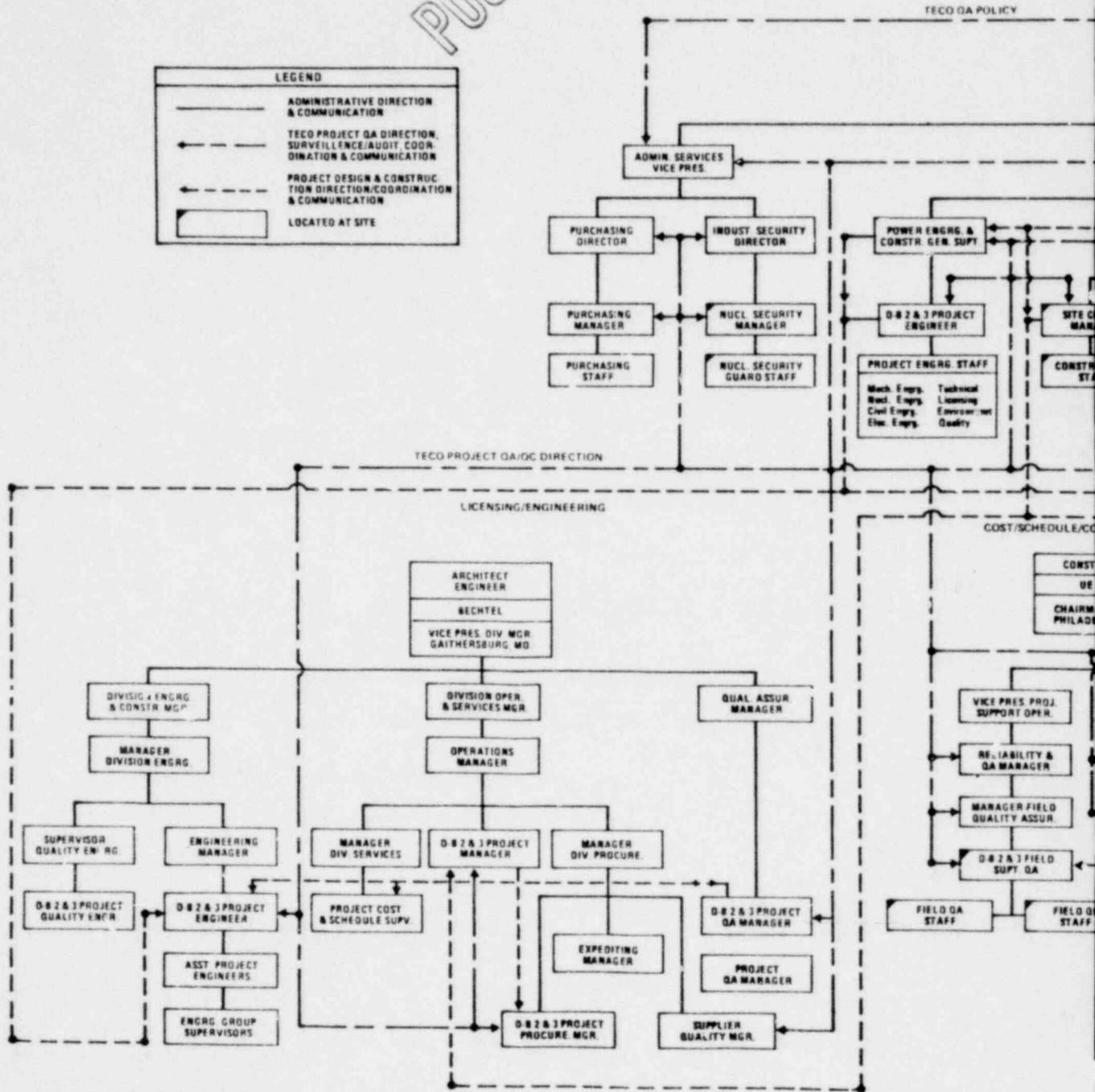
17.2.1 Organization

The Toledo Edison corporate organization is shown in Figure 17.1. The overall project organization, involving Bechtel, UE&C, Toledo Edison, and Babcock & Wilcox, is also shown in Figure 17.1. The Toledo Edison Vice President Facilities Development is responsible, under the Executive Vice President Operations, for the QA activities of Davis-Besse Units 2 and 3, including the establishment of QA policies, goals, and objectives. Reporting directly to the Vice President Facilities Development is the Quality Assurance Director. Figure 17.1 shows the Quality Assurance Director to be free of prime responsibility for schedule and cost and to be on at least the same organizational level as those whose work he verifies. We find, with this corporate structure, that the QA organization has adequate independence and reports at a sufficiently high management level to accomplish its objectives.

The Quality Assurance Director directs and executes the QA program described in the Preliminary Safety Analysis Report. He is responsible for developing the QA program and for monitoring its implementation and effectiveness. The QA program, described in the QA Manual, is approved by the Vice President Facilities Development, Executive Vice President Operations, and the President. The program is implemented through QA procedures, instructions, standards, specifications, and forms that provide the details of how each of the 10 CFR Part 50 Appendix B criteria is met. Within

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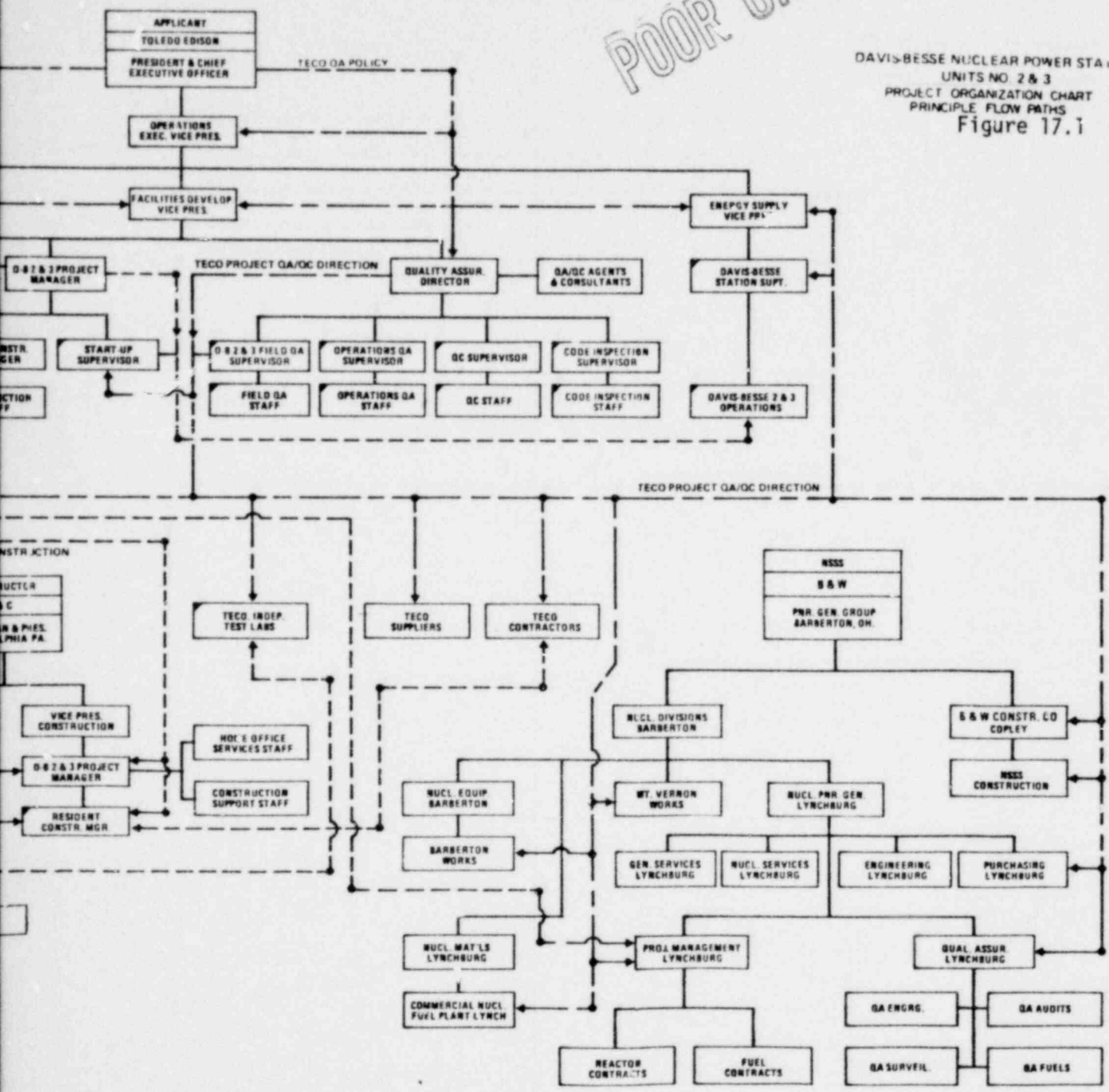
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DAVIS-BESSE NUCLEAR POWER STATION
 UNITS NO. 2 & 3
 PROJECT ORGANIZATION CHART
 PRINCIPLE FLOW PATHS
 Figure 17.1



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Toledo Edison, an internal policy statement issued by the President requires that all personnel involved in or responsible for QA activities shall comply with the requirements of the Toledo Edison QA manual. We find that Toledo Edison has clearly defined the responsibilities and authorities of its QA organization.

Toledo Edison implements its QA functions by means of a home office QA organization and a site QA organization, both under the QA Director. The home office QA organization is directly supervised by the QA Director. The site organization is managed by a Field QA Supervisor, during design, procurement, and construction, and by the Operations QA Supervisor during plant operation. A Quality Control Supervisor and Code Inspection Supervisor also report directly to the QA Director. These Supervisors and their staff are involved in day-to-day inspection activities. The QA organization has the authority to identify quality problems, to recommend or provide solutions through designated channels, and to verify implementation of solutions. The QA Director has the authority, which he may delegate to others in writing, to stop work and to control further processing, delivery, installation, or use of nonconforming items or services until deficiencies have been properly corrected and verified.

To assess the effectiveness of the QA program, the Vice President Facilities Development has been assigned the responsibility for assuring that management reviews are conducted by personnel outside of the QA organization. These reviews assess the adequacy of scope, implementation, and effectiveness of the QA program.

We find that Toledo Edison's description for implementing their QA program, with authority to enforce QA requirements, with authority to stop work, and with corporate level management involvement, is acceptable.

Our evaluation of the Toledo Edison QA organization is that it is free of prime responsibility for schedule or cost; it is independent of the organizations whose work it verifies; it has clearly defined authorities and responsibilities; it is so organized that it can identify quality problems in the other organizations that perform quality related work; it can initiate, recommend, or provide solutions; it can verify implementation of solutions; and it can prevent further processing, shipment, installation or utilization of nonconforming items until proper dispositioning has occurred.

We, therefore, conclude that the Toledo Edison QA organization complies with 10 CFR Part 50, Appendix B, and is acceptable.

17.2.2 Program

Chapter 17 of the Preliminary Safety Analysis Report provides a cross reference that identifies the sections of the Toledo Edison Nuclear Quality Assurance Manual and the Toledo Edison Quality Assurance Procedures that comply with and implement

each of the criteria of 10 CFR Part 50, Appendix B. This QA manual, and the QA procedures that have been written to implement it, are used to coordinate the QA activities of the various organizations within Toledo Edison that are responsible for design, procurement, construction, testing, operations, and services. Based on our review of this cross reference tabulation, we conclude that each criterion of Appendix B to 10 CFR Part 50 has been addressed within Toledo Edison's documented QA procedures and requirements.

Toledo Edison has committed to comply with the Regulatory positions of the applicable Regulatory Guides and with the ANSI Standards listed in Table 17.1. TECO has also committed that its principal contractors (Bechtel, B&W and UE&C) will implement their QA program descriptions set forth in topical reports that we have found acceptable. We find that, with this commitment and our review of Toledo Edison's QA policies and QA program description, Toledo Edison has defined an acceptable QA program.

The structures, systems, and components comprising the safety items subject to the QA program have been identified in the Preliminary Safety Analysis Report.

Toledo Edison will assure that its principal contractors and subcontractors have adequate QA programs, that inspections will be performed by qualified personnel using documented inspection instructions, and that results will be recorded. Toledo Edison will assure, by surveillance and audits, that personnel performing inspections are free from undue cost and schedule pressures of the project.

Toledo Edison has established program requirements, on itself and on its contractors, that assure there will be a documented system of records attesting to quality.

Toledo Edison has developed a detailed indoctrination and training program to assure that personnel who perform quality-related activities are trained and qualified in the principles and techniques of the assigned activities and are instructed as to the purpose, scope, and implementation of quality-related manuals and procedures.

A system of planned and documented audits, described in the Preliminary Safety Analysis Report, will be used by Toledo Edison to verify compliance with all aspects of the QA program and to assess the program effectiveness. Audit results are documented, and are reported to appropriate levels of management for corrective action. Response to audit findings are verified, for implementation and effectiveness, by follow-up audits. We find that Toledo Edison has described a satisfactory QA audit program.

In our review, we have evaluated Toledo Edison's QA program for compliance with the Commission's regulations and applicable regulatory guides and industry standards. Based on this review, we conclude that the Toledo Edison QA program contains the

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TABLE 17.1
REGULATORY GUIDANCE APPLICABLE TO
QUALITY ASSURANCE PROGRAMS

1. Regulatory Guide 1.28, "Quality Assurance Program Requirements (Design and Construction)," June 1972.
2. Regulatory Guide 1.30, "Quality Assurance Requirements for the Installation, Inspection and Testing of Instrumentation and Electric Equipment," August 1972.
3. Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," March 1973.
4. Regulatory Guide 1.38, "Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage and Handling of Items for Water-Cooled Nuclear Power Plants," March 1973.
5. Regulatory Guide 1.39, "Housekeeping Requirements for Water-Cooled Nuclear Power Plants," March 1973.
6. Regulatory Guide 1.58, "Qualification of Nuclear Power Plant Inspection, Examination and Testing Personnel," August 1973.
7. Regulatory Guide 1.64, "Quality Assurance Requirements for the Design of Nuclear Power Plants," October 1973.
8. Regulatory Guide 1.74, "Quality Assurance Terms and Definitions," February 1974.
9. ANSI N45.2.5, "Supplementary Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants" (Draft 3, Rev. 1, January 1974).
10. ANSI N45.2.8, "Quality Assurance Requirements for Installation, Inspection and Testing of Mechanical Equipment and Systems" (Draft 3, Rev. 3, April 1974).
11. ANSI N45.2.9, "Requirements for Collection, Storage, and Maintenance of Quality Assurance Records for Nuclear Power Plants," (Draft 15, Rev. 0, April 1974, including Regulatory staff comments and supplementary guidance in Section D of WASH-1283 (Gray Book), Rev. 1, May 1974).
12. ANSI Standard N45.2.12, "Requirements for Auditing of Quality Assurance Programs for Nuclear Power Plants," Draft 3, Revision 4, February 1974.
13. ANSI N45.2.13, Draft 2, Rev. 4, April 1974, "Quality Assurance Requirements for Procurement of Items and Services for Nuclear Power Plants."

necessary QA provisions, requirements, and controls for compliance with Appendix B to 10 CFR Part 50 and applicable guides and standards, and is acceptable for the design, procurement, and construction of Davis-Besse Units 2 and 3.

17.3 Bechtel QA Program

Bechtel has been designated as the architect/engineer, and is responsible for the design of the balance-of-plant structures, systems and components, i.e., those structures, systems and components of the nuclear power plant not included in the nuclear steam supply system.

Figure 17.1 shows the Bechtel organization responsible for the project as it relates to engineering and quality assurance. The Project Manager for the Davis-Besse project reports through the Operations Manager to the Division Operations and Services Manager, who reports to the Vice President and Division Manager of the Gaithersburg Power Division. The QA manager reports directly to the Vice President and Division Manager, as do the Managers of Division Operations and Services and Division Engineering and Construction. We find the independence of the QA organization acceptable.

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

The Vice President and Division Manager of the Gaithersburg Power Division is responsible for the Bechtel QA program as it applies to Davis-Besse Units 2 and 3. The QA Manager is responsible for executing the QA program, for approving QA procedures and instructions for the QA organization, and for reviewing and concurring in, prior to issuance of, quality assurance provisions contained in quality control procedures and instructions that have been prepared by other departments in the division.

The responsibility for the procurement of items and services for the design and construction of Davis-Besse Units 2 and 3, except for the NSSS, rests with the applicant. Bechtel, as architect-engineer, is responsible for preparing the technical portions of procurement documents. For those cases where Bechtel is contractually involved in procurement, procurement inspection is conducted by the inspection organization in Bechtel procurement. The inspection procedures and instructions used by the procurement inspection organization are subject to review and approval by the QA organization. We find that this check by the QA organization provides adequate assurance of independence from undue pressures of cost and schedules for the procurement inspectors.

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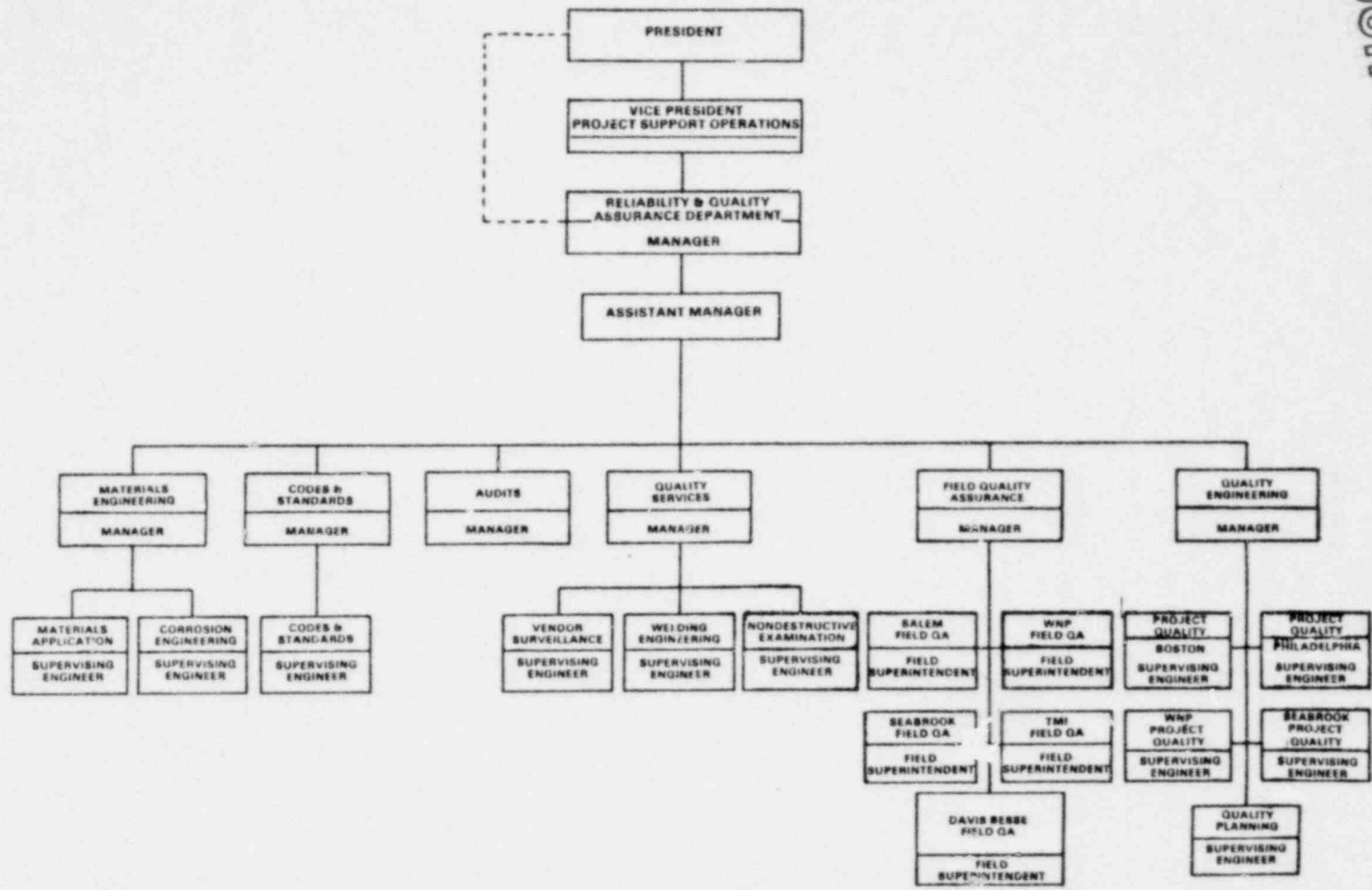


Figure 17.2 Quality Assurance Plan of Organization – United Engineers & Constructors, Inc.

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The applicant and Bechtel have committed that Bechtel's scope of activities for this project will be performed under the QA provisions described in Bechtel's Topical Report BQ-TOP-1, "Quality Assurance Program for Nuclear Power Plants," Revision 2A, and will also conform to unique programmatic requirements imposed by TECO. We find, with this commitment and our review of Bechtel's QA policies and QA program description, that Bechtel has defined an acceptable QA program.

The QA policies, procedures, and instructions for the Bechtel QA program are documented in the QA manual, the procurement inspection department manual, the engineering procedures manual, the quality control manual for ASME nuclear components, and the construction procedures manual. Bechtel has provided a cross index of Bechtel's QA procedures and the related criteria of Appendix B to 10 CFR Part 50. Based on our review of this information, we conclude that implementation of each criterion of Appendix B to 10 CFR Part 50 has been included within Bechtel's documented QA policies, procedures, and instructions for the project.

Bechtel has described a training and indoctrination program for its personnel. This program covers indoctrination and training in standards, policies, and procedures covering specific areas of work; qualification of inspection, examination and testing personnel; indoctrination in procurement inspection requirements; training and qualification of audit personnel; and qualification of personnel to code requirements for pressure boundary and structure welding and nondestructive tests. We find the program acceptable.

Design documents are prepared by Project Engineering personnel and are verified or checked in accordance with engineering procedures. These checks are performed by personnel other than those who performed the original design but who have adequate technical capabilities for checking the work. We find the Bechtel description for design control adequate.

The Preliminary Safety Analysis Report describes a comprehensive program of Bechtel audits that cover the various activities of the QA program. The planned audit activities include project engineering, procurement, field construction, suppliers and subcontractors, and site activities, such as project engineering, design, procurement, construction, and quality control. Management reviews of the status and adequacy of the QA program are accomplished, by management outside of the QA organization, through review of audit reports and periodic reports of the Division QA Manager. Also, the program is reviewed annually by individuals outside the QA organization.

In our review, we have evaluated the Bechtel QA program for compliance with the Commission's regulations and applicable regulatory guides and industry standards. Based on this review, we conclude that the Bechtel QA program contains the necessary QA provisions, requirements, and controls for compliance with Appendix B to 10 CFR Part 50 and applicable guides and standards, and is acceptable.

United Engineers & Constructors

The applicant has contracted, by letter of intent, with United Engineers and Constructors, Inc. (UE&C) as constructor for Davis-Besse Units 2 and 3. UE&C will build the units and, in selected situations where UE&C does not perform the work, will act as construction manager and will administer contracts for construction and for related work performed by subcontractors. In addition, the applicant will conduct QA/QC inspection surveillance of work performed by UE&C. UE&C will also furnish assistance to the applicant in verifying the fulfillment of all contracts.

Figure 17.1 shows the UE&C organizational arrangements for the projects. The responsibility for quality assurance within UE&C rests with the Vice President of Support Operations, who is on the same organizational level as the Vice President of Construction, thus assuring organizational independence.

The Manager of Reliability and Quality Assurance reports directly to the Vice President of Support Operations, and is delegated the full authority and responsibility for UE&C's QA program. He has the authority to control further processing or delivery of nonconforming material.

The organization for the Manager of Reliability and Quality Assurance (R&QA) is summarized in Figure 17.1, and is provided in greater detail on Figure 17.2. As seen from Figure 17.2, UE&C has a Manager of Field Quality Assurance who reports directly to the Manager of R&QA. The Manager of Field QA is responsible for QA activities at those job sites for which UE&C's scope of work includes construction management or construction. He directly supervises the UE&C Field Superintendent of QA assigned to the site. The Field Superintendent of QA directly supervises the field QA staff and field QC staff, as shown in Figure 17.2. The field QA staff and field QC staff are independent of the UE&C construction forces and of their management (Figure 17.2), and include engineers, inspectors, and technicians to provide a combination of expertise in all phases of construction and quality assurance, including quality control. They are responsible for implementing all QA/QC activities at the site, including construction inspection.

As shown in Figure 17.2, other major organizational functions within the R&QA department are Quality Engineering, Materials Engineering, Codes and Standards Quality Services, and Audits.

Based on our review of the description of the corporate organizations in the Preliminary Safety Analysis Report, we find the UE&C has established and described a corporate organization that is capable of developing and implementing a QA program in compliance with Appendix B to 10 CFR Part 50. In addition, this organization is structured such that personnel performing QA functions in the UE&C organization have sufficient authority and organizational freedom to perform their functions effectively and without reservation.

The applicant and UE&C have stated in the application that UE&C's construction services will be performed under the QA program described in UE&C's Topical Report UEC-TR-001-4A, "Quality Assurance Program." Quality-related construction activities will be governed by the UE&C Davis-Besse Project Quality Assurance and Quality Control Manual. Those activities related to ASME Section III, Division 1, components will be governed by the UE&C Nuclear Quality Assurance Manual.

UE&C has committed, in its QA topical report, that its QA program conforms to the requirements and guidance of the regulatory guides and ANSI Standards listed in Revision 4A of its QA topical report. The UE&C QA topical report restates a UE&C corporate policy statement, signed by the President of UE&C, that makes a clear commitment that UE&C work, performed on the safety-related portions of nuclear power plant projects, will be accomplished in accordance with the applicable requirements of Appendix B to 10 CFR Part 50, of ANSI N45.2, and of the appropriate ANSI daughter standards. Any exceptions, alternates, or clarifications to the regulatory guides and ANSI QA Standards are documented in, or incorporated by reference into, the applicant's Safety Analysis Report. The applicant has specified, in the Safety Analysis Report, the specific revision number of the regulatory guides and the specific ANSI QA Standards to be implemented by UE&C on this project. A matrix has been provided in the UE&C topical report to show the principal procedures that implement the requirements of Appendix B to 10 CFR Part 50. A brief description of the procedures has been provided. Based on our review of this information, we conclude that each requirement of Appendix B to 10 CFR Part 50 has been specifically included in written procedures in the UE&C QA program.

UE&C has described a system of planned and documented audits, with provisions for corrective and followup actions. Audits are performed, in accordance with written procedures or checklists, by appropriately trained personnel who have no direct responsibilities in the area audited. Audits are scheduled on the basis of importance, but will be performed at least annually or once during the life of the activity, whichever is shorter.

The UE&C Vice President of Project Support Operations arranges for an annual corporate audit of the activities of the UE&C Reliability and Quality Assurance Department for conformance to provisions of the UE&C Nuclear Quality Assurance Manual.

The staff finds UE&C's description of their audit activities, including the corporate audit of the QA program, acceptable.

Based on our review of the description of the QA program contained in Section 17 of the Preliminary Safety Analysis Report, as amended through Revision 20, we conclude that the UE&C QA program complies with the requirements of Appendix B to 10 CFR Part 50 and is acceptable for the scope of UE&C's responsibilities for the construction phase of the project.

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Babcock & Wilcox QA Program

Babcock & Wilcox is responsible for providing the nuclear steam supply system. Figure 17.3 shows the Babcock & Wilcox corporate organization. The major suppliers of commercial nuclear material, services, and equipment are shown in the double-boxed organizations. Figure 17.4 shows the Nuclear Power Generation Division. This Division is responsible for design and procurement and for assuring the quality of all of the commercial nuclear material, services and equipment furnished by Babcock & Wilcox. The Division imposes the requirements of Appendix B of 10 CFR Part 50 on Babcock & Wilcox's manufacturing organizations, by engineering and manufacturing instructions, and on B&W suppliers by procurement documents.

Each of the organizations within the Babcock & Wilcox corporation that is concerned with manufacturing and fabrication, has an organization responsible for quality assurance/quality control and for providing technical, administrative, and functional direction for its quality assurance program. These organizations report to a management level that will assure independence consistent with Criterion I of 10 CFR Part 50, Appendix B.

The Nuclear Power Generation Division QA organization has authority to stop work, through the issuance of restraint orders, and has the freedom to (1) identify quality problems; (2) initiate, recommend, or provide solutions; and (3) verify implementation of solutions.

Two separate and independent engineering organizations control design, design change and design review. The design requirement organizations (Integration) establishes the design requirements and performs the design reviews. The design organization (Task Engineering), using the design requirements established by Integration, develops and documents the designs. In addition, a design review board, that includes a QA representative and that has no other design responsibilities, approves or disapproves all new product designs, new processes, or major design changes. Designs are reviewed in accordance with applicable requirements.

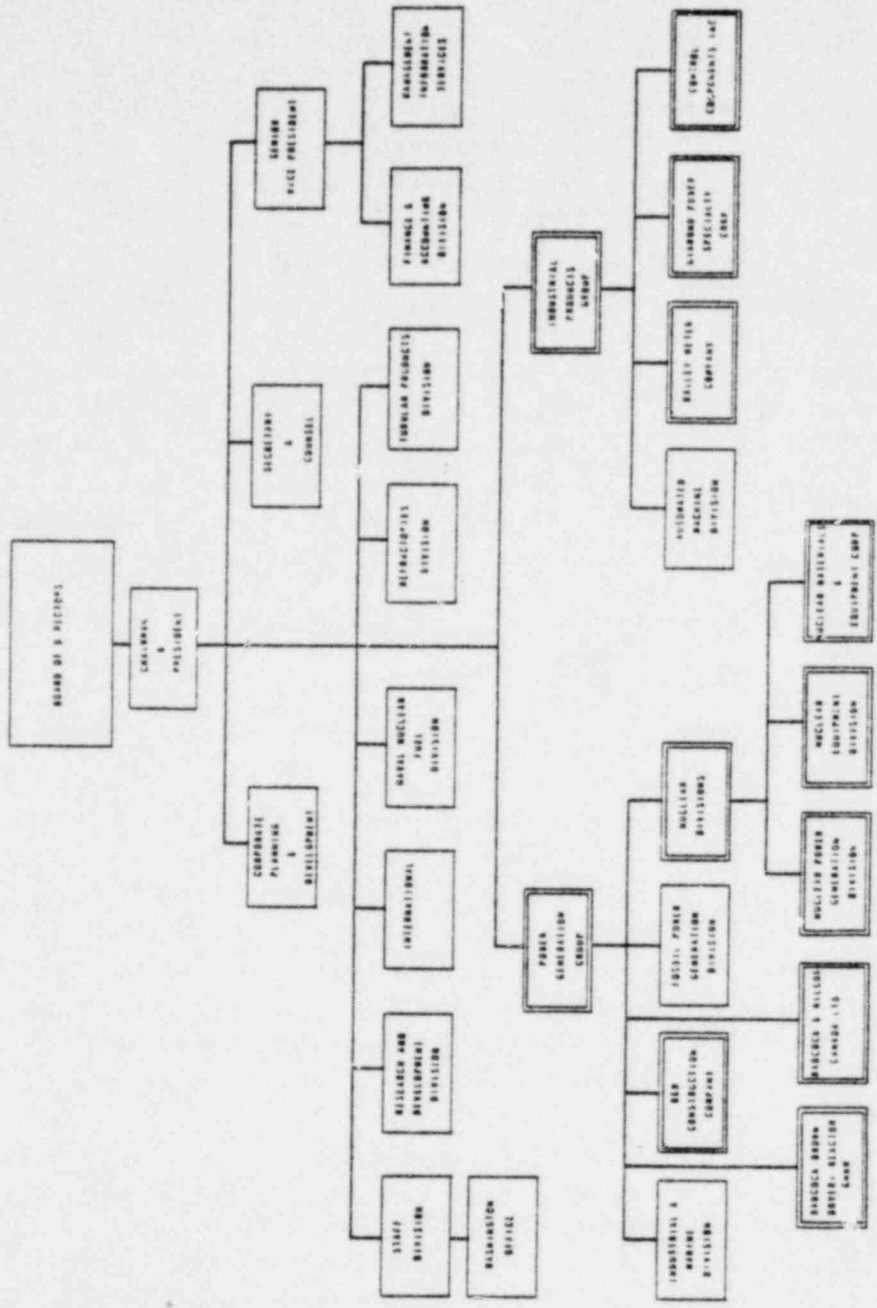
To provide control of purchased safety-related structures, systems, and components, each prospective supplier's QA program must be approved by the Nuclear Power Generation Division QA. QA engineers review purchase requisitions, purchase orders and subsequent change notices. The Division QA is responsible for incorporating the QA requirements into the procurement package, including QA requirements on subvendors. The Division QA reviews and retains supplier documentation that demonstrates acceptable quality. Audits and feedback of nonconformance data are used by QA engineers to measure supplier performance.

The Central Quality Assurance organization,* reporting through a Planning and Technology organization* to the Group Vice President of the Power Generation Group

*Not shown on charts.

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Figure 17.3 The B&W Corporate Organization



Babcock & Wilcox

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(outside of the Division QA organization), regularly assesses the scope, implementation and effectiveness of the QA program.

The QA program applies to all safety-related structures, systems, and components within the Babcock & Wilcox scope of work. The applicant and Babcock & Wilcox have committed that the Babcock & Wilcox scope of activities will be performed in accordance with the QA provisions described in the Babcock & Wilcox Topical Report BAW-10096, Revision 3A, "Quality Assurance Program for Nuclear Equipment." We have reviewed this Topical Report and have found it to be acceptable.

The QA program is defined by the Manager QA and is approved by the Vice President of the Nuclear Power Generating Division and the Vice President of Nuclear Divisions. The President's office has issued a policy and management guide that makes it mandatory that all Babcock & Wilcox divisions comply with Nuclear Power Generation Division quality assurance requirements for safety-related structures, systems and components.

A matrix, that relates the Nuclear Power Generating Division QA procedures to the applicable QA criteria of 10 CFR Part 50, Appendix B, is given in the Preliminary Safety Analysis Report. Based on our review of this matrix, we conclude that each criterion has been specifically included in written procedures within Babcock & Wilcox's QA program.

Babcock & Wilcox executes a comprehensive audit program that provides cognizant Babcock & Wilcox management with information on the effectiveness of the QA program. Babcock & Wilcox audits activities affecting quality at Babcock & Wilcox and supplier facilities. Audit areas include all quality related procedures and operations. Trained personnel, not having direct responsibilities in the area being audited, conduct the QA audits in accordance with defined procedures and checklists.

In this review, we have evaluated the Babcock & Wilcox QA program for compliance with Commission regulations, applicable regulatory guides and industry standards. Based on our review, we conclude that Babcock & Wilcox has described a QA program that contains the necessary QA provisions, requirements and controls for compliance with Appendix B to 10 CFR Part 50 and applicable guides and standards, and that this program is acceptable for the nuclear steam supply system for Davis-Besse Units 2 and 3.

17.6 Implementation of the Quality Assurance Program

The Office of Inspection and Enforcement has conducted inspections to examine the implementation of the Davis-Besse Units 2 and 3 QA program to ascertain its conformance with related 10 CFR Part 50, Appendix B, Preliminary Safety Analysis Report commitments. The examination encompassed the organizations of the applicant and, on a generic basis, the applicant's major contractors. These examinations focused

on quality assurance activities related to the design, procurement, and construction of the nuclear power plants and, for each organization examined, included a review of established procedures and instructions and the execution of provisions contained therein.

Based upon the results of these inspections, the Office of Inspection and Enforcement has determined that there are two areas, relating to the implementation of the QA program, for which documentation must be provided for inspection. These two areas are as follows:

- (1) The applicant has issued a letter of intent to United Engineers and Constructors as constructor for the plant but has not yet awarded a contract. Thus the QA Program implementation activities included in the constructor's scope of work could not be assessed.
- (2) Schedules have been established for development and implementation of quality assurance or quality control procedures and instructions, including those required for (a) site engineering design control, (b) site-initiated procurement control, and (c) construction. However, administrative or construction procedures have not been prepared and thus have not been implemented.

When the Office of Inspection and Enforcement finds that the unresolved issues have been resolved, we will be able to conclude that the applicant's QA program and its implementation are satisfactory for issuance of construction permits. We will report the findings of the Office of Inspection and Enforcement in a supplement to this report.

17.7 Conclusion

Our review of the applicant's quality assurance (QA) program description for the design and construction phase has been performed to evaluate and verify whether all applicable elements of Appendix B to 10 CFR Part 50 are included in the QA program requirements.

This review has established that the QA organizations are structured such that they can effectively carry out their responsibilities related to quality without undue influence from other groups.

Based on our detailed review and evaluation of the QA program description contained in the Preliminary Safety Analysis Report we conclude that:

- (1) The QA organizations within TECo, Bechtel, Babcock & Wilcox, and UE&C are provided (1) sufficient independence from cost and schedule, when opposed to safety considerations, (2) authority to carry out the QA programs, and (3)

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sufficient access to management at a level necessary to perform their QA functions.

- (2) The program describes adequate QA requirements and controls to satisfy the criteria of Appendix B to 10 CFR Part 50 with the exception of implementation of certain portions of the QA program, discussed in Section 17.6 of this report, that must be approved by the Office of Inspection and Enforcement. We will report the findings of the Office of Inspection and Enforcement in a supplement to this report.

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18.0 REVIEW BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The application for the proposed facility is being reviewed by the Advisory Committee on Reactor Safeguards. After the Committee has reported to the Commission the results of its review, we intend to issue a supplement to this Safety Evaluation Report. The supplement will address the significant comments made by the Committee and will describe the steps taken by the staff to resolve any issues raised as a result of the Committee's review. A copy of the Committee's report will be included.

19.0 COMMON DEFENSE AND SECURITY

The applicant states that the activities to be conducted will be within the jurisdiction of the United States and that all of the directors and principal officers of the applicant are citizens of the United States.

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

20.0 FINANCIAL QUALIFICATIONS

The Commission's regulations, that relate to financial data and information that is required to establish financial qualifications for an applicant for a facility construction permit, are Section 50.33(f) of 10 CFR Part 50 and Appendix C to 10 CFR Part 50. To assure that we have the latest information to determine the financial qualifications of an applicant, it is our current practice to review this information during the later stages of our review of an application. We are continuing our review of the financial qualifications of the applicant and will report the results of our evaluation in a supplement to this report.

21.0 CONCLUSIONS

Based on our analysis of the proposed design of Davis-Besse Nuclear Power Station Units 2 and 3 and upon favorable resolution of the outstanding matters set forth in Section 1.8 and discussed in appropriate sections of this report, we will be able to conclude that, in accordance with the provisions of Sections 50.35(a) and 50.40 of 10 CFR Part 50:

- (1) The applicant has described the proposed design of the facility including, but not limited to the principal architectural and engineering criteria for the design, and has identified the major features or components incorporated therein for the protection of the health and safety of the public;
- (2) Such further technical or design information, as may be required to complete the safety analysis and that can be reasonably left for later consideration, will be supplied in the Final Safety Analysis Report;
- (3) Safety features or components that require research and development have been described and identified by the applicant, and there will be conducted research and development programs reasonably designed to resolve safety questions associated with such features or components;
- (4) On the basis of the foregoing, there is reasonable assurance that (a) such safety questions will be satisfactorily resolved at or before the latest date stated in the application for completion of construction of the proposed facility, and (b) taking into consideration the site criteria contained in 10 CFR Part 100, the proposed facilities can be constructed and operated at the proposed location without undue risk to the health and safety of the public;
- (5) The applicant is technically qualified to design and construct the proposed facility;
- (6) The applicant has reasonably estimated the costs and is financially qualified to design and construct the proposed facility; and
- (7) The issuance of a permit for construction of the facility will not be inimical to the common defense and security or to the health and safety of the public.

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APPENDIX A

TABLE 1

Calculated Releases of Radioactive Material in
Gaseous Effluents from Davis-Besse Nuclear Station, Unit No. 1
(Curies per year per unit)

Radionuclide	Decay Tanks	Building Ventilation			Air Ejector Off-Gas	Total
		Reactor	Auxiliary	Turbine		
Kr-83m	a	a	a	a	a	a
Kr-85m	a	1	2	a	1	4
Kr-85	350	46	2	a	a	400
Kr-87	a	a	1	a	a	1
Kr-88	a	2	4	a	3	9
Kr-89	a	a	a	a	a	a
Xe-131m	5	37	2	a	1	45
Xe-133m	a	32	4	a	3	39
Xe-133	9	4700	320	a	200	5200
Xe-135m	a	a	a	a	a	a
Xe-135	a	9	7	a	4	20
Xe-137	a	a	a	a	a	a
Xe-138	a	a	a	a	a	a
I-131	a	1.3(-1) ^b	5.4(-2)	1.1(-3)	3.4(-2)	2.2(-1)
I-133	a	2.8(-2)	7(-2)	1.4(-3)	4.4(-2)	1.4(-1)
Mn-54	4.5(-5)	2.2(-4)	1.8(-4)	c	c	4.4(-4)
Fe-59	1.5(-5)	7.5(-5)	6(-5)	c	c	1.5(-4)
Co-58	1.5(-4)	7.5(-4)	6(-4)	c	c	1.5(-3)
Co-60	7(-5)	3.4(-4)	2.7(-4)	c	c	6.8(-4)
Sr-89	3.3(-6)	1.7(-5)	1.3(-5)	c	c	3.3(-5)
Sr-90	6(-7)	3(-6)	2.4(-6)	c	c	6(-6)
Cs-134	4.5(-5)	2.2(-4)	1.8(-4)	c	c	4.4(-4)
Cs-137	7.5(-5)	3.8(-4)	3(-4)	c	c	7.5(-4)
C-14	7	1	a	a	a	8
Ar-41	a	25	a	a	a	25
H-3	c	280	280	c	c	560

a = less than 1.0 Curie per year for noble gases and carbon-14, less than 10^{-4} Curie per year for iodine

b = exponential notation; $1.0(-4) = 1.0 \times 10^{-4}$

c = less than 1% of total for this nuclide

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TABLE 2

Calculated Releases of Radioactive Materials
in Liquid Effluents from Davis-Besse, Unit No. 1

<u>Nuclide</u>	<u>Curies per year per unit</u>
Corrosion & Activation Products	
Cr-51	2.2(-4) ^a
Mn-54	1(-3)
Fe-55	2.2(-4)
Fe-59	1.2(-4)
Co-58	6.1(-3)
Co-60	9.0(-3)
Zr-95	1.4(-3)
Nb-95	2(-3)
Np-239	6(-5)
Fission Products	
Br-83	3(-5)
Rb-86	2(-5)
Sr-89	5(-5)
Sr-91	1(-5)
Mo-99	3.1(-2)
Tc-99m	2.1(-2)
Ru-103	1.5(-4)
Ru-106	2.4(-3)
Ag-110m	4.4(-4)
Te-127m	3(-5)
Te-127	5(-5)
Te-129m	1.7(-4)
Te-129	1.1(-4)
I-130	1.3(-4)
Te-131m	6(-5)
Te-131	1(-5)
I-131	6.5(-2)
Te-132	1.4(-3)
I-132	2.5(-3)
I-133	3.6(-2)
I-134	1(-5)
Cs-134	2(-2)
I-135	6.3(-3)
Cs-136	2.4(-3)
Cs-137	2.9(-2)
Ba-137m	5(-3)
Ba-140	2(-5)
La-140	2(-5)
Ce-144	5.2(-3)
All Others	6(-5)
Total except Tritium	0.25
Tritium	550

a = exponential notation; 1.0(-4) = 1.0×10^{-4}

b = nuclides whose release rates are less than 10^{-5} Curies per year are not listed individually but are included in the category "All Others."

TABLE 3

Calculated Releases of Radioactive Material in
Gaseous Effluents from Davis-Besse Nuclear Station, Units Nos. 2 & 3
(Curies per year per unit)

Radionuclide	Decay Tanks	Building Ventilation			Air Ejector Off-Gas	Total
		Reactor	Auxiliary	Turbine		
Kr-83m	a	a	a	a	a	a
Kr-85m	a	1	2	a	1	4
Kr-85	370	51	2	a	1	420
Kr-87	a	a	1	a	a	1
Kr-88	a	2	4	a	2	8
Kr-89	a	a	a	a	a	a
Xe-131m	5	39	2	a	1	47
Xe-133m	a	33	4	a	3	40
Xe-133	9	4900	310	a	190	5400
Xe-135m	a	a	a	a	a	a
Xe-135	a	10	7	a	4	21
Xe-137	a	a	a	a	a	a
Xe-138	a	a	a	a	a	a
I-131	a	3.7(-2) ^b	5.7(-2)	1.1(-3)	3.6(-2)	1.3(-1)
I-133	a	7(-3)	7(-2)	1.4(-3)	4.4(-2)	1.1(-1)
Mn-54	4.5(-5)	2.2(-4)	1.8(-4)	c	c	4.4(-4)
Fe-59	1.5(-5)	7.5(-5)	6(-5)	c	c	1.5(-4)
Co-58	1.5(-4)	7.5(-4)	6(-4)	c	c	1.5(-3)
Co-60	7(-5)	3.4(-4)	2.7(-4)	c	c	6.8(-4)
Sr-89	3.3(-6)	1.7(-5)	1.3(-5)	c	c	3.3(-5)
Sr-90	6(-7)	3(-6)	2.4(-6)	c	c	6(-6)
Cs-134	4.5(-5)	2.2(-4)	1.8(-4)	c	c	4.4(-4)
Cs-137	7.5(-5)	3.8(-4)	3(-4)	c	c	7.5(-4)
C-14	7	1	a	a	a	8
Ar-41	a	25	a	a	a	25
H-3	c	290	290	c	c	580

a = less than 1.0 Curie per year for noble gases and carbon-14, less than 10^{-4} Curies per year for iodine

b = exponential notations; $1.0(-4) = 1.0 \times 10^{-4}$

c = less than 1% of total for this nuclide

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TABLE 4

Calculated Releases of Radioactive Materials
in Liquid Effluents from Davis-Besse, Unit Nos. 2 & 3

<u>Nuclide</u>	<u>Curies per year per unit</u>
Corrosion & Activation Products	
Cr-51	1.3(-4) ^a
Mn-54	1(-3)
Fe-55	1.2(-4)
Fe-59	7(-5)
Co-58	5.2(-3)
Co-60	8.8(-3)
Zr-95	1.4(-3)
Nb-95	2(-3)
Np-239	6(-5)
Fission Products	
Br-83	3(-5)
Rb-86	2(-5)
Sr-89	3(-5)
Sr-91	1(-5)
Mo-99	3.1(-2)
Tc-99m	2.1(-2)
Ru-103	1.4(-4)
Ru-106	2.4(-3)
Ag-110m	4.4(-4)
Te-127m	2(-5)
Te-127	3(-5)
Te-129m	1(-4)
Te-129	6(-5)
I-130	1.2(-4)
Te-131m	5(-5)
Te-131	1(-5)
I-131	6.8(-2)
Te-132	1.2(-3)
I-132	2.2(-3)
I-133	3.5(-2)
I-134	1(-5)
Cs-134	2(-2)
I-135	5.9(-3)
Cs-136	2.3(-3)
Cs-137	2.9(-2)
Ba-137m	4.7(-3)
Ba-140	1(-5)
La-140	1(-5)
Ce-144	5.2(-3)
All Others	5(-5)
Total except Tritium	0.25
Tritium	560

a = exponential notation; $1.0(-4) = 1.0 \times 10^{-4}$

b = nuclides whose release rates are less than 10^{-5} Curies per year are not listed individually but are included in the category "All Others."

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APPENDIX B

BIBLIOGRAPHY

NOTE: Documents that we have referenced in or used to prepare this Safety Evaluation Report, excluding those listed in the Preliminary Safety Analysis Report, may be obtained at the source stated in this Bibliography or, where no specific source is given, at most major public libraries. Correspondence between the Commission and the applicant (Preliminary Safety Analysis Report, Environmental Report, and application) and Commission Rules and Regulations and Regulatory Guides may be inspected at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D. C. Correspondence between the Commission and the applicant may also be inspected at the Public Document Room identified in Section 1.1 of this report. Specific documents relied upon by the Commission's staff and referenced in this Safety Evaluation Report are listed as follows:

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APPENDIX C

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS GENERIC ITEMS

The Advisory Committee on Reactor Safeguards (the Committee) periodically issues a report that lists various generic matters applicable to large light-water reactors. Although the Committee and the Commission's staff find that present plant designs are acceptable, they also believe that the overall safety margin of nuclear power plants could be enhanced if these items were to be considered in the designs. They believe that application of these items should be made, to the extent reasonable and practicable as solutions are found, and they recognize that such solutions may occur after completion of the plant. This is consistent with our continuing efforts toward reducing still further the already small risk that nuclear power plants pose to the public health and safety. The most recent such report concerning these generic items was issued in a letter dated November 15, 1977 to Commission Chairman J. Hendrie from Committee Chairman M. Bender.

The status of staff efforts leading to resolution of all these generic matters is contained in our status report on generic items periodically transmitted to the Committee. The latest such status report is contained in a letter dated May 4, 1978, from Acting Director E. Case to Committee Chairman S. Lawroski.

For many of the items, we have provided specific discussion in this report, particularizing the generic status for the proposed facility. These items are listed below, with the appropriate section numbers of this report where such discussions are to be found. The numbering corresponds to that in the November 15, 1977 report of the Committee.

For those items, applicable to the proposed facility, that have not progressed to a point where specific action can be initiated on Davis Besse Units 2 and 3, our status report on generic items referred to above provides the appropriate information.

Group II- Resolution Pending

- (1) Turbine Missiles. Resolved for this facility by the applicant's commitments to provide a turbine-missile damage probability analysis and to design the facility so that protective barriers can be provided if found to be needed as a result of the analysis (Section 3.5.4). The turbine generator overspeed protection design criteria were found acceptable (Section 10.2).

- (2) Effective Operation of Containment Sprays in a Loss-of-Coolant Accident. Resolved for this facility by the use of sodium hydroxide additive to sprays (Sections 6.1, 6.2.2 and 15).
- (3) Possible Fracture of Pressure Vessel Post-Loss-of-Coolant Accident by Thermal Shock. This item is under generic review, as indicated in our status report to the Advisory Committee on Reactor Safeguards dated May 4, 1978.
- (4) Instruments to Detect (Severe) Fuel Failures. This item is partly resolved as reported in the November 15, 1977 letter from the Committee to the Commission. Instrumentation to detect fuel failures associated with normal operation and transients (limited fuel failures) has been shown to be inadequate. The adequacy of instrumentation to detect failures associated with more rapid events, during which substantial fuel failure could occur, has not been demonstrated and this concern is considered unresolved. Further work is necessary to determine the adequacy of current instrumentation for these rapid events and the need for additional instrumentation. Research administered by the Office of Reactor Safety Research, and studies conducted under contracts administered by the Office of Nuclear Reactor Regulation, should provide the information required to evaluate instrumentation limitations and needs. In the interim, we have not identified any credible event (transient or accident sequence) for which a rapid fuel failure detection system would prevent "substantial" fuel failure (including fuel melt) and loss of coolable geometry. We conclude that this ongoing generic item should not preclude issuance of construction permits for Davis-Besse Units 2 and 3.
- (5a) Monitoring for Loose Parts Inside the Pressure Vessel. This item is resolved for this facility by the proposed installation of a loose parts monitoring system (Section 5.2.5).
- (5b) Monitoring for Excessive Vibration Inside the Reactor Pressure Vessel. This item is under generic review, as indicated in our status report to the Committee dated May 4, 1978.
- (6) Non-Random Multiple Failures. This item is under generic review, as indicated in our status report to the Committee dated May 4, 1978.
- (7) Behavior of Reactor Fuel Under Abnormal Conditions. This item is under generic review, as indicated in our status report to the Committee dated May 4, 1978.
- (8) Boiling Water Reactor Recirculation Pump Overspeed During a Loss-of-Coolant Accident. This item is not applicable to Davis-Besse Units 2 and 3, which will be pressurized water reactor facilities.

- (9) The Advisability of Seismic Scram. A seismic scram is not proposed for Davis-Besse Units 2 and 3 and we will not require such a scram. Our position on this item is given in a letter dated May 19, 1977 from E. Case, Acting Director, Office of Nuclear Reactor Regulation, to Committee Chairman Bender, subject, "The Advisability of a Seismic Scram." However, this item will remain under generic review, as indicated in our status report to the Committee dated May 4, 1978.
- (10) Emergency Core Cooling System Capability for Future Plants. This item is under generic review, as indicated in our status report to the Committee dated May 4, 1978.

Group IIA Resolution Pending - Items Since December 18, 1972

- (1) Ice Condenser Containments. This item is not applicable to Davis-Besse Units 2 and 3, which will not utilize the ice condenser containment concept.
- (2) Pressurized Water Reactor Pump Overspeed During a Loss-of-Coolant Accident
- This item is under generic review, as indicated in our status report to the Committee dated May 4, 1978.
- (3) Steam Generator Tube Leakage. This item is resolved for this facility by controls on secondary system chemistry and the design provisions for inservice inspection. (Section 5.4.2)
- (4) ACRS/NRC Periodic 10-Year Review of All Power Reactors. This item is under generic review, as indicated in our status report to the Committee dated May 4, 1978.

Group IIB Resolution Pending - Items Added Since February 13, 1974

- (1) Computer Reactor Protection System. This item is resolved for Davis-Besse Units 2 and 3 by the applicant's commitment to the staff resolution reached for the RPS-II system currently under generic review of the Babcock & Wilcox topical report BAW-10085. (Section 7.2)
- (2) Qualification of New Fuel Geometries. This item is partially resolved by similarity to existing fuel geometries of proven performance and by the Babcock & Wilcox ongoing test programs for the Mark C fuel assembly. The continuation of these test programs and an industry-wide surveillance program provide an ongoing generic review of this item. We conclude this generic qualification program is directed at design confirmation and should not preclude the issuance of construction permits for Davis-Besse Units 2 and 3. (Section 4.2).

- (3) Behavior of Boiling Water Reactor Mark III Containments. This item is not applicable to Davis-Besse Units 2 and 3, which will be pressurized water reactor facilities.
- (4) Stress Corrosion Cracking in Boiling Water Reactor Piping. This item is not applicable to Davis-Besse Units 2 and 3, which will be pressurized water reactor facilities.

Group IIC Resolution Pending - Items Added Since March 12, 1975

- (1) Locking Out of Emergency Core Cooling System Power Operated Valves. This item is resolved for Davis-Besse Units 2 and 3 by the applicant's commitment to lock out power to appropriate valves. (Section 6.3.1)
- (2) Design Features to Control Sabotage. This item is resolved for Davis-Besse Units 2 and 3 by compliance with current NRC staff requirements. (Section 13.7)
- (3) Decontamination and Decommissioning of Reactors. This item is under generic review, as indicated in our status report to the Committee dated May 4, 1978.
- (4) Vessel Support Structures. This item is under generic review of Babcock and Wilcox topical reports BAW-10131 and BAW-10132. This item will be resolved for Davis-Besse Units 2 and 3 prior to a decision to issue an operating license. (Section 3.9.2)
- (5) Water Hammer. This item is under generic review, as indicated in our status report to the Committee dated May 4, 1978.
- (6) Maintenance and Inspection of Plants. This item is resolved for Davis-Besse Units 2 and 3 by compliance with current NRC requirements. (Section 12.1).
- (7) Behavior of Boiling Water Reactor Mark I Containments. This item is not applicable to Davis-Besse Units 2 and 3, which will be pressurized water reactor facilities.

Group IID Resolution Pending - Items Added Since April 16, 1976

- (1A) Safety-Related Interface Between Reactor Island and Balance-of-Plant. This item is not applicable to Davis-Besse Units 2 and 3, which are custom designs.
- (1B) Systems Interactions in Nuclear Power Plants. This item is under generic review, as indicated in our status report to the Committee dated May 4, 1978.

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- (2) Assurance of Continuous Long-Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment. This item is under generic review, as indicated in our status report to the Committee dated May 4, 1978.

Group IIE Resolution Pending - Items Added Since February 24, 1977

- (1) Soil-Structure Interactions. This item is under generic review, as indicated in our status report to the Committee dated May 4, 1978.

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APPENDIX D

CHRONOLOGY
RADIOLOGICAL SAFETY REVIEW

Note: Documents referenced in this chronology are available for public inspection and copying for a fee at the NRC Public Document Room, 1717 H Street, N.W., Washington, D.C. 20555 and at the Ida Rupp Public Library, Port Clinton, Ohio 43452.

January 24, 1974	Letter to TECo regarding anticipated transients without scram.
May 10, 1974	Letter from TECo transmitting application.
June 4, 1974	Letter to TECo concerning acceptance review.
June 12, 1974	Letter to TECo concerning adequacy of the quality assurance program.
June 20, 1974	Meeting with TECo to discuss quality assurance program.
July 19, 1974	Letter to TECo accepting Preliminary Safety Analysis Report and Environmental Report for docketing, subject to certain conditions.
July 24, 1974	Letter from TECo transmitting updated quality assurance program, responding to June 12, 1974 letter.
August 7, 1974	Letter from TECo transmitting application and Environmental Report, responding to July 19, 1974 letter.
August 16, 1974	Preliminary Safety Analysis Report docketed.
August 27, 1974	Letter to TECo acknowledging the docketing of the application.
September 16, 1974	Letter from TECo transmitting Preliminary Safety Analysis Report Revision 1.
September 30, 1974	Letter to TECo concerning safety review schedule.
October 8, 1974	Meeting with TECo to discuss foundation conditions.
October 25, 1974	Letter to TECo regarding anticipated transients without scram.

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October 28, 1974 Letter from TECo responding to January 24, 1974 letter regarding anticipated transients without scram.

October 31, 1974 Letter to TECo requesting additional information.

November 8, 1974 Letter to TECo requesting additional information.

November 15, 1974 Letter to TECo requesting additional information concerning core performance.

November 18, 1974 Letter from TECo transmitting Preliminary Safety Analysis Report Revision 2.

November 22, 1974 Letter to TECo requesting additional information.

December 6, 1974 Letter to TECo requesting additional information regarding ventilation systems.

December 19, 1974 Meeting with TECo to discuss hydrology.

January 8, 1975 Letter from TECo applying BAW-10099 to anticipated transients without scram.

January 22, 1975 Meeting with TECo to discuss replication of Unit 1.

February 7, 1975 Letter from TECo transmitting Preliminary Safety Analysis Report Revision 3.

March 5, 1975 Letter from TECo advising of slip in construction schedule.

March 7, 1975 Letter from TECo transmitting Preliminary Safety Analysis Report Revision 4.

April 30, 1975 Letter to TECo requesting better reproduction of drawings in the Preliminary Safety Analysis Report.

July 1, 1975 Letter to TECo requesting additional information about emergency core cooling analysis.

August 14, 1975 Meeting with TECo to discuss replication.

September 4 & 5, 1975 Meeting with TECo to discuss various systems design criteria.

October 16, 1975 Letter to TECo concerning acceptability of containment sump and containment design pressure.

November 12, 1975 Letter to TECo requesting additional information.

November 13, 1975 Public hearing held to consider environmental matters.

December 9, 1975 Letter to TECo with a page that was missing from the November 12, 1975 letter.

December 15, 1975 Letter from TECo transmitting Preliminary Safety Analysis Report Revision 5.

December 31, 1975 Atomic Safety and Licensing Board issued partial initial decision concerning environmental matters.

December 31, 1975 Letter to TECo authorizing certain site preparation activities.

January 23, 1976 Letter to TECo requesting additional information.

January 29, 1976 Meeting with TECo to discuss various auxiliary systems.

February 20, 1976 Letter from TECo transmitting Preliminary Safety Analysis Report Revision 6.

March 18, 1976 Letter to TECo requesting additional information about reactor vessel flow tests and loose parts monitoring.

March 26, 1976 Meeting with TECo to discuss equipment supports and seismic design.

April 5, 1976 Letter from TECo transmitting Preliminary Safety Analysis Report Revision 7.

May 3, 1976 Letter to TECo transmitting guidance on fire protection.

May 10, 1976 Letter to TECo regarding number of copies required of submittals.

June 8, 1976 Letter to TECo regarding further review of anticipated transients without scram.

June 18, 1976 Letter from TECo transmitting Preliminary Safety Analysis Report Revision 8.

June 29, 1976 Letter from TECo responding to letter of June 8, 1976.

July 23, 1976 Letter from TECo transmitting Preliminary Safety Analysis Report Revision 9.

August 6, 1976	Letter to TECo giving additional information on May 10, 1976 letter.
August 30, 1976	Letter from TECo transmitting Preliminary Safety Analysis Report Revision 10.
September 15, 1976	Meeting with TECo to discuss fire protection, cooling water systems, and concrete thickness.
September 30, 1976	Letter from TECo responding to June 29, 1976 letter, delaying input of information concerning anticipated transients without scram.
September 30, 1976	Letter to TECo transmitting further guidance on fire protection.
October 5, 1976	Letter to TECo requesting additional information concerning loss-of-coolant accident analysis.
October 29, 1976	Letter from TECo concerning schedule for submitting fire protection program, in response to September 30, 1976 letter.
November 1, 1976	Letter from TECo transmitting Preliminary Safety Analysis Report Revision 11.
December 17, 1976	Letter to TECo enclosing errata sheet for September 30, 1976 letter.
January 3, 1977	Letter to TECo regarding emergency core cooling evaluation models (enclosed December 2, 1976 letter to K. Suhrke of Babcock & Wilcox).
January 14, 1977	Letter from TECo transmitting Preliminary Safety Analysis Report Revision 12.
January 17, 1977	Letter to TECo presenting staff positions on outstanding items of review.
January 24, 1977	Letter to TECo requesting additional information about subcompartment pressures.
February 9, 1977	Letter to TECo concerning qualification requirements for plant operating personnel.
February 14, 1977	Letter to TECo correcting an item enclosed with January 17, 1977 letter.
February 15, 1977	Letter from TECo responding to January 3, 1977 letter.

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February 15, 1977 Letter from TECo concerning schedule for responding to January 17, 1977 letter.

March 9, 1977 Letter from TECo, changing the schedule indicated in TECo's October 29, 1976 letter for submittal of fire protection information.

March 11, 1977 Meeting with TECo to discuss containment analysis.

March 28, 1977 Letter from TECo transmitting Preliminary Safety Analysis Report Revision 13.

March 30, 1977 Letter to TECo requesting analysis of fuel handling accident inside containment.

April 26, 1977 Letter to TECo requesting additional information concerning instrumentation and electric power systems.

May 16, 1977 Letter from TECo transmitting Preliminary Safety Analysis Report Revision 14.

May 17, 1977 Meeting with TECo to discuss main steam line break analysis.

May 25, 1977 Letter to TECo advising basis for unacceptability of their combustible gas control system.

June 16, 1977 Letter from TECo transmitting Fire Hazard Analysis Report.

July 13, 1977 Letter to TECo advising unacceptability of their proposed seismic acceleration and of certain foundation design.

July 18, 1977 Letter from TECo transmitting Preliminary Safety Analysis Report Revision 15, providing responses to NRC letters of October 5, 1976, January 17, 1977, January 24, 1977 and April 25, 1977.

July 25, 1977 Letter from TECo transmitting Preliminary Safety Analysis Report Revision 16, providing information on main steam line breaks.

August 17, 1977 Letter from TECo transmitting Preliminary Safety Analysis Report Revision 17, providing responses to NRC letters of January 17, 1977 and April 25, 1977.

August 29, 1977 Letter from TECo responding to July 13, 1977 letter.

August 29, 1977 Letter to TECo correcting a statement made in the July 13, 1977 letter.

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August 29, 1977	Letter to TECo providing additional guidance on fire protection.
August 31, 1977	Letter from TECo transmitting Preliminary Safety Analysis Report Revision 18, responding to July 13, 1977 and October 5, 1976 letters.
September 1, 1977	Meeting with TECo to discuss turbine missile damage analysis.
September 13, 1977	Letter from TECo providing additional information concerning subcompartment pressures.
September 16, 1977	Letter from TECo providing additional information about turbine missiles.
September 29, 1977	Meeting with TECo to discuss combustible gas control.
October 13, 1977	Letter from TECo transmitting Preliminary Safety Analysis Report Revision 19, responding to January 24, 1977 letter.
October 25, 1977	Letter to TECo providing guidance on plant physical security.
November 30, 1977	Letter to TECo requesting additional information on loss-of-coolant accident analysis.
January 16, 1978	Letter from TECo responding to November 30, 1977 letter on loss-of-coolant accident analysis.
January 25, 1978	Letter to TECo requesting additional financial information.
January 27, 1978	Letter from TECo transmitting Preliminary Safety Analysis Report Revision 20, updating the quality assurance program and the project organization.
February 3, 1978	Letter to TECo addressing outstanding issues.
February 10, 1978	Letter to TECo deleting one of the outstanding issues of the February 3, 1978 letter.
February 28, 1978	Letter from TECo responding to five outstanding items of the February 3, 1978 letter.
March 7, 1978	Letter from TECo responding to nine outstanding items of the February 3, 1978 letter.
March 10, 1978	Letter from TECo responding to four outstanding items of the February 3, 1978 letter.

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March 10, 1978 Letter from TECo, providing additional financial information requested in January 24, 1978 letter.

March 17, 1978 Letter to TECo giving staff position on main steam line breaks.

March 22, 1978 Letter from TECo responding to one outstanding item of February 3, 1978 letter.

March 30, 1978 Letter from TECo responding to outstanding item, concerning containment subcompartment analysis, of February 3, 1978 letter.

April 3, 1978 Letter from TECo providing corrections to TECo's letter concerning outstanding items.

April 11, 1978 Letter from TECo responding to item of February 3, 1978 letter concerning protection against main steam line break.

April 14, 1978 Letter to TECo requesting additional information concerning component supports.

April 24, 1978 Letter from TECo correcting Preliminary Safety Analysis Report information regarding control rod assembly neutron absorbing material.

April 24, 1978 Letter from TECo correcting Preliminary Safety Analysis Report information concerning titles of certain operating personnel.

May 1, 1978 Letter from TECo updating information concerning quality assurance.

May 15, 1978 Letter from TECo responding to March 17, 1978 letter concerning main steam line breaks.

May 23, 1978 Letter from TECo responding to April 14, 1978 letter.

May 25, 1978 Letter from TECo requesting authorization to perform certain activities pursuant to K50.10(e)(3) of 10 CFR Part 50.

June 1, 1978 Letter from TECo providing additional financial information.

June 6, 1978 Letter to TECo requesting additional financial information.

June 6, 1978 Letter from TECo amending the request of May 25, 1978.

June 6, 1978 Letter from TECo providing additional commitments concerning two outstanding items of our February 3, 1978 letter.

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June 12, 1978 Letter to TECo regarding capability for cold shutdown.

June 26, 1978 Letter from TECo committing to meeting our positions on three outstanding items of our February 3, 1978 letter.

June 28, 1978 Letter from TECo providing a commitment to monitor the settlement of the turbine building, the auxiliary building, and the office building.

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
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