Docket No. 50-346

July 24, 1970

Report to the ACRS

Davis-Besse Nuclear Power Station

Construction Permit

U.S. Atomic Energy Commission
Division of Reactor Licensing

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1.0 INTRODUCTION AND SUMMARY

1.1 General

On August 1, 1969, the applicants, The Toledo Edison Company (TEC) and The Cleveland Electric Illuminating Company (CEIC), as co-owners (TEC 52.5% and CEIC 47.5%), filed an application for a construction permit and operating license for a nuclear power plant designated as the Davis-Besse Nuclear Power Station. Commercial operation is scheduled for December 1974. The 900-acre plant site is located on the southwestern shore of Lake Erie in Ottawa County, Ohio, approximately 26 miles east of Toledo, Ohio.

The plant will have a two-loop Babcock and Wilcox PWR nuclear steam supply system. The reactor is designed for an initial core power level of 2633 megawatts thermal (MWt). The applicants have evaluated the site parameters, principal structures, engineered safety features and accident analyses based on the ultimate core power level of 2772 MWt.

The turbine generator will be supplied by the General Electric Company.

The Toledo Edison Company will have the responsibility for the design, construction and operation of the facility. The architect-engineer design and construction management of the Davis-Besse plant will be performed by the Bechtel Corporation (Gaithersburg).

The containment and shield building will be physically larger than, but functionally similar to, the systems proposed for the Prairie Island and Kewaunee nuclear power stations. The controls, instrumentation, and

engineered safety features are similar to those used on the Three Mile Island Nuclear Power Station Units 1 and 2.

For the design basis loss-of-coolant accident, we have calculated a two-hour thyroid dose of 140 rem at the exclusion radius and a 30-day low population zone thyroid dose of 160 rem.

The containment spray system will not use any additives for removal and retention of radioactive iodine.

We are presently reviewing an exemption request submitted on June 4, 1970 by the applicants to permit construction of the facility's foundations and buildings up to about six inches below the site grade level.

1.2 Major Areas of Review

Because of the similarity of the Davis-Besse containment design to the Prairie Island and Kewaunee designs, and the nuclear steam supply system to the other B&W PWR's, our review has been based to a large extent on comparison with previously approved features. Features which were significantly different were identified and evaluated in more detail. Emphasis was placed on (1) unique features, such as site-related items, (2) the increase in core power level from 2452 to 2633 MWt, (3) changes in design of B&W nuclear steam supply system such as removal of the core vent valves, and (4) the quality assurance program.

A chronology of the principal events during the review is given in Table 1.2.

TABLE 1.2

CHRONOLOGY

REGULATORY REVIEW OF DAVIS-BESSE CONSTRUCTION PERMIT APPLICATION

Application for Construction Permit filed	August 1, 1969
Initial meeting with Toledo Edison Company and Bechtel	September 10, 1969
Technical meeting to review site-related matters	November 7, 1969
Technical meeting to review facility design	December 17, 1969
Applicants filed Amendment No. 1 correcting errors in PSAR, changes in building foundations, and methods of treating solution cavities.	December 17, 1969
Meeting with Toledo Edison Company, Bechtel, and Babcock & Wilcox to discuss Davis-Besse Quality Assurance Program.	January 28, 1970
Request for additional information submitted to applicants.	February 12, 1970
Amendment No. 2 containing description of Davis-Besse Quality Assurance Program and changes in seismic design for increasing the operating basis earthquake horizontal acceleration from 0.06g to 0.08g.	March 2, 1970
Meeting with applicants to review structural and critical components, seismic and tornado design bases and methods of analysis.	March 13, 1970
Meeting with applicants to discuss seismic response spectra and time-history accelerogram to be used for seismic design of Class I structures and components.	April 2, 1970
Amendment No. 3 containing applicants' response to February 12, 1970 request for additional information.	April 22, 1970
Amendment No. 4 containing documentation of matters agreed upon at the April 2, 1970 meeting.	April 30, 1970

Amendment No. 5 containing applicants' response to February 12, 1970 request for information and additional information on site geology and environmental program.	May 15, 1970
Technical meeting to discuss seismic response spectrum and time-history accelerogram.	May 19, 1970
Technical meeting to discuss site-related matters and applicants' Amendments Nos. 3 and 5.	May 20, 1970
Site visit by staff and ACRS Subcommittee.	May 26, 1970
Amendment No. 6 containing seismic design and clarification of matters discussed at May 20, 1970 technical meeting.	June 12, 1970
Amendment No. 7 describing exploration and verification procedures for possible solution cavities and fissures in site bedrock.	July 1, 1970
Technical meeting to discuss the flood protection level required for safe shutdown of the facility.	July 17, 1970

1.3 Summary

At the writing of this report, we have three areas which are still unresolved. We have also identified other areas where additional information will be provided by the applicants after issuance of the construction permit. These matters are discussed in this report and are summarized below.

1.3.1 Unresolved Matters

a. Flood Protection

The applicants have indicated the present flood protection for the plant is the 585-foot mean sea level (MSL) elevation. This flood protection elevation provides a 3.5-foot margin above the extreme high lake water level which the applicants determined to be 581.5 feet MSL. We have reviewed the method of analysis used by the applicants to arrive at the extreme lake water level and do not consider the method used adequate to calculate the probable maximum lake water level at the site. The applicants have performed additional analyses, which we have recommended, to arrive at the storm surge and wave runup values. The results of these calculations were reviewed with us and our consultant from CERC in a meeting with the applicants on July 17, 1970. The applicants' proposed flood protection level, which we understand will be 590 feet MSL, will be documented prior to the ACRS Subcommittee meeting on August 4, 1970. We will accept this flood protection level.

b. Pipe Whip Criteria

The applicants' proposed pipe whip criteria do not protect the secondary system from failure due to a primary system pipe rupture. We have informed the applicants that we plan to require either that (1) the primary system piping have restraints to prevent failure of the secondary system due to pipe whip effects or (2) the containment design be capable of withstanding the pressure transient resulting from a loss-of-coolant (LOCA [14.1 ft^2 break]) coincident with failure of one secondary system without exceeding design pressure and retaining a margin of at least 10%.

The applicants have not agreed to provide either of the above alternatives.

c. Pressure Vessel Cavity

The applicants propose to design the pressure vessel cavity to withstand the pressure transient resulting from a 3.0 ft^2 primary system rupture. They have indicated that this break size (3.0 ft^2) is the largest possible break which could occur within the pressure vessel cavity.

We have informed the applicants that we plan to require that the pressure vessel cavity design be capable of withstanding the pressure transient resulting from the largest primary pipe rupture (14.1 ft^2) without resulting in failure of any systems required for core cooling and will not generate any missiles which could

result in failure of the containment or the engineered safety features. These requirements are consistent with those applied to Millstone Unit 2 which was reviewed at the May 1970 ACRS meeting. The applicants have not agreed to provide this design capability.

1.3.2 Areas of Continuing Review

In addition to the areas which will be reviewed in detail at the operating stage, we plan to continue our review of the following matters during construction.

a. Confirmatory Vibration Program (Section 3.0)

We have indicated to the applicants that a confirmatory vibration program may be required to assure no undetected vibrational problems exist in the assembled reactor system. The applicants have indicated that they will follow the startup and preoperational test results for B&W nuclear steam supply systems and will be guided by these results in establishing the necessary program or test.

b. Design Bases Earthquake and Blowdown Forces (Section 12.7)

The topical report BAW 10008* is currently being revised to cover additional areas of concern developed by us during the review of the Oconee application. This topical report required additional revision

*BAW 10008, "Reactor Internals Stress and Deflection Due to Loss-of-Coolant Accident and Maximum Hypothetical Earthquake."

to cover the Davis-Besse nozzle-supported pressure vessel design. We intend to continue our review of this topical report during the construction of this plant and the operating license review of the Oconee application.

c. Post-accident Hydrogen Control (Section 5.9)

The applicants have described the program they have initiated to determine the evolution and control of hydrogen within the containment building following a loss-of-coolant accident. The applicants have been informed that purging as a means of controlling the hydrogen concentration will be acceptable only as a backup to another control method such as catalytic or flame recombiners.

d. Containment Building Design (Section 5.2)

We have reviewed the criteria established for the containment building design and find them acceptable. The containment building is an ASME Boiler and Pressure Vessel Code, Section III, Class B vessel which will be constructed by Chicago Bridge & Iron. We have requested the applicants to submit a preliminary design report for the containment building for our review to assure the criteria established in the PSAR are being implemented. This preliminary design report will not be completed until about November 1970.

e. Steam Line Failure Accident (Section 9.5)

We are currently evaluating the steam line failure accident in depth for the Oconee operating license review. The consequences of this accident, if operator does not place the feedwater control system

in the manual mode, can result in the reactor returning to criticality due to cool down effects. It had been previously believed that a return to criticality would not occur on B&W plants because of the low water inventory in the steam generators; however, if operator action is not taken, additional feedwater would be added to the steam generator thereby resulting in additional cool down of the primary system. The analysis of the fuel clad temperatures and possibility of departure from nucleate boiling occurring are being reviewed with B&W for the Oconee operating license. B&W has stated, however, that the criterion for no fuel damage will still apply for this accident. We plan to follow this matter to assure that any design modifications required are applied to the Davis-Besse design.

f. Interlock (Section 7.2.3)

The applicants have indicated there are no instrumentation and control interlocks whose failure would result in fuel failure. In addition, the applicants have indicated if such problems were discovered as the design progresses the interlocks would be designed to the requirements of IEEE-279. We will continue to review the design to assure that the above criterion is met.

g. ECCS Protection Against Active Component Failure (Section 6.1)

The effects of active component failure during ECCS operation are under further review by us and B&W based upon results of our review of the North Anna facility. The concern is directed toward the inadvertent operation of motor-operated valves that may negate ECCS operation.

1.3.3 Conclusion

We conclude that the proposed facility can be constructed and operated without undue risk to the health and safety of the public provided the matters discussed above are satisfactorily resolved.

2.0 SITE & ENVIRONMENT

2.1 General Description

The Davis-Besse site is located on the southwestern shore of Lake Erie in Ottawa County, Ohio approximately 26 miles east of Toledo, Ohio. The site consists of approximately 900 acres owned as tenants in common by The Toledo Edison Company (TEC) and The Cleveland Electric Illuminating Company (CEIC). A large portion of the site was acquired under an agreement with the Bureau of Sport Fisheries and Wildlife of the Department of the Interior, United States Government. This agreement will permit the unused marsh areas of the site to be used as a National Wildlife Refuge. The site topography is virtually flat with no natural promontories.

The Toledo Express Airport is the closest commercial airport and is 38 miles west of the site. A small, non-commercial airport with a paved runway is located at Port Clinton, Ohio, 13 miles east-southeast of the site.

The applicants have indicated there has been no mining or extraction of coal, oil, gas, or salt from beneath the site.

2.2 Population Distribution

The minimum exclusion distance will be 2400 feet, within which there are no residences. The low population zone boundary of two miles will have a projected population of 1213 (summer) and 702 (permanent) residents in the year 2000. The site is located approximately 20 miles from the nearest city limits of Toledo and Sandusky, which had 1960 populations of 379,133 and 31,989, respectively.

The applicants have indicated that there are no hospitals, schools, or detention institutions within the low population zone radius of two miles.

Based upon the projected population distribution around the plant, we conclude that the 2400-foot exclusion distance and the low population zone distance of two miles are acceptable.

2.3 Meteorology

The Toledo Edison Company has provided about six months of meteorological data to be used in developing meteorological assumptions for the design basis accident (DBA). These data consist of wind speed and direction at the 20-foot elevation and temperature change with elevation. During the six-month period, diffusion conditions more conservative than Pasquill Type F and 1.5 m/sec occurred 5% of the time which is our occurrence limit for meteorological assumptions for the first two hours of the DBA. The six-month period did not include summer when, according to Environmental Science Services Administration (ESSA) data, the wind speeds at nearby Toledo, Ohio are lowest. Thus, we and our consultant at ESSA conclude that Type F and 1.0 m/sec are the appropriate assumptions

for the first two hours of the DBA. For this period, the applicants assumed Type F and 2.0 m/sec based on wind speeds at the 100-foot elevation rather than the 20-foot elevation.

The applicants will continue to collect meteorological data at the 20-foot elevation to provide a basis for the atmospheric dilution factors to be used in the Technical Specifications for the routine releases of airborne radionuclides and the accident analyses with the application for the operating license.

Comments by ESSA will be forwarded to the Committee.

We conclude that the applicants' proposed meteorological program is acceptable.

2.4 Geology and Seismology

A geological study of the proposed site provided by the applicants shows that bedrock exists below 14 to 30 feet of glacial till and locustrine clays. The bedrock consists of slightly dipping Paleozoic sedimentary rocks which have varying amounts of soluble anhydrite and soluble gypsum. Indications of solution cavities were found in the bedrock at a test excavation 400 feet south of the proposed plant location. In response to our concerns about the potential of large cavities underlying the plant site, the applicants have agreed to undertake a program of foundation bedrock inspection, exploration, and, where necessary, grouting under the Class I structures. The applicants have identified a grouting procedure which is acceptable to us. The cavity exploration program is presented in Amendment No. 7. We and our consultants from the U. S. Geological Survey (USGS) have reviewed this

program and conclude that it should be adequate to discover any significant solution cavities or fissures which might occur in the site bedrock. The report from the USGS will be provided to the Committee prior to the August meeting.

The seismic design aspects of the facility are discussed in Sections 4.0 and 5.0 including the comments from our consultant on these matters; i.e., John Blume and Associates. Their report will be provided to the Committee prior to the August meeting.

The earthquakes experienced within 50 miles of the site have not exceeded a Modified Mercalli (MM) intensity of V. There are no known faults within the site boundary. For design purposes, the applicants consider that earthquakes with an MM VI intensity at the site are probable and that earthquakes with an MM VII intensity at the site are possible. A maximum horizontal ground acceleration of 0.15 gravity will be assumed for the design basis earthquake and a horizontal ground acceleration of 0.08 gravity will be assumed for the operating basis earthquake. We and the U. S. Coast and Geodetic Survey (USC&GS) conclude that the proposed design acceleration factors are acceptable. The report from the USC&GS will be provided to the Committee prior to the August meeting.

2.5 Hydrology

The Davis-Besse plant will be located about 3000 feet from the shore of Lake Erie. The applicants have assumed that the probable maximum low water level will be lower than the bottom of the intake canal. To assure the availability of shutdown cooling water for this event, the applicants

have designed a Class I intake forebay that will contain 10 million gallons of stored water. The applicants have determined, and we agree, that the 10 million gallons of water would provide more than 70 days shutdown cooling water.

We have also reviewed the effects of flooding due to increased water levels in Lake Erie. The vital components and structures of the plant will be protected to a combination of (surge) wind setup, seiche and wave runup resulting from the probable maximum meteorological conditions. We and our consultants, Coastal Engineering Research Center (CERC) met with the applicants on July 17, 1970 to discuss flooding at the site. At this meeting we agreed that the water level, including wind setup and seiche occurring simultaneously with the maximum seasonal variation in lake level, would result in a probable maximum peak water level of 15.1 feet above 568.6 feet low water datum (568.6 plus 15.1 feet equals 583.7 feet International Great Lakes Datum (IGLD). The runup at the plant site associated with the maximum water level will be determined by the applicants and reviewed with us. The applicants verbally agreed that the vital components and structures of the plant essential for safe shutdown will be protected to the combined probable maximum water level of 15.1 feet elevation 583.7 feet IGLD plus the associated wave runup. We and our consultant, CERC, conclude that this analysis to establish the design basis is acceptable. We expect to have the final protection elevation available prior to the Committee's August meeting.

2.6 Environmental Monitoring

The applicants will initiate an environmental monitoring program at least 18 months prior to plant operation. This program will establish the type, number, frequency, and methods of analyzing samples. The samples will consist of lake and well water, soil, air particulates, farm products, lake biota, fish and bottom sediments.

The applicants initiated a limnology study in 1968 to evaluate the past, present, and projected future use of Lake Erie. Included in this study will be a field investigation to determine physical, chemical and biological characteristics of the offshore lake regime which could be affected by the station effluents.

The above studies and programs are being conducted and developed with cooperation and recommendations of the state and federal agencies. The applicants' program is being reviewed by the Fish and Wildlife Service and its recommendations will be forwarded to the applicants and the Committee prior to the August meeting.

The applicants have indicated that they will continue to cooperate with the various government agencies in formulating and implementing the environmental programs. We will prepare an environmental statement for this facility covering those matters in accordance with the National Environmental Policy Act.

We conclude that the environmental monitoring program for this facility is acceptable.

2.7 Restricted Areas

There are three areas located near the station site which have been established as restricted areas for use by branches of the Armed Forces

and industry as impact areas for air-to-ground gunnery and bombing training missions, small arms firing and anti-aircraft weapons. These areas are clearly identified in Appendix 2A of the PSAR and the type of weapons used for each restricted area is specified. The control of activities within the restricted areas is enforced by the Adjutant General, State of Ohio, and such agencies to whom he may delegate this authority. The applicants' response to Question 1.10 of our request for additional information indicates that the structures and vital components which require missile protection design will be protected by a reinforced concrete wall of at least 18 inches thickness and will provide adequate protection against any presently used missiles which might be generated from the restricted areas.

The applicants have indicated the activities within the restricted areas will continue to decrease and eventually the restricted areas may be eliminated. At the May 26, 1970 site visit and ACRS Subcommittee meeting on site matters, the applicants indicated that the status of the activity within the restricted areas had changed from the description presented in the PSAR. The applicants indicated that the TRW, Inc. test firing of the 25 mm Armor Piercing Discarding Sabot-T and 35 mm cannon has been discontinued. A new company, Cadillac Gage, a subsidiary of Excello Corporation, has taken over the TRW, Inc. test area to test tanks firing a mortar-type weapon with a 2500 to 3000 meter range, which would not be capable of reaching the plant.

The revised restricted area activity is described in Amendment No. 6.

We and our consultant, Mr. J. Proctor of the U. S. Naval Ordnance Laboratory, have reviewed the penetration capability of projectiles which could be generated from the restricted area and find none of the present projectiles could penetrate or result in scabbing of the minimum reinforced concrete thickness of 18 inches.

We conclude that the facility design for missile protection will provide adequate protection against any current missiles which would be inadvertently generated from the adjacent restricted areas.

3.0 REACTOR DESIGN

3.1 General

The mechanical, nuclear, thermal and hydraulic design of the Davis-Besse reactor is similar to previously reviewed B&W nuclear steam supply systems except for the following changes: (1) the design power level is increased from 2452 MWt to 2633 MWt, (2) the vent valves have been eliminated, and (3) the pressure vessel will be supported by its recirculation line nozzles.

Table 3.1 shows a comparison of the proposed reactor design parameters with the parameters of the latest reviewed pressurized water nuclear steam supply system of Westinghouse and Combustion Engineering.

3.2 Control Rod Drive Assemblies & Burnable Poison Rods

The Davis-Besse reactor will have 49 control rod drive assemblies and 8 partial length control rod drive assemblies. The control rod drive assemblies will employ the roller-nut type drive with a scram time specification of 1.4 seconds for 2/3 insertion. The roller-nut control rod drive has been established as the standard B&W drive unit. The research and development of this drive has been completed. These R&D tests and results are reported in BAW 10007, "Control Rod Drive System Test Program." The control rod drive used to position the partial length rods for control of xenon oscillations will be the same as a standard drive except the roller-nut will not have the capability to scram.

The number of standard control rod drive assemblies (CRA) required has been reduced from 61 for B&W systems designed prior to Midland to 49 for

TABLE 3.1

COMPARISON OF REACTOR DESIGN PARAMETERS

Parameter	Beaver Valley	Millstone Unit 2	Midland	Davis-Besse
Nuclear Steam Supply Supplier	West.	Comb. Engr.	B&W	B&W
Rated Core Thermal Power; MWt	2652	2560	2452	2633
Average Thermal Output; kW/ft	6.7	5.9	5.4	5.8
Maximum Thermal Output; kW/ft	17.9	18.2	16.8	17.8
Maximum Thermal Output at 112% Power; kW/ft	20.0	20.4	18.8	19.9
Maximum Heat Flux; Btu/hr-ft^2	543,300	525,800	410,300	538,730
Average Heat Flux; Btu/hr-ft^2	207,600	169,600	163,725	175,811
Total Flow; lbs/hr	100.7x10^6	122x10^6	131.3x10^6	131.3x10^6
Effective Flow for Heat Transfer; lbs/hr	96.2x10^6	118.5x10^6	124.2x10^6	124.2x10^6
Effective Flow Area for Heat Transfer; ft^2	41.8	53.2	49.19	49.19
Average Mass Velocity; lbs/hr-ft^2	2.30x10^6	2.23x10^6	2.52x10^6	2.52x10^6
Core Coolant Inlet Temp.; °F	543.5	550	555	557
Core Outlet Temp.; °F	613.7	605	605	648.8
Nominal Outlet Temp. of Hot Channel; °F	644.6	650	642.8	648.8
Maximum Clad Surface Temp.; °F	657	657	654	650
Maximum Fuel Temp. at 100% Power; °F	3980	4110	4150	4464
Maximum Fuel Temp. at 112% Power; °F	4280	4430	4400	4720
Minimum DNB Ratio at Nominal Conditions	1.85	1.95	2.21	1.92
Minimum DNB Ratio at Design Transients	1.3	1.46	1.71	1.50

the Midland and Davis-Besse reactors. This reduction in the control rod drive assemblies reduces the total reactivity worth from 10% Δ k/k for 61 CRA's to 8% Δk/k for 49 CRA's.

We have reviewed the design criteria for the proposed reactor and find they are unchanged from previously reviewed B&W systems. The following Table 3.2.1 shows a comparison of several of this plant's core parameters with the Three Mile Island Unit 2 and Midland cores.

In addition to the above movable poison control systems, the design will have 72 fixed burnable poison assemblies. The burnable poison rods will assure that the moderator temperature coefficient is negative at all times. These burnable poison rods will consist of Al_2O_3-B_4C pellets contained in a Zircaloy-4 tube. The fixed poison rods will be designed to withstand all operating loads including those resulting from hydraulic forces and thermal gradients.

The Davis-Besse pressure vessel closure head will have 69 nozzles with only 57 nozzles required for mounting of the control rod drive assemblies.

On the basis that no changes in design criteria have resulted for either normal or accident conditions due to the reduction of the number of CRA's proposed for this plant and that the capability still exists to permit addition of CRA's if required, we conclude that the proposed control rod drive system and burnable poison rods are acceptable for the construction permit stage of review.

TABLE 3.2.1

COMPARISON OF THREE MILE ISLAND UNIT 2,

MIDLAND, AND DAVIS-BESSE CORE NUCLEAR PARAMETERS

	TMI #2	Midland	Davis-Besse
No. of CRA's	69	49	49
No. of Partial Length CRA's	0	8	8
Reactivity Worth of CRA's; % $\Delta k/k$	10	8	8
Fuel Enrichment; %	2.29/2.64/2.9 Zone	2.30/2.30/2.64 Zone	2.32/2.32/2.68 Zo
Minimum Movable CRA Worth; % $\Delta k/k$	-5.6	-4.5	-4.5
Maximum Worth of Stuck Rod; % $\Delta k/k$	-3.0	-2.5	-2.5
No. of fixed burnable poison assemblies	Not specified	72	72
Type of fuel assembly	Canned Type	Canless Type	Canless Typ
Reactivity worth of partial length CRA; % $\Delta k/k$	-------	0.2 to 0.4	0.2 to 0.
Effective Multiplication			
Cold, zero power, no burnable poison rods	1.302	1.271	1.276
Hot, zero power, no burnable poison rods	1.247	1.218	1.222
Hot, rated power, no burnable poison rods	1.229	1.200	1.204
Hot, rated power, with burnable poison rods	------	1.115	1.119
Hot, equil., xenon with burnable poison rods	1.192	1.084	1.088
Minimum Available CRA Worth % $\Delta k/k$			
All CRA Inserted	-8.4	-7.0	-7.0
One CRA Stuck Out	-5.4	-4.5	-4.5
Minimum Hot Shutdown Margin			
All CRA Inserted	-6.2	-5.8	-5.8
One CRA Stuck Out	-3.2	-3.3	-3.3
Soluble Boron Poison Requirement(ppm)			
Cold, zero power, no CRA in, K_{eff} 0.99	1820	1290	1316
Hot, rated power, Xe + Sm equil., no CRA in	1360	680	711
Boron Concentration of ECCS (ppm)	2270	2270	1800
Transient Xenon Control % $\Delta k/k$	1.4	0.8	0.8

3.3 Increased Rated Power

The Davis-Besse nuclear steam supply system is designed for a rated power level of 2633 MWt. This power level is an increase of approximately 7.4% above previously approved B&W systems rated power levels of 2452 MWt. In Section 3 of the PSAR, the applicants present the information and changes in the Davis-Besse core design to support this power increase. There have been no changes in the Davis-Besse design criteria established for core performance from the previously approved B&W systems.

Three principal design changes result in the capability to increase the rated power level for the core without reducing the previously established safety margins. These three changes are: (1) installation of burnable poison rods which assures a negative moderator temperature coefficient throughout the core life; (2) modification to the Mark II canless type fuel assembly which reduces the flux peak ratio of the worst fuel pin to the average fuel pin from 1.10 to 1.06; the enthalpy peaking factor is reduced from 1.18 to 1.04 due to removal of the fuel element can; and (3) the removal of the internal vent valves will result in an increase of core flow by about 4.6% since in the plant design that uses vent valves, one vent valve is assumed to fail to seat properly and results in core bypass flow.

The following table shows a comparison of some of the calculated core parameters for the Three Mile Island Unit No. 2, Midland, and the Davis-Besse plants.

TABLE 3.3

COMPARISON OF THREE MILE ISLAND UNIT 2, MIDLAND, AND DAVIS-BESSE REACTOR CORE PARAMETERS

	TMI#2	Midland	D-B
Type of fuel assembly	Canned	Canless	Canless
Rated Power	2452 MWt	2452 MWt	2633 MWt
Hot Channel Max/Avg Heat Flux Ratio	3.24	3.12	3.06
DNBR (W-3) @ 112% rated power	1.30	1.71	1.50
DNBR (W-3) @ 100% rated power	1.68	2.21	1.92

The above data show that the design parameters for the Davis-Besse reactor core are not significantly different from other similarly approved plants. Further, in view of the completion schedule for these plants (several B&W designs will be in operation by 1971), there will be sufficient opportunity to assess the adequacy of the B&W design with respect to nuclear, thermal and hydraulic performance.

3.4 Elimination of Vent Valves

The Davis-Besse nuclear steam supply system is the first B&W design which will not have internal vent valves. The proposed design has eliminated the need for these vent valves by elevating the steam generator relative to the pressure vessel. The proposed design elevations will eliminate the possibility of preventing core flooding for a cold leg break due to a water column causing back pressure above the core.

We have reviewed the proposed design elevations and conclude that they are adequate to eliminate the need for core barrel vent valves.

3.5 Reactor Internals

For normal design loads of mechanical, hydraulic and thermal origin, including the operating basis earthquake and anticipated transients, the reactor internals will be designed to function within the stress intensity criteria of Section III of the ASME Boiler and Pressure Vessel Code.

All internals components are designated as Class I seismic items, and will be designed to withstand loads resulting from a combined design basis earthquake and loss-of-coolant accident. Strain limits for the internals under this combined load will be held to less than 20% of the uniform ultimate strain for this material (304S.S.) corresponding to an elastically calculated stress limit of not greater than 2/3 of the ultimate tensile strength. Allowable deflection limits will generally be within 50% of loss-of-function deformation limits. We consider these design limits to be acceptable.

The applicants have referenced topical report BAW 10008, Parts 1 and 2, as outlining the methods of analysis to be employed for the internals and fuel assemblies under loss-of-coolant and design basis earthquake loadings. This report was previously submitted as applicable to the components in skirt-supported reactor vessels; however, the nozzle-supported reactor vessel for this plant requires a different dynamic model to determine seismic response. In addition, changes in plant arrangement and elimination of internals vent valves will result in different blowdown

loadings than given in the topical report. The proposed nozzle-supported vessel design is similar to previously acceptable designs, and we and our consultants will review the topical report covering the detailed analysis as part of the post-construction permit review.

3.6 Vibration Testing

The major core and core support components have been analyzed to provide assurance that the limits on vibratory excitation under steady state and transient conditions are not exceeded. These analyses have considered inlet flow impingement and turbulent flow, as well as natural frequency calculations, to establish that a factor of at least two exists between conditions of possible resonance and excitation frequencies.

The applicants will include provisions for confirmatory vibration testing in the design of this plant. This testing may include the installation of vibration instrumentation for preoperational tests and detailed inspection of reactor internals after cold and hot functional tests. The lead plant of this design, which will be instrumented for vibration, is yet unspecified; however, the nuclear steam system supplier, Babcock & Wilcox, stated that such tests will be completed prior to the completion of Davis-Besse and will identify potential problem areas, if any, for this plant. In Amendment No. 6 the applicants have indicated the design of the core components will not preclude their removal for inspection or vibration testing following preoperational testing. We conclude that this commitment for confirmatory vibration testing is satisfactory at this stage of review.

4.0 REACTOR COOLANT SYSTEM

4.1 General

The reactor coolant system is designated as a Class I (seismic) system and is required by its design criteria to withstand normal operating loads of mechanical, hydraulic, and thermal origin (including all design transients) plus operating basis earthquake loads (OBE) within appropriate code allowable stress limits.

The applicants are currently reviewing the industry codes, code cases, and addenda that will be applicable to the design of pressure vessels, piping, valves, and pumps within the reactor coolant pressure boundary for this plant. This review is being conducted at our request in order to establish the applicants' extent of compliance with the AEC Proposed Rules on Codes and Standards.

The applicants state that earthquake loads have been determined by dynamic analyses. We and our seismic design consultant have reviewed the analytical techniques used to establish earthquake loads and conclude they are adequate.

In Amendment No. 6 the applicants have addressed the concern of the staff and ACRS regarding the seismic design organization, responsibilities, documentation, and auditing to assure the adequacy of the seismic design. We have reviewed the applicants' response and find that it is acceptable.

4.2 Other Class I (Seismic) Mechanical Equipment

All welding procedures and welders concerned with the fabrication of pumps and valves will be qualified to Section IX of the ASME Boiler and Pressure Vessel Code.

Hydrostatic tests of pump casings, valve bodies and valve seats are to be in accordance with the ANSI B16.5 and MSS SP-61 code and standard and will be witnessed by the applicants' representative.

The proposed inspection program for pumps and valves requires independent review of the physical and chemical test data for pressure boundary materials as well as independent review of nondestructive examinations of valve bodies, valve bonnets, and pump casings. These requirements result in a fabrication and inspection program which contains the essential elements of the ASME Code for Nuclear Pumps and Valves. We find these requirements acceptable.

All equipment for the engineered safety features will be designed to withstand the design basis earthquake without loss of function. This equipment will include seismic design requirements which will be based on, or checked against, the outcome of the structural dynamic analysis and will include, where necessary, the dynamic feedback of flexible equipment. We find this approach acceptable.

4.3 Reactor Vessel

The Davis-Besse reactor vessel will be designed and fabricated in accordance with the 1968 edition of the ASME Boiler and Pressure Vessel Code, Section III, Class A, plus the Summer 1968 Addendum and Code Case 1332-4. The vessel will be essentially identical to vessels for Russellville, Crystal River 3 and 4, Three Mile Island Unit 1, Rancho Seco Unit 1, Midland, and Oconee Unit 1.

Based on our review, we have concluded that the proposed design and fabrication specifications and procedures are acceptable.

4.4 Electroslag Welding

The Davis-Besse steam generators will have secondary side shell courses fabricated from plates joined by longitudinal weld seams using the electroslag process. We have reviewed the planning for compliance with the ASME Code Case 1355, which allows the use of the electroslag weld process, as compared to the additional testing performed to qualify the electroslag welding process used in the fabrication of the Dresden 2 and 3 reactor vessels. Based on this comparison and our review of the submitted information and considering the steam generator shell a Class C vessel, we have concluded that the electroslag welding process, as planned, should result in acceptable weldments for the steam generators.

4.5 Reactor Vessel Material Surveillance Program

The estimated end-of-life neutron fluence for the reactor vessel is 2.4×10^{19} nvt, based on a 40-year service lifetime and a load factor of 0.80; however, Babcock & Wilcox has selected a design value of 3.0×10^{19} nvt. B&W has verified its calculational model, the Code, through three separate nuclear experiments. These experiments are outlined and referenced in Section 4 of the PSAR. On the basis our past reviews of other PWR plants and experimental verification performed by B&W, we find the proposed neutron fluence values acceptable for the Davis-Besse Station.

B&W has not yet submitted its revised topical report on reactor vessel material surveillance programs (as mentioned in the Midland ACRS report). The answer to Question 4.9 in Amendment No. 3 furnished sufficient information for us to evaluate the proposed program. The program contains six

core mid-plane capsules and is consistent with programs that have been accepted on previous PWR plants and meets or exceeds our requirements with respect to total number of capsules, archive material provisions, and material chemistry documentation. We will evaluate the quantity and types of specimens in the capsules and the withdrawal schedule when B&W submits the revised topical report. B&W, at a technical meeting on January 21, 1970, stated that they are exceeding our requirements in these respects.

4.6 Leak Detection

The reactor coolant pressure boundary leak detection systems for Davis-Besse are similar to those provided at Midland and other PWR plants. These systems, which include air particulate monitoring, radiogas monitoring, humidity detection, and containment sump level monitoring, are considered acceptable. The Davis-Besse array of instrumentation is redundant, diverse, and provides timely alarms. Although the applicants have not yet documented the sensitivities of the leak detection systems, our discussions with the applicants indicate adequate sensitivities will be provided. We conclude that the proposed leak detection systems are acceptable.

4.7 Missile Protection and Flywheel Integrity

The applicants, in answer to Question 12.3.7 in Amendment No. 5, calculated that a turbine missile could not penetrate the containment, the control room, or the fuel pool roof. Our review of the calculations indicates a conservative approach. The applicants stated that the facility design is such that a turbine missile will not cause a LOCA or prevent shutdown of the reactor. The auxiliary building which contains engineered

safety features is to be designed with a minimum roof thickness of 18 inches of reinforced concrete as missile protection. This same protection extends to tornado generated missiles. We find the applicants' missile design protection programs acceptable.

Toledo Edison has presented an extensive internal missile protection study and design criteria in answer to Question 4.8 in Amendment No. 3. We conclude the study is reasonable and criteria are acceptable, and if properly implemented, will result in acceptable protection of the primary system, other vital systems, and the containment liner from missile hazards.

The primary pump-motor flywheels proposed for Davis-Besse are manufactured by Westinghouse and are similar to those used in many other PWR plants. The flywheels are fabricated of A533B steel plate and subjected to extensive quality controls. On the basis of our previous evaluations, we conclude that the flywheel design and construction are acceptable when supplemented by the proposed preservice and inservice surveillance requirements.

4.8 Inservice Inspection

Toledo Edison is applying the rules of Section XI of the ASME Code, Rules for Inservice Inspection of Nuclear Reactor Coolant Systems, as the basis for determining the areas and components of the primary system requiring access for future inservice inspections. The applicants will include inservice inspection requirements for the primary pump-motor flywheels in their program.

The applicants, in answer to Question 4.7 in Amendment No. 5 of the PSAR regarding inservice inspection of the vital systems other than the reactor coolant pressure boundary, have indicated that access provisions will be included in the design arrangements to facilitate inservice inspection of areas and components of these systems.

We conclude that Toledo Edison's programs for inservice inspection are acceptable. The final program will be reviewed at the operating license stage.

5.0 CONTAINMENT AND CLASS I STRUCTURAL DESIGN

The containment for the station consists of two structures; a low leakage steel shell containment vessel surrounded by a reinforced concrete shield building. The double containment concept is similar to the containment planned for the Kewaunee and Prairie Island plants.

5.1 Nuclear Structures

5.1.1 Environment and Foundation Structural Considerations

Design basis loadings arising from seismic, wind, and tornado forces are considered in the design. Detailed study of regional and site geology and seismology has resulted in selection of peak foundation acceleration values for the operating bases earthquake (OBE) and design bases earthquake (DBE) of 0.08g and 0.15g, respectively. All Class I structures, except for the borated water and diesel oil storage tanks, are located on competent rock. The seismic design has been reviewed by our Consultants, John Blume and Associates, and found to be acceptable.

5.1.2 Structural Description, Design Criteria and Loads

The primary containment is a cylindrical steel shell structure with a spherical dome and ellipsoidal bottom. The structures interior to it are of massive reinforced concrete construction. The primary containment is surrounded by a reinforced concrete shield building. The auxiliary building is of reinforced concrete column, beam and slab construction, while the turbine building has structural steel framing.

5.1.3 Containment Structural Design and Design Analysis

The primary containment design is based on the rules of the ASME Boiler and Pressure Vessel Code, Section III, Class B. The design analysis will include consideration of seismic and local effects. The structures interior to the containment will be designed for applicable thermal, local pressure,

jet and missile effects, including the conventional dead and live loads. The containment structure is being designed to currently accepted standards. The shield building is also being designed to acceptable standards in a manner similar to structures already accepted (e.g., Prairie Island).

5.1.4 Seismic Input

The seismic design response spectra will be obtained from the smoothed upper average of the Helena spectra. The upper Helena time-history will be adjusted in amplitude and frequency to significantly envelop the response spectra specified for the site. We and our consultants conclude that the seismic input criteria proposed by the applicants provide an acceptable basis for seismic design.

5.1.5 Seismic Structural Design and Design Analysis

The applicants indicate that careful attention will be given to the seismic design of the facility. The principal structures of the facility (containment interior structure, containment, shield building) are being analyzed by the modal analysis-response spectrum technique. Floor level response spectra are being developed from earthquake time-history spectra to determine proper inertial force levels for piping, equipment, and instrumentation. The applicants will select the floor level response spectra following a comparison of a fixed base mass model and a spring supported mass model which considers the range of dynamic properties of foundation material. If the results differ for both models, the most conservative values will be used in the design.

5.1.6 Tornado Protection Design

The facility is designed to assure safe shutdown considering the effects of tornadic wind forces and tornado-driven missiles. In addition,

the facility design includes protection against the effects of tornadoes which could either cause a LOCA or result in the uncontrolled release of radioactivity. Appendix 5A-2 of the PSAR lists these buildings and systems.

The design loads are based upon a tornado having a tangential wind velocity of 300 mph and a translational velocity of 60 mph. In addition to these wind loads the design will withstand a 3 psi pressure drop in 3 seconds.

We have reviewed the tornado protection design and conclude it is acceptable.

5.1.7 Structural Materials, Construction Techniques and Quality Control

The structural materials, construction techniques, and quality control are similar to those successfully applied on other recent nuclear facility construction projects (e.g., Oconee). Particular attention has been given to corrosion in the selection of materials in contact with or embedded in the foundation because of the fairly corrosive nature of the soil at this site.

5.1.8 Structural Acceptance Testing

Structural acceptance pneumatic testing for the primary containment is in accordance with ASME Code provisions. The applicants will install a strong-motion accelerograph and three peak recording accelerometers to assist in seismic motion recording in the event of an earthquake.

5.1.9 Conclusion

Based on the criteria for structural design and construction presented by the applicants in the PSAR and information furnished during meetings with the applicants in the course of our review, we conclude that the structural features of the facility will be designed and constructed so as to provide acceptable safety margins.

5.2 Containment Design

The Davis-Besse primary containment houses the reactor primary coolant system, core flooding tanks, let-down coolers and normal and emergency ventilation systems. A cylindrical reinforced concrete shield building with a hemispherical roof surrounding the containment vessel provides secondary containment for leakage of fission products from the containment vessel during a hypothetical accident, biological shielding during normal operations and under hypothetical accident conditions, and environmental protection to the primary containment vessel for adverse atmospheric conditions and external missiles.

The primary containment vessel and the shield building will be supported on a concrete foundation founded on competent bedrock. Above this foundation, there will be no structural ties between the containment structures to restrict differential movement.

The annular space of 4.5 feet between the structures is adequate for construction operations and periodic inspections.

Table 5.2 compares the principal design parameters for the containment structures for Davis-Besse with those of Prairie Island.

Penetrations common to both the primary and secondary containments are attached to the steel primary containment structure and pass through the shield building with sufficient clearance to accommodate lateral and radial differential movement of the two structures. A flexible membrane seal closure at the outside of the shield building penetration seals the containment structures annulus. Hot process lines in the annulus are enclosed within guard pipes vented back to the primary containment so that a rupture of the hot process line will not affect the annulus pressure. We find that penetration design

TABLE 5.2

COMPARISON OF CONTAINMENT DESIGN PARAMETERS

Parameter	Davis-Besse	Prairie Island
Primary Containment		
Net Free Volume	2,800,000 ft^3	1,320,000 ft^3
Inside Diameter	130 ft	105 ft
*Design Internal Pressure	36 psig @ 264°F	42 psig @ 268°F
*Maximum Internal Pressure	40 psig @ 264°F	46 psig @ 268°F
Maximum Shell Thickness	1-1/2 inches	1-1/2 inches
Design Leak Rate (% of internal free volume in 24 hours)	max. 0.5% @ 40 psig - 264°F	max. 0.5% @ 46 psig - 268°F
Test Pressure	45 psig	52 psig
Design External Pressure	0.50 psig	0.80 psig
Shield Building (Secondary Containment)		
Annulus Free Volume	465,000 ft^3	400,000 ft^3
Thickness Wall Dome	2.5/2.0 ft	2.5/2.0 ft
Design Leak Rate	approx. 1% of annulus volume @ 1/4 inch of H_2O in 24 hours	10% of annulus volume @ 1/4 inch of H_2O in 24 hours

*In accordance with ASME Boiler and Pressure Vessel Code, Section III, Class B.

is consistent with practices used in other recently-approved plants and is acceptable for Davis-Besse.

Protection from loss of function to the primary containment structure from missles generated inside the structure, including pipe whip, is provided by concrete shield walls, concrete operating floor, and control rod drive mechanism shield. The applicants indicate that the detailed protection analysis will be made during the detailed station design to implement protection criteria.

5.3 Design Pressure

The applicants have calculated the primary containment pressure response to a loss-of-coolant accident (LOCA) from various breaks in the primary system piping. The controlling design internal pressure of 36.0 psig was calculated by the applicants for a 3.0 ft^2 double-ended pipe break, assuming minimum engineered safety system operation (i.e., one core flooding tank and one diesel generator available to operate either the containment vessel sprays or air coolers). The 3.0 ft^2 primary coolant rupture results in the maximum pressure transient due to additional core energy transferred to the containment during the primary system blowdown. Our independent calculation of the controlling containment pressure for a LOCA is 36.8 psig for a 3.0 ft^2 double-ended pipe break. The applicants have provided a 10% margin on the 36.0 psig design peak pressure to establish a 40 psig/264°F maximum internal pressure/temperature for the primary containment vessel.

We conclude that the primary containment design pressure selected together with the accident assumptions used are acceptable; however, because of the concern about coincident failure of the secondary system due to pipe whip (see Section 5.7), additional margin may be required.

5.4 Leak Rate

The containment vessel will be tested at the conclusion of construction, and after all penetrations have been installed, to verify that the leak rate associated with the maximum internal pressure of 40 psig and accident temperature and air conditions does not exceed 0.5% of the containment vessel internal free volume in 24 hours. The acceptance leak rate for the initial leak test of the containment at 40 psig will be 0.25%/day.

Leakage from the primary containment, following a containment isolation signal, will be collected in the secondary containment structure (shield building) with a filtered ventilation system designed to remove iodine before discharge to the atmosphere.

Following a successful overpressure test at 45 psig, a leakage test on the primary containment will be run at 40 psig using the "reference system method" to determine the leak rate. The tests are similar to those conducted on previously-approved plants, and are acceptable for Davis-Besse.

Leakage from the shield building and penetration rooms, estimated at 1% of the free volume per day at 1/4 inch of water, will be measured using a test fan, damper and calibrated duct orifice. This leak rate is much lower than the secondary containment leakage estimated for previous similar plants (10% of free volume per day at 1/4 inch of water). In addition, the secondary containment will be tested at anticipated positive pressures to measure outward leakage at +6 inches of water.

5.5 Ventilation System

The shield building and penetration room emergency ventilation and filtration system are designed to collect and filter potential leakage from the primary containment vessel after actuation by a containment isolation

signal (CIS). The system provides a negative pressure within the secondary containment following the loss-of-coolant accident.

Primary containment leakage is filtered through roughing, high efficiency particulate, and two charcoal filters in series prior to discharging through the station vent at an elevation of about 240 feet.

The applicants have calculated that with offsite power available, the ventilation system will be operating as early as 12 seconds after CIS; for a loss of offsite power, the maximum time for the vent system to reach full capacity is estimated to be 46 seconds. The applicants estimate that the initial positive pressure buildup will last less than 2 minutes before a negative pressure is achieved. Maximum positive pressure in the annulus will be limited to 6 inches of water.

Our dose analyses are based on a negative pressure being achieved within 2 minutes and we believe that starting times up to 60 seconds will achieve this result.

The applicants propose to test the ventilation system periodically for operability and performance.

We conclude from our evaluation of the secondary containment ventilation system, including filter performance and its similarity to the previously approved Prairie Island system, that it is acceptable and the doses resulting from any postulated accident situations will be well within the 10 CFR Part 100 guidelines.

5.6 Intake Structure

The intake structure houses the service water pumps, fire water pumps, and circulating water pumps. The enclosures for the service water pumps are designed to withstand the load requirements for Class I structures and provide tornado missile protection.

Approximately 700 feet of the intake canal directly in front of the intake structure will be designed and constructed as a Class I structure. The design will provide a 9.5 million gallon storage water reservoir to maintain a means of decay heat removal assuming isolation of the plant from Lake Erie through failure of the canal structure beyond the Class I section. This emergency cooling water reserve is sufficient to provide necessary heat removal for approximately 73 days under either normal shutdown or accident conditions.

We conclude that the structural provisions for the intake structure, canal, and emergency cooling water reservoir are acceptable.

5.7 Pipe Whip Protection

The applicants have specified, in response to question 4.11 of Amendment 6, the design criteria proposed for protection against failure of systems and components from effects of pipe whip. These criteria include protection of the containment building against failure caused by the pipe whip effects of the main steam and feedwater line failures. However, the criteria do not include protection against failure of the steam lines due to rupture of a primary system line. Thus, we have indicated to the applicants that to prevent the containment design pressure of 40 psig from being exceeded, the secondary system should be protected from failing due to pipe whip forces as a result of primary system rupture.

Based on our review of the proposed pipe whip criteria set forth in Amendment 6, we conclude the proposed criteria would be acceptable, provided the secondary system (steam generator, steam lines, etc.) are protected against

failure from the pipe whip forces of a primary coolant pipe break or the containment designed to withstand the LOCA coincident with a secondary loop blowdown without exceeding the design pressure. This matter is unresolved.

5.8 Pressure Vessel Cavity

The applicants have indicated that the largest possible pipe break size within the reactor pressure vessel cavity is 3.0 ft^2. The calculated cavity pressure corresponding to this break is about 55 psig. We have informed the applicants that we will require the pressure vessel cavity to be designed to withstand pressure effects equivalent to those from a double-ended pipe rupture, 14.1 ft^2, within the reactor cavity.

This matter is unresolved at this time. Our position on this matter is consistent with Millstone 2.

5.9 Post-Accident Hydrogen Control

In Amendment 3, the applicants submitted a response to our question (12.6.9) concerning post-accident hydrogen evolution and control. The applicants have made a preliminary evaluation of the hydrogen concentration in the containment following a LOCA and have acknowledged that a potential may exist for the hydrogen concentration in the containment to reach the flammability limit of 4.1 v/o. The applicants have indicated that they are currently undertaking three tasks regarding the controlling of hydrogen concentrations within the containment building. These three tasks are; (1) review of the containment air circulation capability to see if it will prevent the possibility of local high concentrations of hydrogen from occurring, (2) analyze and develop methods for controlled purging to the

atmosphere, and (3) study the possible use of catalytic recombiners for controlling hydrogen concentration.

The applicants indicate a 3.5 v/o hydrogen concentration within the containment building following a LOCA would not be reached until about 180 days after the accident.

We have reviewed the hydrogen concentration within the Davis-Besse containment using the following assumptions:

1. g (H_2) equal to 0.5 H_2/100 ev (core & sump)
2. 5% clad reaction
3. No aluminum reaction since spray is borated water with no additives.
4. TID-14844 fission product releases.

Using the above assumptions the hydrogen concentration within the containment building would not reach the lower flammability limit of 4.1 v/o until about 83 days following the accident.

The applicants have indicated they are continuing to study the hydrogen evolution and control problem and will provide the necessary means to meet requirements which may be established for an acceptable means of controlling the hydrogen within the containment building.

The applicants have been informed that we do not consider the matter resolved at this time, and control of the containment hydrogen concentration following a LOCA by purging to the atmosphere would be acceptable only as a backup to some other means of hydrogen control such as catalytic recombiners. We conclude that the applicants' program to cope with hydrogen evolution and control within the containment is adequate.

6.0 ENGINEERED SAFETY FEATURES

6.1 Emergency Core Cooling System (ECCS)

The ECCS design for this plant is similar to the systems' proposal for previously reviewed and approved nuclear facilities which utilize the B&W designed nuclear steam supply system. Since the emergency high pressure injection system will not be used to provide primary coolant makeup during normal operation, the system will use only two high pressure injection pumps instead of three pumps as in previously reviewed B&W designs. The major equipment making up the ECCS consists of (a) two high pressure injection pumps (up to 500 gpm per pump), (b) two low pressure injection pumps (up to 3000 gpm per pump), a d (c) two core flooding tanks (940 ft^3/tank). The minimum ECCS consists of two core flooding tanks, one high pressure injection pump, and one low pressure injection pump. The sizing of the ECCS is based upon the system providing adequate core cooling for a primary system rupture of the smallest size ($\sim$0.05 ft^2) up to a double-ended hot leg pipe rupture (14.1 ft^2) without exceeding a maximum fuel clad temperature of 2300°F. The ultimate core power level of 2772 MWt was used to establish the ECCS sizing requirements.

The actuation of the ECCS will be initiated in the event of (a) an abnormally low reactor coolant system pressure of 1500 psig or (b) a containment pressure of 4 psig. Either of these two signals will automatically increase the high pressure injection flow to the reactor coolant system. Termination of the high pressure injection requires operator action. The core flooding tanks are designed to start discharging into the reactor pressure vessel when the primary coolant system pressure reaches 600 psig. The gas

overpressure in the core flooding tanks and the sizing of the piping is sufficient to ensure reflooding the core within 25 seconds after the largest pipe rupture (14.1 ft^2). The low pressure injection system also functions as the normal decay heat removal system. Actuation of this low pressure injection system (3000 gpm per pump) is initiated by either (a) a primary coolant system pressure of 200 psig or (b) a 4 psig containment pressure.

The initial source of coolant for the ECCS system is the borated water storage tank which has a capacity of 360,000 gallons of borated water with a concentration of 1800 ppm boron. When the level of the borated water storage tank reaches a pre-set low level, the valves controlling suction to the ECCS pumps will automatically switch to a recirculation mode taking suction from the containment sump.

The piping and pumps for the ECCS are sized and located to assure the net positive suction head (NPSH) requirement is met even if the containment pressure were to drop 0.5 psi below atmospheric pressure.

The heat transfer capability of the three fan coolers at the saturation temperature corresponding to containment accident pressure is in excess of the core heat generation rate at the time the suction for low pressure injection is switched from the borated water storage tank to the recirculation mode of operation.

As a result of our review of the North Anna (Westinghouse) plant, several motor-operated valves have been identified which could experience active failure and degrade the emergency core cooling to an unacceptable level. We have reviewed the Davis-Besse ECCS and find that the motor-driven valves used to isolate the core flooding tanks during depressurization below 600 psig could

experience an active failure and degrade the ECCS below an acceptable level. We plan to require these motor-drive isolation valves to be protected against active failure or be redundant. We have informed the applicants of our requirement and they have agreed to review the active failure of this valve and provide an adequate resolution of our concern.

We will report the status of this problem prior to the ACRS subcommittee meeting scheduled for August 4, 1970.

The emergency core cooling system (ECCS) for Davis-Besse has been analyzed using a modified version of the FLASH-1 code. This code describes the reactor coolant system by the use of two control volumes for the primary loops and one for the pressurizer. The system is grouped into the two control volumes on the basis of temperature distribution. Resistances to flow are calculated by dividing the reactor coolant system into 24 regions and calculating the volume-weighted flow resistance for a given rupture location based on normal flow resistances. The model incorporates a variable velocity steam bubble rise model.

Recent results obtained with the use of the multi-node SATAN code by Westinghouse for their Indian Point 2 evaluations and by INC with their multi-node RELAP code have raised questions concerning the ability to reliably predict the thermal-hydraulic response of a reactor core during blowdown following a large cold-leg rupture with the analytical methods presently being used. In view of these concerns we intend to require the applicant to provide additional evidence, obtained with the use of suitable multi-node analytical techniques, to verify that the ECCS system is capable of limiting core temperatures to acceptable levels.

6.2 Containment Spray System

A containment spray system has been provided to remove heat from the containment atmosphere in the event of a loss-of-coolant accident and thereby assure that the peak containment pressure will not exceed 40 psig. The containment spray system is sized to furnish sufficient containment cooling with the containment air circulation fan coolers inoperative. The spray system consists of two half-capacity pumps (1300 gpm per pump), two half-capacity spray headers and necessary valves and piping.

Actuation of the spray system requires a high-high containment pressure (20 psig) coincident with an emergency injection-actuation signal.

The two half-capacity spray systems with a total of 150×10^6 Btu/hr heat removal capacity and with minimum ECCS heat removal capacity will reduce the containment pressure after the pressure peak for all primary pipe ruptures to less than 20 psig within 20 minutes after the occurrence of the rupture.

The two half-capacity spray pumps are located in separated compartments at the lowest elevation to assure NPSH for the pumps during and following the LOCA. Ventilation of the rooms containing these pumps will be accomplished using the shield building emergency ventilation system following an accident to assure any airborne activity is filtered prior to release to the environment.

The borated water supplied to the containment spray system will not contain any iodine removal additive such as sodium hydroxide or sodium thiosulfate.

On the basis of our review of the proposed design and design criteria, we conclude that the containment spray system design is acceptable and the peak containment pressure will be limited to approximately 90% of the containment design pressure of 40 psig.

6.3 Containment Fan Coolers

In addition to the containment spray system, the containment will have three half-capacity fan cooler systems. Any two of the three fan coolers have a heat removal capacity equivalent to the containment spray system. The containment minimum heat removal requirements are met by the following systems: 50% spray capacity plus one fan cooler, two sprays, or two fan coolers. Each fan cooler consists of a finned-tube cooling coil and a direct-drive fan. Cooling water for the fan cooler is supplied by one of three service water pumps which are located at the intake pump house. The service cooling water is a once-through system. During normal operation, the discharge line of each fan cooler has a modulating control valve in parallel with a stop valve. The modulating control valve provides automatic control of the containment temperature during normal operation. In the event of a LOCA, the emergency injection actuation signal will open the stop valve and permit full water flow through the fan coolers.

Excessive leakage of the unborated fan cooler water is annunciated in the control room by high water level in the fan cooler condensate sump and isolation valves can be actuated from the control room.

The cooling coils are similar to the design used in the Haddam Neck and Palisades plants.

Based on our review of the design and design criteria, we conclude that the fan cooler system design is acceptable.

6.4 Containment Isolation System

The containment isolation system will isolate all piping and penetration through the containment which are not required for operation of the engineered safety features system. The design criteria for isolation capability minimize

the leakage through all penetrations not serving accident-consequence limiting systems by using a double barrier concept. These double barriers take the form of closed piping systems inside and outside the containment in combination with various types of isolation valves.

The isolation of the containment is actuated by a Containment Isolation Signal (CIS) which is initiated by three logic systems: (1) high pressure emergency injection signal closes noncritical systems isolation valves, (2) critical systems isolation valves for component cooling water are delayed and closed by the same actuation signal that initiates the containment spray system, and (3) the containment purge system isolation is initiated by either the high pressure emergency injection signal or a high radiation signal. The high radiation signals are initiated from a gas and particulate sensor located in the normal ventilation stack or from the same type sensor located in the containment.

Containment isolation valves will have the capability to be tested individually by manual actuation. The instrumentation and control circuits are such that no single failure will prevent containment isolation and each isolation valve is designed to close on loss of power or air supply.

All valves and equipment which are intended to be isolation barriers are protected against potential missiles and jet forces both inside and outside the containment.

Based upon our review of the containment isolation system and its similarity to previously-approved systems, we conclude that this plant's containment isolation system is acceptable.

6.5 Emergency Ventilation System

The emergency ventilation system will consist of two independent, full-capacity fan-filter systems. The containment isolation signal (CIS) will start

both emergency fan-filter systems following a LOCA. The emergency ventilation systems will reduce the air pressure within the shield building annulus to a negative pressure of 1/4 inch of water in less than two minutes. After the pressure in the shield building annulus is reduced to a negative 1/4 inch of water, the volume of air discharged to the stack is limited to only that required to meet the inleakage. The remaining capacity of the fans is returned to the shield building annulus.

Areas which could receive iodine activity due to leakage from systems in direct communication with the containment atmosphere are vented through the emergency ventilation system. The charcoal filter system used in the emergency ventilation system will reduce the iodine dose at the exclusion distance and low population zone boundary, for the design bases accident, to well below the 10 CFR Part 100 guideline values.

In response to question 5.7 in Amendment 6 the applicants have indicated all areas ventilated by the emergency ventilation system. The emergency ventilation system will provide iodine removal for possible through-leakage from the containment building. In addition, the emergency ventilation filter system will be used to ventilate the fuel storage pool area and provide iodine filtration capability.

We conclude that the emergency ventilation system is acceptable.

7.0 INSTRUMENTATION, CONTROL, AND POWER SYSTEMS

7.1 General

Our review encompassed the auxiliary electric power system and the plant protection system instrumentation. The Commission's proposed General Design Criteria (GDC), as published in the Federal Register on July 11, 1967, and the proposed IEEE Criteria for Nuclear Power Plant Protection Systems (IEEE 279) dated August 1968, served, where applicable, as the bases for evaluating the adequacy of the designs.

The reactor protection instrumentation and control systems, as well as the instrumentation which initiates and controls the engineered safety features, (ESF) are substantially the same as those found acceptable in the Three Mile Island Nuclear Power Station Unit 2, Docket No. 50-320. The Three Mile Island design of these instrumentation and control systems was reviewed and considered by the ACRS at its July 1969, meeting. The following discussion concerns only those features of the design which differ from those of Three Mile Island and for which new or additional information has been received.

7.2 Instrumentation and Control

In the Three Mile Island #2 design, three instrument channels are provided to monitor each variable required to initiate ESF. These instruments are arranged in a two-out-of-three (2/3) coincidence logic for initiation of the protective action. The Davis-Besse design uses four instrument channels arranged in a two-out-of-four (2/4) coincidence

logic in initiating a protective action. We have concluded that this modification provides added redundancy and is acceptable. The applicants have stated that the systems will satisfy the requirements of IEEE 279.

7.2.1 Diverse ECCS Initiation Signals

In the Davis-Besse design, emergency coolant injection is initiated by either low reactor coolant pressure or by high containment pressure; however, reactor trip is only initiated by low reactor coolant pressure. Since the analyses of the effectiveness of ECCS take credit for a reactor trip, we informed the applicants that the high containment pressure signal should also initiate reactor trip. They have agreed to provide diversity to assure reactor trip on initiation of ECCS. The applicants have been advised that the instrumentation selected must survive the accident environment.

This problem will receive detailed study during final design. We conclude that the applicants' commitment is satisfactory for the construction permit review.

7.2.2 Operation With Reactor Coolant Pumps Out of Service

The applicants have stated that operation of the reactor with less than the full complement of Reactor Coolant Pumps (4) does not require adjustment of reactor protection trip settings to more conservative values. The applicants have stated that the reactor would be scrammed on loss of a second coolant pump and further stated that on loss of a single pump, no adjustment of set points is required for safety.

The applicants also state that operation on two reactor coolant pumps will be provided for in the design of the reactor protection system. After reactor shutdown, manual adjustment of trip setpoints will be made in accordance with Technical Specification requirements to permit operation at reduced power. The permissible power level and trip setpoints will be established during the detailed design of the system. We intend to also require during detailed design, written operating procedures and hydraulic and thermal analyses for both one and two loop operation. We will also review the same concerns expressed for Oconee on one loop operation before accepting this mode of operation.

We conclude that these commitments are satisfactory for the construction permit review.

7.2.3 Control Rod Assembly Interlocks

In the Davis-Besse design several interlocks have been provided to restrict control rod reactivity addition and/or reactivity addition rate. The applicants have been advised that those interlocks whose failure would permit fuel damage must be designed to the requirements of IEEE 279. The applicants have stated that fuel failure will not result from exceeding an operating limit normally protected by a single control rod assembly interlock.

We conclude that this criterion is satisfactory and will continue to review the matter during construction.

7.3 Electric Power

7.3.1 Offsite Power

Power for the Davis-Besse Nuclear Power Station will be supplied from a single 345 kV switchyard which is connected to the CAPCO grid by three 345 kV transmission lines. Two of the three lines are installed on the same right-of-way for seven miles of their length; the lines are supported on independent structures set far enough apart to avoid the possibility of a structural collapse of one line causing an outage of both lines. The third transmission line is routed independently along a separate right-of-way.

Initially the switchyard will be arranged in a five breaker ring-bus configuration, with two full capacity main buses. The applicants plan to modify this design to a full breaker-and-one-half arrangement during the detailed design phase. Either arrangement meets the single failure criterion. Two redundant 125 volt d-c protective relaying systems are provided, with each system being composed of a separate battery, charger, and cables. In the event of a single failure, a loss of control power will not be experienced.

The CAPCO grid stability has been analyzed and the applicants have reported that the loss of this unit or the loss of the largest single generating unit on the interconnection will not result in the loss of offsite power to the plant.

During normal operation, power is supplied by the station auxiliary transformer which is connected to the main generator 25 kV isolated phase bus. Each of the secondary windings of the auxiliary transformer is connected to the two 13.8 kV main buses. In the event that power is lost on either 13.8 kV bus, the system will initiate fast automatic transfer to reserve sources (startup transformers). Each of the two startup transformers is supplied from different 345 kV bus sections and is the reserve source for only one redundant emergency bus. If either startup transformer fails, the other startup transformer can supply minimum engineered safety features without further switching. However, with manual switching the remaining transformer is capable of supplying all engineered safety features.

We conclude that the offsite power system is acceptable.

7.3.2 Onsite Power

The design of the onsite power system utilizes the splitbus concept. The redundant engineered safety feature equipment trains are divided between two 4.16 kV buses such that either one will supply minimum safety requirements. One diesel-generator is connected to each bus. A third 4.16 kV bus with additional ESF loads may be connected to one of two aforementioned emergency buses. The equipment connected to this third bus is not needed to meet redundancy requirements, but is utilized for replacement of redundant counterpart equipment removed

from service. Its connection to either bus is effected by operator action from the control room. Redundant interlocks which meet the single failure criterion are provided to preclude the possibility of inadvertently paralleling the emergency generators.

The two redundant diesel-generators will be located in separately ventilated rooms of a seismic Class I structure. Auxiliary systems for these machines are redundant and independent; the fuel oil supply is adequate for the operation of minimum engineered safety features for at least ten days. The applicants have stated and will document in the forthcoming amendment that the continuous rating of the diesel generators will be selected such that the connected loads will not exceed the continuous rating for 8000 hours.

The d-c battery system for the station includes two redundant 250/125 volt bus systems. Three chargers are provided for each system, two of which are normally connected (one each to the ± 125 volt d-c buses) and a spare charger is assigned to each system to serve as a reserve to either of the chargers normally connected. Redundant feeders from each 125 volt d-c bus are connected to four 125 volt d-c distribution panels. Normally the preferred feeder breaker at each panel will be closed while the alternate breaker is left open. Single failures in the battery systems should not prevent supplying power required for minimum ESF.

The batteries are mounted on racks and housed in separate rooms which are designed to satisfy seismic Class I standards. Additionally, each of these rooms will be provided with independent ventilation systems. The 125 V d-c distribution panels provide power to the ESF and reactor protection system instrumentation utilizing the split-bus concept.

We have concluded that the design of the onsite power system is acceptable.

7.4 Seismic, Radiation, and Environmental Testing

7.4.1 Seismic Testing

The reactor protective system, emergency electric power system and instrumentation and controls for the engineered safety features and shutdown cooling systems are designed as seismic Class I systems. The systems are designed to function before, during and after the maximum peak seismic acceleration. Specifications for components of these systems will contain the requirements for the submission of test data, appropriate operating experience or calculations which will substantiate that the components will not suffer loss of function under the design basis seismic loadings.

7.4.2 Radiation Testing

The applicants have stated that the design criterion for all electrical cable is that it shall not fail when subjected to the accident radiation doses after the normal long term operating conditions. In

addition, all material and equipment associated with safety related systems will be required by specification to perform their functions when exposed to the 40-year integrated and DBA dose. Tests will be performed or existing test data will be used to show that these items are satisfactory for use in the specified environment.

7.4.3 Environmental Testing

The applicants have identified instrumentation and equipment including cables, located within containment, which are requried to operate during and subsequent to an accident. The applicants have stated that "type tests" have been or will be required to show satisfactory operation in an equivalent environment of pressure, temperature, and humidity for the time period required.

We agree with the applicants' criteria with respect to the testing program and conclude that the commitment is satisfactory for this review.

7.5 Cable Design, Selection, Routing, and Identification

The applicants have documented the criteria for cable design, selection, and routing. With adherence to the criteria, the probability of loss of redundant channels of protection from a single cause such as fire will be adequately low. The criteria for identification of safety-related circuits and components are acceptable.

7.6 Turbine Overspeed Protection

The turbine-generator will be designed by General Electric and will use a 1800 rpm tandem compound, four-flow exhaust, indoor turbine unit. This unit will use a electrohydraulic control system with an overspeed trip of 110%. The emergency trip system is an independent, redundant control system providing protection against turbine overspeed, loss of condenser vacuum, thrust bearing wear and generator electrical faults. A mechanical centrifugal device plus the emergency trip system provide a 111% overspeed trip.

We conclude this overspeed protection proposed is acceptable, but we nonetheless plan to follow the design aspects during construction.

8.0 AUXILIARY SYSTEMS

8.1 General

The following systems are the same as those provided on other B&W PWR facilities and found to be acceptable: Chemical and Volume Control, Residual Heat Removal, Service Water, Spent Fuel Cooling, Sampling, Vent and Drain and Ventilation. The locations of these systems are within the Class I auxiliary building which provides protection against tornado missiles. All leakage or spillage from these systems will be confined to the auxiliary building and collected in sumps.

Our evaluations of the other auxiliary systems; the Radwaste System, Spent Fuel Storage, Intake and Discharge Canals, Auxiliary Feedwater System, and Boric Acid Injection System are summarized in the following sections.

8.2 Radwaste System

8.2.1 Liquid Radwaste

The major source of radioactive liquid waste results from the primary coolant which is removed and stored during reactor startup operations and collection of leakage from the primary coolant system during operation of the facility. The liquid radwaste treatment system includes a degasifier, primary demineralizers, evaporators, mixed bed demineralizers and filters. This system is designed to reduce the concentrations of all radionuclides except tritium in the water. The PSAR (p.2.4-2) lists the decontamination factors as follows:

Kr and Xe	6.86×10^4
Cs, Mo and Y	6.86×10^7
Cr, Mn, Co, and Fe (insoluble corrosion products)...	6.86×10^5
All others	6.86×10^9

The liquid radwaste system for the Davis-Besse facility will have the capability of treatment and reuse of the major portion of the liquid waste generated from normal plant operation without discharging radioactivity into Lake Erie. The applicants have indicated that under circumstances wherein the storage of liquid waste becomes inadequate or as the cleanup demineralizer requires recharging, liquid waste may be discharged by a batch type process. The activity level of this batch type discharge of liquid waste will be well below 10 CFR Part 20 limits after being discharged into the discharge canal for additional dilution.

The batch discharging of liquid wastes will be interlocked with the main condenser cooling pumps to assure a positive dilution flow of the discharge canal water to Lake Erie.

Other low level radioactive liquid wastes generated in the plant which contain oil or detergents are collected in a separate storage tank monitored and either released into the discharge canal or cycled through an evaporator to reduce activity levels before being discharged to the main condenser discharge flow.

The borated water tank and the primary water tank may contain significant quantities of tritium. The tanks are above ground, outside, and adjacent to the facility and they are not protected against tornadoes. The primary water storage tank is not protected against earthquakes. We asked the applicants to determine the maximum offsite doses that could result from tank ruptures, earthquakes, floods, or tornadoes. The applicants provided an analysis (pgs. 2.4-4 and 11.4-1 of the PSAR) which concludes that a dose to the whole body less than 0.02 rem would result from the abrupt release of tank water to Lake Erie. The highest dose would occur at Camp Perry (2.8 miles away) where the nearest potable water intake is located. We determined that an offsite airborne dose of 0.05 rems to the whole body could result from an incident using the following assumptions: (a) the larger tank (360,000 gallons) was full of water with the highest tritium concentration assumed by the applicant (9 μc/cc), (b) 10% of the water in the tank became air-

borne, (c) atmospheric diffusion resulted from a Type F condition, a 1 m/sec wind speed, and a building wake effect, and (d) the breathing rate of the individual at the site boundary was $3.47 \times 10^{-4} m^3/sec$.

8.2.2 Gaseous Waste System

Gaseous waste is collected from the liquid waste by a degasifying process and from the cover gas of liquid waste storage tanks. The gaseous waste is collected and compressed for storage in three decay tanks for 30 - 60 days to permit decay of the radioactivity prior to discharging to the atmosphere through a HEPA filter. The gaseous waste is continuously monitored during discharge to the atmosphere with automatic shutoff activity levels to assure 10 CFR Part 20 limits are not exceeded.

8.2.3 Radwaste Systems Monitoring and Structures

Samples of radwaste gases and liquids can be collected at points within and at the end of the radwaste systems. Instruments will monitor and record the radiation from the waste being discharged, and they will activate alarms and control valves if the radiation is high.

The radwaste system is in the auxiliary building which is a Class I structure. Surge and gas decay tanks are designed to Class I standards. All other radwaste system equipment and piping is Class II. Liquids released by the rupture of Class II radwaste tanks will be contained within the Class I auxiliary building.

8.2.4 Conclusion

On the basis of our review, the information presented in the PSAR, and the design and design criteria established for liquid, gaseous, and solid radioactive waste system and associated instrumentation, we conclude that these systems will provide adequate protection against accidental releases sufficient to meet the 10 CFR Part 20 requirements and decontamination factor capability to reduce the radioactivity released to the environs to as low as practicable.

8.3 Spent Fuel Storage

The spent fuel storage pool is located in the Class I auxiliary building. The auxiliary building will have a minimum reinforced concrete wall and roof thickness of 18 inches. The fuel cask loading pool is separated from the fuel storage area to prevent dropping of the cask in the fuel storage area. The applicants have indicated in Table 2 on page 12.3.7-3 of the PSAR that the fuel storage pool will be protected against fuel element damage or loss of water from the turbine-generator missile and possible scabbing which might result from this missile.

The ventilation of the fuel pool storage area will use the emergency ventilation filter system to assure iodine filtration capability. The thyroid doses resulting from a fuel handling accident are discussed in Section 9.0.

We have reviewed the proposed design and conclude that it is acceptable.

8.4 Intake & Discharge Canal

The extreme low water level for Lake Erie is calculated to be 555.85 feet above mean sea level. The intake canal will be dredged to a depth of 554 feet MSL on shore and 557.6 feet MSL offshore.

The intake and discharge canals will be designed as Class II structures except for approximately a 700-foot length of the intake canal connected to the intake pump structure, which will be designed to Class I seismic requirements. The intake and discharge canals are approximately 7000 feet long, 200 feet wide at bottom and have a bank slope of 1:3. Normal flow through these canals will be approximately 685,000 gpm.

The Class I portion of the intake canal provides 10 million gallons of cooling water between the water level of 560 feet MSL and the bottom of the service water pump suction inlet. This 10 million gallons of cooling water plus 250,000 gallons of condensate storage water is sufficient to bring the plant from power operation to cold shutdown and remove the decay heat for more than 73 days. This 73-day cooling capability covers either normal or accident conditions.

At a water level of 562 feet MSL, a low water level alarms in the control room and at a water level of 560 feet MSL, the main condenser circulating flow is automatically shut off. The system that provides automatic shutoff will be defined during final design.

The intake canal will have screens to prevent debris from entering the plant cooling systems and will have deicing capability.

The applicants have indicated, and we agree, that adequate cooling water for removal of decay heat could be re-established within about 14 days in the event normal lake supply was cut off in the intake canal. Each of the three Class I (seismic) service water pumps and two Class I piping systems can provide the cooling requirement for removing decay heat.

The applicants have indicated in the event the thermal discharge limits for the plant to Lake Erie become too restrictive, they will consider the use of cooling towers. There are no site features or features of the plant design which would preclude the use of cooling towers as an alternative to the proposed intake and discharge canals.

We have reviewed the intake and discharge canal design criteria, and we conclude that the design will provide an adequate cooling capability in the event Lake Erie cannot be used as a heat sink due to either a seismic event or extreme low lake water level.

8.5 Auxiliary Feedwater System

The auxiliary feedwater system will consist of two redundant 100% capacity steam turbine-driven feedwater pumps. Each pump is sized to meet the steam generator feedwater requirement to remove the decay heat 40 seconds after a reactor trip from the ultimate power level of 2772 MWt. Backup to the auxiliary feedwater system water supply (two condensate storage tanks 250,000 gal/tank) is provided by the fire

protection system and the service water system. All active components of the system are accessible for inspection during plant operation. The system will be periodically tested during operation.

We have reviewed the auxiliary feedwater system and conclude that it is acceptable.

9.0 ACCIDENT ANALYSES

9.1 Summary of Results

We have evaluated the potential consequences of six design basis accidents. Our calculated offsite doses are given in Table 9.1. Computation of the fission product inventory for all of the accidents analyses are based on the assumption that the core had been operated at the ultimate power level of 2772 MWt except for the steam line rupture accident and steam generator tube rupture accident which were evaluated for the rated power level of 2633 MWt. These latter two accidents do not cause any fuel failures and the radioactivity release consequences are dependent only on the radioactivity in the primary coolant and the primary to secondary leakage assumed. The assumptions used in our evaluation of these six accidents are presented in the following sections.

TABLE 9.1

ACCIDENT CONSEQUENCES

Accident	Two-Hour Site Boundary Doses-Rem (2400 feet)		LPZ Course of Accident Dose -Rem (2 miles)	
	Thyroid	Whole Body	Thyroid	Whole Body
Loss of Coolant	140	10	160	7
Refueling	65	2	14	<1
Gas Decay Tank Rupture	--	6	--	1
Steam Line Rupture	2	<1	<1	<1
Steam Generator Tube Rupture	20	4	4	1
Rod Ejection	70	<5	40	<5

9.2 Loss-of-Coolant Accident

The loss-of-coolant accident analysis is based on the following assumption. The applicants' assumptions in analyzing this accident are indicated in parentheses.

1. TID 14844 fission product releases from core (same)
2. Iodine platecut factor of 2 (same)
3. Charcoal filter efficiency for 2 filters in series for all iodine 95% (95%)
4. Primary containment leak rate 0.5%/day for the first 24 hours and 0.25%/day for the remaining course of the accident. (2.5%/day)
5. Ground release (applicant assumes release at 100 feet)
6. Meteorology
   a. Pasquill Condition F (same)
   b. Wind Speed 1m/sec (2 m/sec)
   c. Building wake effect (same)
7. Breathing rates
   a. 3.47×10^{-4} m^3/sec 0-8 hours (same)
   b. 1.75×10^{-4} m^3/sec 8-24 hours (same)
   c. 2.32×10^{-4} m^3/sec after 24 hours (same)

We have based our evaluation of the loss-of-coolant accident on a containment leak rate of 0.5%/day and assumed any leakage from the primary containment is released unfiltered while the shield building annulus has a positive pressure. Our calculated doses are presented in Table 9.1.

9.3 Refueling Accident

The applicants have evaluated this accident assuming 56 fuel pins experience cladding failure and subsequent release of the gap activity. We have evaluated the potential doses for the refueling accident based on the following assumptions (applicants' assumptions are shown in parentheses):

1. All 208 fuel pins of a fuel element are damaged (56)
2. 20% of noble gases in fuel pins released (gap activity)
3. 10% of iodines in fuel pins released (gap activity)
4. Decontamination factor of 10 for iodines retained in pool water (1000)
5. Charcoal filter efficiency 95 (same)
6. Max/Avg peaking factor 1.8 (1.6)
7. Decay time prior to accident 72 hours (same)
8. Breathing rate for 2-hour dose 3.47×10^{-4} m^3/sec (same)
9. Meteorology
   a. Ground release (same)
   b. Wind speed 1 m/sec (2 m/sec)
   c. Pasquill Condition F(same)
   d. Building Wake Factor 1300 m^2 (same)

9.4 Gas Decay Tank Rupture

We have analyzed the radiological consequences of a gas tank rupture by assuming the tank contains a quantity of noble gases equivalent to that which we calculated would be present in the primary system operating with 1% of the core fuel rods experiencing clad defects. We used the following assumptions in the analysis:

1. Entire noble gas content of one primary coolant volume (11,440 ft^3) is in the decay tank 24 hours after reactor shutdown. The noble gas inventory assumed corresponds to the applicants' proposed design basis for plant operation.
2. Entire contents of the decay tanks are released to the atmosphere.
3. Average energy of fission products released is 0.7 Mev/disintegration.
4. Meteorology is same as refueling accident.

9.5 Steam Line Rupture

As in Midland, the applicants have analyzed the consequences of a double-ended 36-inch steam line rupture. In this analysis, an analog-hybrid computer program was used to study the transient characteristics of the reactor coolant system and the steam generator. This model included a detailed analog description of the secondary side of the steam generator, energy balances for the principal steam generator components and the entire reactor coolant system including the pressurizer. The model also includes reactor kinetics trip system logic and fuel pin simulation with Doppler and moderator temperature coefficient feedback. These analyses indicate that the reactor will trip six seconds after failure and will not return to criticality thereafter. The maximum thermal power during the transient is 106% of rated power (2633 MWt). Thus, no fuel damage is expected.

We have estimated the radiological consequences of a steam line rupture assuming (1) the reactor trip associated with the steam line failure results in a loss of offsite power and (2) the plant has been operating with one gpm primary-to-secondary leakage and that this leakage occurs in the steam generator

not affected by the steam line rupture, and therefore, depressurization in the primary system must be accomplished by boiloff in the steam generators not affected by the steam line failure, (3) primary coolant contains the applicants' design basis fission product inventory for operation, (4) secondary system activity corresponding to that which would result in a calculated dose of 1.5 rem to the thyroid at the site boundary, assuming loss-of-offsite power and loss of condenser flow, and (5) breathing rate and meteorology are the same as for the refueling accident.

The above analysis of the steam line break accident is based upon the reactor not returning critical due to blowdown of a single steam generator. In response to question 12.7 the applicants have indicated the analysis and consequences of the steam line break with no operator action assumed. This analysis indicates the reactor would return to criticality and go subcritical after 20 seconds with no fuel damage occuring. We are currently reviewing this accident in greater depth for the Oconee operating license. Since the criterion for no fuel damage is unchanged, the doses calculated for the accident would remain as stated in Table 9.1.

9.6 Steam Generator Tube Rupture

We have analyzed the results of a steam generator tube failure assuming a double-ended rupture of one steam generator tube occurs with unrestricted discharge from each end. The applicants have stated that this will trip the reactor on low pressure in about eight minutes. Isolation of the affected steam generator can be effected since the operator can identify the problem from (1) the low reactor coolant pressure, (2) the pressurizer level, and (3) the early

increase in radioactivity in the steam line from the affected steam generator. Under these conditions it will require 15 minutes to cool the reactor system down to the temperature corresponding to the saturation pressure at which the safety dump valve is set. Following the safety valve release, the reactor will trip on low pressure in about eight minutes. Assuming the operator takes no action until the reactor trips, the total time required to isolate the secondary side of the affected steam generator is about 23 minutes. The double-ended rupture of a steam generator tube is well within the capacity of the core cooling system. Thus, core damage is not assumed to occur.

We have calculated the offsite doses assuming (1) the tube rupture occurs concurrent with the loss of offsite power resulting in a loss of condenser flow, thus preventing the use of the condenser for decontamination, (2) the operator does not isolate the affected steam generator, (3) 5930 ft^3 of the primary system coolant blows down to the steam generator and is released in 1.7 hours (applicants' data), (4) an iodine partitioning factor of 10 for halogens released from the primary coolant, (5) the plant has been operating with primary coolant activity corresponding to the applicants' design basis fission product inventory, (6) secondary system activity corresponding to that which would result in a calculated dose of 1.5 rem to the thyroid at the site boundary, (7) all noble gases and halogens on the secondary side are released within two hours and (8) breathing rate and meteorology are the same as the refueling accident.

We are currently reviewing this accident and the assumptions used to calculate the resulting thyroid and whole body doses. Current PWR's being reviewed for operating licenses are required in the technical specifications

to limit the primary coolant activity to a value less than the failed fuel criterion used in assumption (5). Thus, in the event that a decontamination factor (4) of 10 cannot be justified, the resultant dose can be reduced by reducing the permissible activity level in the primary coolant.

9.7 Rod Ejection Accident

The applicants have analyzed the rod ejection accident for beginning-of-life and end-of-life conditions at both 2772 MWt and zero power. The maximum worth of a single control rod at power with no xenon present is 0.46% delta k/k. The maximum worth of a single control rod at hot zero power critical conditions is 0.56% delta k/k. The applicants have analyzed the transients resulting from ejected rods of various worths using the KAPP-1 digital computer program. This code contains a two-dimensional heattransfer model, and a point kinetics physics model a check on the KAPP-1 calculation, the rod ejection accident was also analyzed for a limited number of cases using the WIGL-2 digital computer program, a one-dimensional space-dependent kinetics code. The WIGL-2 calculations performed have been for the full power, beginning-of-life case.

The analyses performed by the applicants indicate that no DNB or fuel damage will result from a rod ejection accident at zero power critical. These analyses indicate that the peak fuel enthalpy for the hottest rod would be approximately 75 calories per gram.

For a rod ejection accident at 2772 MWt, the point kinetics model predicts a peak fuel enthalpy of approximately 170 calories per gram, a peak thermal power of 126% at beginning-of-life, and 4.1% of the fuel rods experience DNB. (For comparison, the space-dependent kinetics model predicts a peak fuel enthalpy of 130 cal/gm).

We have estimated the potential offsite doses resulting from this accident, assuming (1) 4.1% of the fuel rods in the core perforate releasing 100% of the noble gases and 50% of the iodine in these rods to the primary coolant, (2) the plant has been operating with primary coolant activity corresponding to the applicants' design basis fission product inventory, (3) the 1 gpm primary-to-secondary system leakage in the steam generator, six hundred pounds of water leak to the secondary side within two hours and 1600 pounds leak to the secondary side during the course of the accident, (4) loss of offsite power requiring heat rejection by boiloff to the atmosphere in the steam generators, and (5) boiloff in the steam generators results in the release of equilibrium secondary activity and the activity in the primary system leakage to the atmosphere with a partition factor of 10 for iodine. The resultant calculated doses are presented in Table 9.1.

As previously stated, the space-dependent and point-kinetics results have been compared for a limited number of rod worths for the beginning of life ultimate power level case. This comparison indicates that the space-time dependent solution yields a lower peak fuel enthalpy for rod worths of 0.5% delta k/k or less and, therefore, it is conservative to analyze the consequences of ejection of the maximum worth rod at power (0.46% $\Delta k/k$) using the point-kinetics method.

10.0 CONDUCT OF OPERATION

10.1 Technical Qualifications

The Davis-Besse Power Station will be co-owned by The Toledo Edison Company and The Cleveland Electric Illuminating Company. The Toledo Edison Company will have the responsibility for the overall design, construction and operation of the Davis-Besse plant. The Bechtel Company (Gaithersburg) will perform the architect-engineering services and the Bechtel Corporation will provide construction management services. Babcock and Wilcox will supply the nuclear steam supply system. The turbine generator will be supplied by the General Electric Company.

The applicants have experience in the design, construction, and operation of fossil-fueled electric power stations and have participated as members of the Atomic Power Development Associates (APDA) in the design, development and operation of the fast breeder reactor, the Enrico Fermi Atomic Power Plant.

The Toledo Edison Company's Engineering staff consists of 90 employees holding engineering degrees of various disciplines.

The Bechtel Corporation has been actively engaged in design and construction of nuclear plants and is currently engaged in the design and construction of 23 BWR and PWR nuclear power plants.

Babcock and Wilcox is currently engaged in the design, construction and installation of 10 pressurized water nuclear steam supply systems.

On the basis of the above considerations and our contact with project personnel during our review, we have concluded that the applicants and their contractors, collectively, are technically qualified to design and construct the proposed Davis-Besse Nuclear Power Plant.

10.2 Operating Organization

The operating organization for the Davis-Besse Plant will consist of 57 full-time employees. Functional responsibilities are divided into four groups, each headed by a supervisor reporting to the Plant Superintendent. The Operations Group consists of 25 men who will handle operations. The three remaining groups consist of Maintenance (13 men), Technical (9 men), and Chemical and Health Physics (5 men). The applicant proposes a normal shift of five operations personnel, who will also receive training in chemistry and health physics. The five-man shift will consist of one shift supervisor (senior operator's license), one plant control operator (licensed operator), one reactor operator (licensed operator), one major equipment operator (unlicensed), one auxiliary equipment operator (unlicensed). In addition, six other members of the operating organization will have senior operators' licenses. We have concluded that the proposed shift size and composition are acceptable.

10.3 Training

Toledo Edison has submitted the scope and schedule of each phase of a plant operations training program which is acceptable. Most of the men selected for training will come from existing positions within the company and will have experience in fossil-fired plants. The training program consists of six phases. Phase one consists of nuclear theory that will be taught at Toledo Edison. Phase two, PWR observation, will consist of three months at an operating PWR. The Plant Superintendent, Operations Engineer, Maintenance Engineer, all technical staff engineers, technical leader, and all shift supervisors will take part in this phase. Phase three, PWR Technology, will be presented by Babcock and Wilcox at their Lynchburg facility. Phase three will require a period of six weeks for all personnel who received phase two training. The plant control operators and the reactor operators will receive this course prior to reporting for on-site training. Phase four, PWR operations, will be conducted at the Lynchburg facility. This course consists of six weeks of classroom and operational training on the simulator, two weeks of training on the Lynchburg Pool Reactor and four weeks of shift operation on the simulator. The course will be completed by the Plant Superintendent, Operations Engineer, Technical Engineer, Results Engineer, one general engineer, technical leader, and all shift supervisors. Phase five, On-the-Job-Training, will be at Davis-Besse and will last for approximately eight months. Phase six, Specialist Schools and

Training, will include courses in nuclear engineering, fuel management, instrumentation, radiation protection, radiochemistry and maintenance of major equipment. Based on the foregoing, we have concluded that the training program proposed by the applicants is satisfactory.

10.4 Review Boards

The applicants have defined the composition and responsibilities of two review groups. The station review board will be composed of Davis-Besse's Station Superintendent, Supervisor of the Operations, Technical, Health Physics and Maintenance groups who will meet periodically to review normal operations, written procedures, abnormal occurrences or departure from the technical specifications. The Company Nuclear Review Board will be composed of company engineers and outside consultants as required to audit and review station operations and activities of the Station Review Board. It is anticipated the Vice President, Power Group; Chief Mechanical Engineer; General Superintendent, Power Production; Davis-Besse Station Superintendent, Station Electrical Engineer; and Nuclear Engineer in the Mechanical Engineering Division will be members of this board. Specific responsibilities include the review of any major abnormal occurrence or departure from the technical specifications. We conclude these review groups are adequate at this time for review and audit of proposed operation of the facility.

10.5 Emergency Plan

Toledo Edison submitted in Amendment 3 to the PSAR their outline for an Emergency Plan. This plan is to cover emergencies such as fire, medical injury and illness, radiation and contamination accidents, and other emergency conditions that may result from operational malfunctions, natural disasters, and civil disturbances. The following Station Emergency Plan is acceptable at this time.

Outside agencies such as the U.S. Coast Guard, Ohio State Highway Patrol, State of Ohio Department of Health, Ottawa County Civil Defense, local police and fire departments, the AEC, and area hospitals and medical clinics, will be called upon as needed. These agencies will be familiar with the Davis-Besse Nuclear Power Station, with the emergency plan and with their expected role in assisting with an emergency situation. The services of professional consultants in the areas of medical radiation and radiation accident control will be obtained if deemed necessary.

Procedures will be provided to maintain the emergency plan up to date and responsive to personnel and organization changes in participating organizations and agencies.

Initiation of designated alarms will alert the Control Room to the possible existence of an accident in which the release of radioactive material could occur to the environment.

After receiving one or more of the designated alarms, the Operator will evaluate the conditions producing the alarms, and if instrumentation indicates that radiation has resulted which presents a danger to

station personnel or the surrounding population, he will sound the radiation emergency plan alarm.

To evaluate the radiation levels on-site, readings will be taken from the station radiation monitoring system in the Control Room, and a monitoring team will survey the station and immediate environment. Appropriate records will be kept of the results of the surveys.

To evaluate the radiation levels off-site, a monitoring team will survey designated areas reflecting accident and post-accident wind conditions. Appropriate records will be maintained.

If radiation monitoring teams have reported high radiation levels on the off-site survey and these radiation levels are above the average projected dose limit recommended by the Federal Radiation Council, and radiation levels are increasing, appropriate local, state and federal authorities will be notified. These authorities will be informed that a major release of radioactive material occurred at the Davis-Besse Station; evacuation of the low population zone may then be performed by prearranged methods.

In the event a medical injury occurs that necessitates hospitalization before decontamination, action will be taken for a preliminary survey of the contaminated individual, and for the alerting of designated hospital authorities providing them with pertinent information so that applicable phases of the emergency plan may be initiated.

The emergency plan will make provisions for the obtaining of additional personnel from within the Company, from other nuclear power stations, outside agencies, and appropriate vendors if required in the event of an emergency.

Part of the emergency plan will pertain to the requirements of a plan to return the station site and environment to an acceptable standard of radiation safety.

10.6 Initial Tests and Operations

The applicant's proposed initial tests prior to normal operations have received preliminary review. Davis-Besse personnel will conduct all testing. Technical direction and assistance will be provided by Babcock and Wilcox Company. The testing program will receive a more comprehensive review at the OL stage.

10.7 Conclusions

On the basis of our review of the information submitted, we conclude that the applicants' organization and their contractors are technically competent to design and construct the proposed plant. The applicants' proposed training, testing and emergency preparedness programs are acceptable.

11.0 QUALITY ASSURANCE

We have reviewed the quality assurance program presented by the applicants for the design, construction and operation of the Davis-Besse Nuclear Power Station with regard to the applicants' stated objective of meeting the intent of the AEC proposed "Nuclear Power Plant Quality Assurance Criteria," Appendix B of 10 CFR 50. The Davis-Besse Quality Assurance Program is described in the PSAR Volume I, Appendix 1B, Amendment No. 2.

The Toledo Edison Company (TEC) will have the ultimate responsibility for the QA program. Bechtel acting as agent for TEC will be responsible for the day-to-day implementation of the QA program. Babcock and Wilcox Nuclear Power Generation Department (NPGD) will have the day-to-day responsibility for the nuclear steam supply system.

The TEC organization has an experienced Quality Assurance Engineer (QAE) reporting directly to the Vice President in charge of the Power Group, which has the ultimate responsibility for the Davis-Besse plant. The QAE has prepared a Toledo Edison Quality Assurance Manual which complies with the AEC criteria and provides written procedures for TEC's implementation of the QA program. The manual incorporates by reference the QA manuals of the principal contractors.

A description of the duties of the Toledo Edison QAE and Engineering Staff are addressed in the PSAR. The QAE will have the authority to stop work in the event of nonconformance with drawings, specifications and/or procedures established for major critical structures, substructures, systems and subsystems.

Bechtel as the architect-engineer and construction manager has written six manuals to provide instructions, guidelines, procedures, check lists, and appropriate documentation forms to assure implementation of the QA program. All design drawings and calculations originating within the Bechtel organization will receive at least one internal independent review and check prior to releasing to TEC where it receives an additional review and approval before issuance for procurement. The six manuals to be used to supplement the Toledo Edison Quality Assurance Manual are: Bechtel Procurement Department Inspection Manual, Bechtel Field Inspection Manual, Bechtel Field Procurement Manual, Davis-Besse Field Procedure Index, Davis-Besse Engineering Procedures Manual, and the Davis-Besse Construction Management Procedure.

TEC and the Bechtel Quality Assurance Coordinator will audit the Bechtel QA program to assure that it is being implemented.

Babcock & Wilcox as supplier of the nuclear steam system (NSSS) has established a Quality Assurance Program to cover the areas of NSS design, manufacturing, procedures, specifications and erection. The B&W NPGD Quality Assurance group administers the QA program and reports

directly to the Vice President in charge of the NPGD. B&W implements the B&W QA program by use of standards and written procedures. Auditing of the B&W QA/QC to assure the QA program is being implemented will be performed periodically by TEC assisted by Bechtel when requested by TEC.

The Toledo Edison Company, assisted by the two prime contractors, Bechtel and B&W, will provide specifications, procedures, and auditing necessary to assure that the subcontractors used for construction and manufacturing of critical structures and components will meet the requirements and intent of the proposed "Nuclear Power Plant Quality Assurance Criteria," Appendix B of 10 CFR 50 throughout the design, construction, and operation of this plant.

The Division of Compliance has made an initial inspection of the applicants' quality assurance program and has made verifications of the program in the following areas:

a. The applicants have developed a manual for guidance of their staff in performance of QA functions. The manual incorporates by reference the QA manuals of the principal contractors.

b. The applicants intend to rely on their contractors for day-to-day implementation of the requisite QA and QC activities.

c. The applicants plan to do limited design review with their own staff, but have developed an audit program to provide assurance that their contractors are implementing the QA activities described in their internal manuals.

d. The applicants have developed a schedule and procedures for audit which appear acceptable.

e. At the present time the applicants' QA staff consists of a single individual. The applicants plan to augment their staff "as the need arises," but have no definitive schedule for doing so.

Compliance believes that the current program is acceptable, and will examine the timeliness of future staff additions during the continuing inspection program.

Based on our discussions with the applicants, Bechtel, and B&W and the information contained in Appendix 1B of the PSAR, we conclude that the overall Quality Assurance Program for the Davis-Besse Nuclear Power Station is acceptable.

12.0 RESEARCH AND DEVELOPMENT

Specific areas requiring research and development prior to completion of the design are summarized below.

12.1 Core Stability and Power Distribution Monitoring

The B&W program on xenon oscillations consists of the following analyses:

1. Modal analysis
2. One and two dimensional digital analysis
3. Three dimensional analysis

The results of the modal analysis have been submitted as Topical Report BAW-10010, "Stability Margin for Xenon Oscillations - Modal Analysis." One dimensional digital analysis will be used to determine the validity of the modal analysis approach. The results of the one and two dimensional digital analyses will be complied as a topical report shortly. The three dimensional digital analysis results will be submitted for our review later this year. The entire program is scheduled for completion well before the scheduled startup of the Davis-Besse facility. This program is required to establish the stability characteristics of the core and demonstrate that the partial length control rod system can control any core instability to assure the desired operation of the plant.

We are not sure if the B&W program will be able to demonstrate that sufficient information can be derived from external detectors alone. We have reservations that a known power distribution can be achieved after the reactor has been operated because axial burnup is not uniform. Further, fuel or control rod replacement or errors on fuel element position

or enrichment may also perturb the flux distribution. In addition, we believe that insufficient experience exists with operation of large power reactors to ascertain when out-of-core detectors must be recalibrated. If the planned R&D program does not produce completely convincing evidence that the out-of-core detection system is sufficient, we will require that a minimum number of in-core detectors, properly positioned throughout the core, be available to the operator at all times when the reactor is operating at power.

12.2 Fuel Rod Clad Failure

B&W has initiated a study of fuel clad failure mechanisms associated with a loss-of-coolant accident which includes an evaluation of existing data and scoping tests to obtain data on potential failure mechanisms. These tests consist of the following:

1. Eutectic formation - test data indicate that a liquid eutectic forms at temperatures above 1700°F at the point of contact between the stainless steel spacer grid and the zircaloy clad. The applicants report that no interference with emergency core cooling would result from this eutectic mechanism. The work in this area is complete and the results will be reported shortly.
2. Brittle failure - clad specimens heated to 2300°F and quenched in room temperature water did not experience brittle failure. A

reduction in ductility occurred but strength was not reduced. The applicants have reported that work in this area is complete. We are awaiting a report on the experiments.

3. Clad swelling - single rod tests have been run to investigate the effects of clad swelling, the heatup rate, internal pressure, hydriding of zircaloy, and preoxidation of the cladding. The applicants have reported that results to date indicate that: (1) the low pressure tests produced a larger increase in diameter due to greater ductility at higher temperatures, (2) the lower hot rates produce greater swelling, (3) the hydrogen content plays no major role in the event of diametral swelling, (4) the preoxidation generally resulted in less swelling (On this basis, it was decided to delete the systematic study of preoxidation effects on swelling), (5) the perforations were randomly located on the cladding, (6) the failure time is extremely short, (7) the first point of swelling was not necessarily the one which ruptured or swelled the greatest. Multirod experiments are planned using oven heating. A 4x4 rod bundle will be heated in an oven with the four central rods pressurized.

The analytical study of fuel clad failure is in the planning stage. This program will consist of evaluation of the axial

as well as radial temperature distribution thoughout the core. The changes in flow channel resistance to flow was calculated and incorporated into the channel analysis. The program is designed on the basis that the major unknown is the amount and location of flow blockage that could result from clad deformation in a loss-of-coolant accident.

The multi-pin tests will provide data to determine the possibile interaction between pins undergoing a temperature excursion. These data, coupled with the data resulting from completion of the FLECHT Program (Full Length Emergency Cooling Heat Transfer Test) scheduled for completion this year, will provide further information on the capability of the emergency core cooling system to function as designed.

12.3 Once-Through Steam Generator

B&W has conducted tests on 7, 19 and 37 tube mockups of the once-through steam generator in the following areas: heat transfer and heat capacity, control and dynamic response, structural integrity under normal and accident conditions, vibration, feedwater heating by spray nozzles, tube leakage propagation, and simulated steamline failure tests. This program is complete and is reported in BAW-10002. We are reviewing this report at the present time and have identified areas where further justification must be supplied before we can accept the B&W conclusion that the tests substantiate the design. Our discussions with B&W are being conducted in the course of our review of the Oconee FSAR.

12.4 Control Rod Drive Test

The B&W control rod drive test program to develop the roller-nut type drive has been completed and is reported in Topical Report BAW-10007, "Control Rod Drive System Test Program." We are reviewing this report in the course of our POL review for Oconee. Several areas have been identified to B&W where more details of the tests results should be addressed. Review of the areas is continuing in the course of our review of the Oconee FSAR.

12.5 Self-Power Detector Tests

The B&W research and development program for self-power detectors is completed (longevity testing is continuing) and is reported in BAW-10001, "Incore Instrumentation Test Program." The testing of the self-power detectors has indicated this system is capable of measuring neutron flux with a relative accuracy of ±5 percent in a PWR environment over a three year time span. This device has an inherent slow time constant and is not used in any direct safety actions. As indicated in Section 12.1 of this report, if out-of-core detectors are not capable of detecting core instability, we will require a minimum incore instrumentation when the reactor is operated at rated power. We are reviewing this topical report in the course of our review of the Oconee FSAR.

12.6 Core Thermal & Hydraulic Design

B&W is conducting a continuous research and development program for heat transfer and fluid flow investigations. The experimental programs criteria are developed from the thermal and hydraulic design

limits set forth in Section 3 of the PSAR. A topical report, BAW-10012, "Reactor Vessel Model Flow Test" is currently being reviewed for the Oconee operating license. We will continue to review these matters to assure that sufficient safety margin is available to prevent events which could cause departure from nucleate boiling and subsequent fuel failures. These matters will be followed as a generic concern for the B&W nuclear steam system thermal and hydraulic parameters as indicated in Section 6.1 we will require B&W to reanalyze the thermal response of the fuel clad following a LOCA to assure that the ECCS will provide sufficient cooling to limit the maximum clad temperature to below 2300°F. B&W is currently reanalyzing the LOCA and ECCS capability using a more sophisticated code (Flash 2.5).

12.7 Blowdown Forces on Internals and Core

The stresses and deflection of the reactor internals are being analyzed by B&W for the nozzle supported pressure vessel. This analysis for the skirt supported vessel is reported in topical report BAW 10008, "Reactor Internals Stress and Deflection Due to a Loss-of-Coolant Accident (LOCA) and Maximum Hypothetical Earthquake." The results reported in this topical report are currently being reviewed for the Oconee operating license review. B&W will submit the nozzle pressure vessel topical report and include those additional matters of concern developed during the Oconee review.

12.8 Conclusion

Based on our review of the research and development programs proposed, we conclude that these programs are timely, are reasonably designed to accomplish their respective development objectives, will provide adequate information on which to base analyses of the design and performance, and should lead to acceptable designs for the respective systems.