Safety Evaluation Report

Supp. No. 1
NUREG-0136

U. S. Nuclear Regulatory Commission

related to operation of

DAVIS-BESSE NUCLEAR POWER STATION UNIT 1

Office of Nuclear Reactor Regulation

Docket No. 50-346

April 1977

The Toledo Edison Company and
The Cleveland Electric Illuminating Company

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SUPPLEMENT NO. 1

TO THE

SAFETY EVALUATION REPORT

BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

UNITED STATES NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF

TOLEDO EDISON COMPANY

AND

CLEVELAND ELECTRIC ILLUMINATING COMPANY

DAVIS BESSE NUCLEAR POWER STATION

UNIT 1

DOCKET NO. 50-346

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1.0 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

Since publication of the Davis Besse Nuclear Power Station, Unit 1 Safety Evaluation Report in December 1976, we have received and reviewed several amendments to the Final Safety Analysis Report, held a number of meetings with the applicant, and met with the Advisory Committee on Reactor Safeguards. These events are identified in the Chronology, Appendix A to this supplement. As a result of these actions, the issues identified as outstanding review items in Section 1.7 of the Safety Evaluation Report have been resolved as noted below in Sections 1.7 and 1.8.

This supplement provides (1) our evaluation of additional information received from the applicant since issuance of the Safety Evaluation Report regarding previously identified outstanding review items, (2) a discussion of comments made by the Advisory Committee on Reactor Safeguards in its report of January 14, 1977, and (3) our evaluation of additional or revised information related to new issues that have arisen since the issuance of the Safety Evaluation Report.

Sections of this supplement carry the same numbers as those of the Safety Evaluation Report which they supplement or modify, and except where specifically noted, do not replace sections of the Safety Evaluation Report.

1.7 Outstanding Review Items

Items previously identified as outstanding have been resolved since publication of the Safety Evaluation Report as indicated below. Also, resolution of some of these items require limitations on plant operation and are identified below. New issues addressed since the Safety Evaluation Report issuance are identified accordingly, and for all items, additional discussion is presented in the referenced sections of this supplement.

1.7.1 Resolved Items

- Acceptability of the second year of onsite meteorological data (Section 2.3.4 and Section 15.3).
- Leakage Detection System (Section 5.2.4).

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- Performance of surveillance capsule specimen holder tube based on reactor internals vibration test assessment (Section 5.3).

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- Design modifications to the reactor barrel guide blocks (Section 5.3-new).

- Evaluation of the reactor cavity pressure response analysis (Section 6.2.1).

- Analysis for pressure response of the shield building following a postulated loss-of-coolant accident (Section 6.2.3).

- Leak testing of the check valves in low pressure injection discharge line interface with the high pressure system (Section 6.3.2-new).

- Water hammer surveillance requirement for emergency core cooling system piping (Section 6.3.2-new).

- Evaluation of emergency core cooling system performance considering minimum containment pressure, submerged valves, effects of boron precipitation, and single failure criteria (Section 6.3.3).

- Acceptable net positive suction head of emergency core cooling system in recirculation mode (Section 6.3.4).

- Review of door seals and control room pressure tests to verify control habitability (Section 6.4).

- Review of the safety related electrical logic and schematic diagrams and the verification of the implementation of the design (Section 7.1).

- Engineered safety features actuation system automatic testing system disconnect (Section 7.3.2-new).

- Core flooding tank isolation valves (Section 7.3.5).

- Evaluation of the modified steam and feedwater line rupture control system (Section 7.4.1).

- Inoperable status indication of containment and steam generator isolation (Section 7.5-new).

- Evaluation of separation criteria for redundant safety related electrical cables in trays, wireways, and conduits (Section 7.9.2).

- Evaluation of backup protection and short circuit interrupt tests for containment electrical penetrations (Section 7.10).

- Evaluation of the qualification of safety-related equipment in a postulated main steamline break accident environment (Section 7.7 and Section 6.2.1).

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- The applicant revised their Industrial Security Plan after our issuance of the Safety Evaluation Report. We found the revised Industrial Security Plan to be acceptable. (Section 13.6-new).

- Evaluation of the applicant's financial qualifications to operate the facility (Section 20.0).

1.7.2 Items with Conditions to Operating License

The evaluation of the following items required conditions to the operating license that will require further Commission approval and license amendments before the stated condition can be removed.

- Seismic reanalysis of certain plant systems for a 0.2g safe shutdown earthquake and use of Regulatory Guide 1.60 design response spectra (Section 3.7-new).

- Evaluation of fuel rod bowing effects (Section 4.4).

- Analysis of the reactor coolant system response to pressure transient that can potentially occur during startup and shutdown (Section 5.2.2).

- Inadvertent closure of decay heat removal system isolation valves during decay heat removal operation (Section 5.5.3).

- Isolation of the low pressure reactor heat removal system from the primary system (Section 5.5.3).

- Submittal of large break loss-of-coolant accident spectrum to document exact margins within emergency core cooling acceptance criteria of 10 CFR Part 50.46 (Section 6.3.3.1).

- Install flow measuring devices to monitor adequacy of boron dilution modes of plant operation (Section 6.3.3.4).

- Plant operating restrictions with less than three reactor coolant pumps in operation (Section 6.3.3.6).

- Verification of the reactor protection system equipment electrical noise qualification testing (Section 7.2).

- Modification in redundant reactor coolant flow transmitters in order to meet single failure criteria (Section 7.2).

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·· Evaluation of facility fire protection capability in accordance with Appendix A to Branch Technical Position APCSB 9.5.1 (Section 9.6.1).

·· Effects of degraded offsite power on safety equipment (Section 8.2).

·· Leakage test limits for Americium-Beryllium-Copper neutron startup sources (Section 12.6-new).

1.8 Generic Issues

(1) The following items have been resolved since issuance of our Safety Evaluation Report.

··· Pellet cladding mechanical interaction (Section 4.2.1).

··· Emergency core cooling analysis modifications (Section 6.3.3).

(2) New information and/or the status of this item has changed since issuance of our Safety Evaluation Report and required license conditions as described in Section 1.7.2.

··· Evaluation of fuel rod bowing effects (Section 4.4).

·· Evaluation of facility fire protection capability in accordance with Appendix A to Branch Technical Position APCSB 9.5.1 (Section 9.6.1).

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2.0 SITE CHARACTERISTICS

2.3 Meteorology

2.3.4 Short-Term (Accident) Diffusion Estimates

We stated in the Safety Evaluation Report that the applicant would be required to submit an additional year of onsite meteorological data for the period August 1975 to August 1976 and a combined two-year period of data for August 1974 to August 1976. We also stated that we would evaluate the data sets to verify that the relative concentration values (X/Q) reported in our Safety Evaluation Report for the time period August 1974 to August 1975 were representative and conservative for the Davis Besse site.

We have evaluated the additional data sets submitted by the applicant on November 29, 1976. These data sets were in the form of joint frequency distributions of wind speed and wind direction measured from the 35-foot level for atmospheric stability (defined by the vertical temperature gradient between 35 feet and 250 feet). Data recovery for these periods was about 93 percent.

We have calculated the relative concentration values for the data sets of August 1975 to August 1976 and August 1974 to August 1976. We assumed a ground-level release with a building wake factor, CA, of 1300 square meters in our calculation of short-term accidental releases from buildings and vents (0-2 hours at the exclusion distance and 0-8 hours at the low population zone distance. The relative concentration values for the various time periods following an accidental release were calculated by using the diffusion model described in Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors, Revision 2, June 1974."

We find that the relative concentration values reported in our Safety Evaluation Report using data from the time period August 1974 to August 1975 are more conservative than those calculated using the August 1975 to August 1976 and the August 1974 to August 1976 data sets.

Further, we have concluded that the two years of onsite meteorological data for the time period August 1974 to August 1976 provide an adequate meteorological description of the Davis Besse site and vicinity. The August 1974 to August 1976 relative concentration value for the 0-2 hour time period, which is exceeded five percent of the time, at the exclusion distance of 732 meters, is no different than the magnitude (2.2×10^{-4} seconds per cubic meter) as reported in our Safety Evaluation Report. The relative concentration values for the various time periods at the low population zone boundary

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distance of 3200 meters are about fifteen percent lower than those reported in the Safety Evaluation Report and are listed below:

Time Period	Relative Concentration (X/Q) (seconds per cubic meter)
0-8 hours	8.2×10^{-6}
8-24 hours	5.7×10^{-6}
1-4 days	2.6×10^{-6}
4-30 days	8.0×10^{-7}

We find these data (August 1974 to August 1976) provide an acceptable basis for calculations of reasonably conservative relative concentration values for the assessment of postulated post-accidents. The doses presented in Section 15.0 of the Safety Evaluation Report have been reevaluated where applicable with these slightly lower concentration values and are presented in Section 15.0 of this supplement.

2.5 Geology, Seismology, and Foundation Engineering

We stated in our Safety Evaluation Report that we were evaluating the seismicity at the site for the proposed facility units, Davis Besse, Units 2 and 3. We stated that, based on our seismic evaluation of the site for Units 2 and 3 we would report any conclusions applicable to Unit 1.

The vibratory motion for seismic design at the Davis Besse Unit 1 site is based on a Modified Mercalli intensity VII-VIII. This corresponds to the Anna, Ohio, event of March 8, 1937, the highest intensity earthquake in the central stable region that has not been definitely associated with structure. It had a felt area of 150,000 square miles and an estimated magnitude of (5 to 6) based upon limited instrumental data, felt area and intensity-magnitude correlations. The extent of the felt area indicates that it could not have been a very shallow (less than five kilometers in depth) event. The foundation conditions at the Davis Besse 1 site 100 miles north of Anna consist of 15 feet of glacial till overlying bedrock (dolomite and shale). The seismic Category I structures are founded either on or near the bedrock surface. Vibratory ground motion estimates that best approximate the safe shutdown earthquake would then be that derived from accelerograms recorded on rock near magnitude (5 to 6) earthquakes that had maximum intensities of VII-VIII.

The applicant designed the Davis Besse Unit 1 facility prior to the issuance of Appendix A to 10 CFR Part 100 and Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants." The design response spectrum was based on a modification of the east-west accelerogram recorded during the Helena, Montana earthquake of October 31, 1935. This earthquake had an instrumentally determined magnitude of 6, a maximum epicentral intensity of VIII, and a felt area of 140,000 square miles.

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Due to regional variations in seismic wave attenuation, western earthquakes have smaller felt areas than eastern earthquakes of similar magnitude. The accelerogram which had a peak accel‹ ation of 0.16g was recorded on a rock site approximately five miles from the epicenter. Considering the parameters of the safe shutdown earthquake required for the Davis Besse site (MM VII-VIII) this accelerogram may be considered a good choice for a design basis with respect to expected earthquake magnitude, intensity, epicentral distance and foundation c.nditions. Moreover, the 1935 Helena earthquake appears to be conservative with respect to earthquakes that could be expected to occur in the vicinity of the Davis Besse site.

We consider that the design basis ground motion used by the applicant adequate to represent the earthquake hazard at the Davis Besse, Unit 1 site. However, because of changes in the regulatory approach to selection of seismic design bases discussed with the Advisory Committee on Reactor Safeguards at its meeting held on January 6, 1977 for Davis Besse, Unit 1, we are requiring the Toledo Edison Company to reevaluate portions of the plant during the first fuel cycle period for a safe shutdown earthquake acceleration of 0.2g applied at the foundation of the plant. This matter is discussed in further detail in Section 3.7 and Section 18.0 of this supplement.

3.0 DESIGN CRITERIA-STRUCTURES, SYSTEMS AND COMPONENTS

3.7 Seismic Design

3.7.1 Seismic Input

Because of changes in the regulatory approach to selection of seismic design bases discussed with the Advisory Committe on Reactor Safeguards at its meeting held on January 6, 1977 for Davis Besse, Unit 1, the Committee indicated in its letter that it "believes an acceleration of 0.2g would be more appropriate for the safe shutdown earthquake acceleration at a site such as this in the Central Stable Region." As discussed in Section 18.0 of this report, the Committee recommended that the staff review in detail the plant systems needed to accomplish safe shutdown of the reactor and continued shutdown heat removal for a safe shutdown earthquake acceleration of 0.2g and that Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," should be applied at the foundation level of the facility.

We are currently preparing guidelines for the applicant on this matter as part of our request for a seismic reanalysis. We agree with the Committee that the seismic reevaluation need not delay the start of operation of Davis Besse, Unit 1. We also agree with the Committee regarding the scope of the seismic reanalysis and believe that the reanalysis and evaluation can be completed during the first fuel cycle. We will condition the license to require that the analysis and evaluation be completed prior to startup following the first regularly scheduled refueling outage.

We conclude that the likelihood of a seismic event occurring during the first fuel cycle period that could adversely effect the facility and the health and safety of the public is acceptably small.

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4.0 REACTOR

4.2 Mechanical Design

4.2.1 Fuel

In the Safety Analysis Report, we identified pellet cladding interaction as a possible fuel failure mechanism, and noted that (1) we are pursuing this problem on a generic basis, and (2) experience with Oconee 1 showed no failures related to this phenomenon. We identified a tentative schedule for our action and stated that we believe that technical specification limits on coolant activity provide adequate protection against operation with excessive failed fuel.

Our review to date of the effects of pellet cladding interaction has so far not identified any safety problem. The Babcock & Wilcox fuel rod design incorporates features directed at reducing cladding strain due to pellet cladding interaction. These include pellet chamfering, prepressurization, incorporation of plenum regions at both top and bottom of the fuel rod, and thicker cladding. Based on experimental and commercial reactor data available, these design features provide reasonable assurance that the potential for pellet cladding interaction failure will not occur until later in the fuel design lifetime. While the failure threshholds are probably lower at higher burnups, the fuel duty is also less severe. Therefore, operating restrictions are not presently warranted. If in the future our continuing program in this area should identify any safety problems, appropriate requirements will be imposed at that time.

4.4 Thermal and Hydraulic Design

We stated in our Safety Evaluation Report that the effects of rod bow on thermal and hydraulic performance was a matter for which operational restrictions can be imposed if necessary. We requested the applicant to evaluate rod bow restrictions for Davis Besse, Unit 1 and to revise the technical specifications to accommodate any rod bow restrictions. The applicant responded with a revision to his technical specifications. We reviewed the submittal and a determination was made that the applicant's evaluation on the amount of reduction in departure from nucleate boiling ratio to account for rod bow was not adequate. The applicant stated that the amount of rod bow which the proposed technical specifications is based upon, including credit for thermal margins, is 5.9 percent. Our calculations indicate that this amount of rod bow is predicted to be exceeded after a relatively short period of burnup. Therefore, a condition to the operating license for Davis Besse, Unit 1 will stipulate:

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> If the Babcock & Wilcox proposed rod bow model has not been approved by the Commission upon completion of one hundred (100) effective full

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power days of operation for the facility, the Toledo Edison Company must revise the plant technical specifications to reflect the rod bow model used by the Commission for Babcock & Wilcox plants which requires the following departure from nucleate boiling ratio penalties as a function of burnup:

Burnup (Megawatt-days per metric ton)	Departure From Nucleate Boiling Ratio Penalty
0 - 5,651	5.9 percent
5,651 - 15,000	8.2 percent
15,000 - 24,000	9.8 percent
24,000 - 33,000	11.2 percent

We stated in our Safety Evaluation Report that we had requested that the applicant implement a program of inspection and test of the core support internal vent valves. The facility technical specifications 4.4.10.1b specify such a test and is acceptable to the staff. Also, as stated in our evaluation report dated November 19, 1975 for the Babcock and Wilcox Report, "B&W Operating Experience of Reactor Internals Vent Valves," we will require that reports to the Commission be made should any loose parts monitoring anomalies be attributed to a vibrating vent valve or vent valve components.

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5.0 REACTOR COOLANT SYSTEM

5.2 Integrity of the Reactor Coolant Pressure Boundary

5.2.2 Overpressure Protection

We stated in our Safety Evaluation Report that we were continuing our review of overpressure protection for Davis Besse, Unit 1 during startup and shutdown.

We have evaluated incidents known as pressure transients (events that have exceeded the technical specification temperature-pressure limits of the reactor vessel) and issued a technical report in November 1976, "NUREG-0138, "Staff Discussion of Fifteen Technical Issues listed in Attachment to November 3, 1976 Memorandum from Director, NRR to NRR Staff." The report concluded in part that pressure transients are of a concern during startup and shutdown because, at these relatively low temperatures, the vessel has less toughness than at operating temperatures and irradiation increases the temperature at which steel attains maximum toughness. The Appendix G to 10 CFR Part 50 limits change during the life of the plant as it becomes irradiated, and because it would be impractical to change these limits, they are calculated for an extended period of time. Thus, the limits in effect at a given time may be based on properties expected in the vessel five or more years in the future, making them conservative during the early portion of this period. The report concluded that large safety margins to failure exist for unirradiated reactor vessels, and new plants can be permitted to be licensed under existing safety criteria. Nevertheless, we concluded that administrative procedures and overpressure protection devices should be upgraded to reduce the likelihood of future pressure transient events.

On December 7, 1976, the applicant submitted an analysis to show compliance with Appendix G to 10 CFR Part 50 pressure-temperature limits during startup and shutdown. The staff reviewed the applicant's analysis and requested further information from the applicant. The applicant responded on February 18, 1977 with a discussion which provided the staff with further assurance that Appendix G limits would not be violated. However, to further reduce the likelihood of pressure transient events, we require that the applicant ensure that the decay heat removal relief valve will actuate prior to automatic closure of the isolation valves. This change will allow the relief valve to be available for mitigating the consequences of an overpressure event. A condition to the operating license will stipulate that prior to entering Mode 2 (Startup), the applicant shall make a modification which ensures that the decay heat removal relief valve will actuate prior to automatic closure of the isolation valves. The change will allow a longer period of time for the relief valve to be available to the reactor coolant system for mitigating the consequences of overpressure event.

We note that Davis Besse, Unit 1 and other Babcock and Wilcox plants utilize nitrogen gas to maintain a gas bubble in the pressurizer whenever a steam bubble is not maintained, so that no plant operation will involve a solid water condition.

We find the above means to minimize the likelihood of exceeding Appendix G limits for the first fuel cycle to be acceptable. However, additional means must be incorporated prior to the start of the second fuel cycle to further reduce the potential for exceeding Appendix G limits. The applicant has proposed a long-term solution which is under review. We will condition the operating license to require that the licensee implement, prior to the end of the first regularly scheduled refueling outage, a long-term means of overpressure protection that is acceptable to the staff.

5.2.4 Leakage Detection System

We stated in the Safety Evaluation Report that we had not completed our review of the leakage detection system for Davis Besse Unit 1. Our review and conclusions for the facility leakage detection system are described below.

Coolant leakage within the containment may be an indication of a small through-wall flaw in the reactor coolant pressure boundary. The systems provided to detect leakage to the containment employ diverse leak detection methods, have sufficient sensitivity to measure small leaks, and can identify the leakage source to the extent practical. The major systems provided are the containment vessel sump, radiogas and air particulate radioactivity monitors. Intersystem leakage will be detected by abnormal readings from radioactivity monitors in the secondary system.

The construction permit for Davis Besse Unit 1 was issued prior to the issuance of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems." We have determined, however, that the leakage detection systems provided for Davis Besse Unit 1 are generally in accordance with the recommendations of Regulatory Guide 1.45. The principal exceptions are that not all of the components of the containment vessel sump monitors are specifically seismically qualified and the containment vessel sump flow is not alarmed in the control room. A leakage detection system which is provided in addition to those recommended by Regulatory Guide 1.45, is the containment vessel air cooler condensate monitors.

Based on the degree of conformance with the recommendations of Regulatory Guide 1.45 and on the use of systems in addition to those recommended by Regulatory Guide 1.45, we conclude that the leakage detection system satisfies the requirements of Criterion 30 of the General Design Criteria and is acceptable.

5.3 Reactor Vessel Integrity

We stated in our Safety Evaluation Report that we would evaluate any modifications to the existing reactor surveillance program to ensure continued compliance with Appendix H, 10 CFR Part 50.

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The toughness properties of the reactor vessel beltline material will be monitored throughout the service life by a material surveillance program which conforms to Appendix H, 10 CFR Part 50 and American Society for Testing and Materials Standard E 185-73 to the maximum extent practical for a vessel ordered prior to the publication of Appendix H. The technical basis and general description of the program is contained in Topical Report BAW-10100A, "Reactor Vessel Material Surveillance Program," which we have reviewed and found acceptable.

In addition, the calculated fluence values for exposure time have been updated using a different analytical model combined with analytical predictions of the effect of refueled core configurations on relative power distribution. This analytical model has been verified and refined by comparison with surveillance capsule specimen analyses recently removed from several 177-FA Babcock & Wilcox reactors. As a result of the redesign of the holder tubes and the updated analytical model, the predicted neutron flux received by the specimens is more than three times as high as that received by the vessel inner surface. Therefore, the applicant has proposed a surveillance specimen withdrawal schedule different from Appendix H, Section II.C.3.c. We had requested a technical justification to demonstrate that the rate of irradiation does not affect the measured fracture toughness properties of the weld metal, base metal and heat affected zone.

However, our evaluation of the available information concludes that the Davis Besse, Unit 1 reactor vessel material surveillance program with the redesigned surveillance specimen holder tube locations is acceptable at least through the first fuel cycle when the first capsule is scheduled to be withdrawn and evaluated. The technical basis for this conclusion is that the results during this period of operation will be conservative since the irradiation effects on the surveillance specimens will occur sooner than those for the reactor vessel.

Our investigation of the influence of irradiation rate on Appendix H requirements is continuing on a generic basis. If the results from the applicant's material surveillance program or from the Commission's sponsored programs indicate that the rate of irradiation has a significant effect on measured fracture toughness properties, we will require that the applicant submit for approval a modification to his surveillance capsule withdrawal schedule after the first fuel cycle.

We stated in the Safety Evaluation Report that the redesigned reactor vessel surveillance specimen holder tubes would be tested for design verification during the hot functional test for Davis Besse Unit 1.

The applicant has installed the redesigned surveillance specimen holder tubes described in the Babcock & Wilcox Topical Report, Supplement 1, "Structural Analysis of 177-FA Redesigned Surveillance Specimen Holder Tube."

Test data required to verify the adequacy of the redesigned surveillance specimen holder tubes were obtained through the use of accelerometer and strain gauge instrumentation which was installed prior to the hot functional test. Our review and evaluation of the test data obtained during the hot functional test has verified our conclusions as stated in the Safety Analysis Report that we find the redesign of the surveillance specimen holder tubes to be acceptable.

An inspection of the reactor core barrel and the reactor vessel after the completion of the hot functional test for Davis Besse, Unit 1 revealed that five out of 24 of the guide block and lug faces had been in contact during the test.

One guide block was found to be loose due to severance of a guide bolt which connects to the core barrel. The applicant has proposed to reinstall the guide block assembly to the core barrel by welding around the guide block with a one-quarter inch fillet weld in addition to the installation of dowel pins. We have evaluated the design adequacy of the measures proposed by the applicant to position and secure the guide blocks to the core barrel. Based on our evaluation, we conclude that the proposed design provides adequate assurance that the guide blocks will not fail due to any load from the core barrel that might occur during the life of the plant. The reinstalled guide blocks have been inspected and have been found to be satisfactory.

Our review of the contact between the guide block and lug faces indicated that this condition was not sufficient to alter the primary vibration modes of the core barrel during hot functional test conditions. Therefore, all vibration data obtained during the hot functional test remains valid and our conclusions remain the same.

Based on our acceptance of the design and verification of the design by tests, we conclude that this design is capable of withstanding the dynamic environment to which it will be subjected, and will provide adequate safety margins against structural failure.

5.5 Component and Subsystem Design

5.5.3 Decay Heat Removal System

We required the applicant to address the potential for and consequences of an inadvertent closure of a decay heat removal isolation valve during shutdown operations. The applicant has proposed removing power from the two decay heat removal valves during shutdown operations. While this procedure would resolve the problem of an inadvertent valve closure causing damage to the decay heat removal pumps, it is our judgment that this proposal could compromise the barrier between the high and low pressure piping by increasing the potential for the plant starting up with only one valve closed. With normal power available to a decay heat removal valve inadvertently left open by the operator, the automatic closure feature would provide backup to this operator error when primary side pressure was increased. With normal power not available to a decay heat removal valve inadvertently left open by the operator, the applicant has stated that sufficient alarms and visual indications would be available to the operator to alert him to take a corrective action; however, no backup automatic closure would

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exist and the plant could continue at power operation with only one barrier available between the high pressure and low pressure piping. We find this unacceptable.

A condition to the operating license for Davis Besse, Unit 1 will stipulate that until such time as an acceptable design alternative is implemented to accommodate the consequences of an inadvertent closure of a decay heat removal valve during decay heat removal operations, Toledo Edison Company shall maintain power on decay heat removal valves DH11 and DH12 and operate the decay heat removal system with only one train at a time in order to ensure the availability of one train of the decay heat removal system at all times.

With regard to the bypass loop containing two manually operated valves around the decay heat removal suction line isolation valves, the Advisory Committee on Reactor Safeguards (Section 18.0 of this report) has stated that further attention should be given to the means employed for isolation of the low pressure reactor heat removal system from the primary coolant system while the latter is pressurized, and that reliable means be developed to assure such isolation. We note that administrative controls on the manual bypass valves DH21 and DH23 have been changed to require a key to open their normally locked-closed status. Nevertheless, it is our judgment that additional means are necessary to further minimize the potential for inadvertent opening of the bypass valves during high pressure operation. Discussions have taken place with the applicant with regard to a flange of the spectacle type which could be installed between the two bypass valves. Such a spectacle flange would further decrease the potential for the bypass path being opened at the wrong time, yet still retain the capability of maintaining decay heat capability should one of the decay heat removal suction line valves fail in a closed position.

We will require that a reliability study be made for a spectrum of hypothesized design modifications to be compared with the present design of the low pressure residual heat removal system. We will evaluate the design modifications to determine if the modifications enhance the safety of the system and determine that the final system is acceptable to further minimize the potential for inadvertent opening of the bypass valves during high pressure operation. As a condition to the operating license for Davis Besse, Unit 1, we will require that final resolution and design modifications be completed prior to the start of operation following the first refueling outage.

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6.0 ENGINEERED SAFETY FEATURES

6.2 Containment Systems

6.2.1 Containment Functional Design

We stated in our Safety Analysis Report that our confirmatory analyses of a postulated loss-of-coolant accident were predicting pressures for the reactor cavity and steam generator compartments which exceeded the values used in the structural design of these compartments. For these analyses, the applicant had assumed double-ended (14.14 square feet) hot leg breaks in these subcompartments. We also stated that we were investigating the differences between our confirmatory analyses and the applicant's analyses with the applicant and that the resolution of this matter would be discussed in a subsequent report.

By letter dated November 15, 1976, the applicant submitted additional information which we required in order to perform our confirmatory analyses. In this letter the applicant provided the inertia terms that are required to accurately model the inertial effects of blowdown in a subcompartment. Using the parameters supplied by the applicant, we are able to confirm that the design pressure for the steam generator pressurizer compartment is acceptable.

In Amendment No. 39 to the Final Safety Analysis Report, dated November 1976, the applicant revised the postulated pipe break analyses in the reactor cavity. Rather than assume a double-ended (14.14 square feet) hot leg break, the applicant took credit for piping supports and restraints to limit the maximum credible hot leg break to a single-ended (7.07 square feet) break size. We have verified that the applicant has properly accounted for the pipe supports and restraints, and that the postulated single-ended hot leg break will be the maximum break size. In addition to this break, the applicant also analyzed a double-ended (8.55 square feet) cold leg break for the reactor cavity analysis. As a result, the double-ended cold leg break is the limiting break for the reactor cavity. Using the revised postulated pipe break sizes, we are able to confirm the design pressure for the reactor cavity is acceptable.

The CRAFT computer code was used to calculate the mass and energy release rates for the subcompartment analyses. This version of CRAFT uses the MOODY correlation to calculate flow at the break, which may not be conservative for the calculation of subcooled flow rates. However, since the limiting pipe breaks for each subcompartment are double-ended breaks, the change in the subcooled portion of the blowdown would be negligible. Therefore, we conclude that the mass and energy release rates as presented by the applicant are acceptable.

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On the basis of our review, we have concluded that the structural designs of the containment steam generator pressurizer compartment and the reactor cavity are acceptable.

We stated in our Safety Evaluation Report that the environmental qualification of safety-related equipment required to be operable following a postulated main steam line break had not been resolved. Although a reanalysis of the postulated main steam line break accident may show higher than design temperatures existing for a short period of time, recent qualification testing of safety-related equipment indicates that the equipment is insensitive to short term temperature spikes typically calculated for the main steam line break accident. Additional confirmation of the main steam line break accident analysis for the Davis Besse Nuclear Power Station, Unit 1 will be resolved on a generic basis. If safety concerns should be identified in our generic review of the environmental qualification of safety-related equipment required to be operable following the postulated main steam line break accident analysis for other Babcock & Wilcox plants, we will reopen our review of this matter for Davis Besse, Unit 1 (see Section 7.7 of this supplement).

6.2.3 Secondary Containment Functional Design

We stated in the Safety Evaluation Report that certain assumptions made by the applicant regarding the analysis of the shield building pressure response following a postulated loss-of-coolant accident were unacceptable. We stated that we would request the applicant to reanalyze the shield building depressurization and that we would report the results of our review in a subsequent report.

Following a postulated loss-of-coolant accident, containment leakage is assumed to take place at the design leakage rate of 0.5 percent per day by weight. Offsite radiological dose calculations do not assume radioactive holdup within the shield building until the emergency ventilation system fans draw the shield building pressure down to a negative 0.25 inch of water (gauge). Therefore, it is conservative to maximize the time required for the shield building to be depressurized.

The applicant has submitted an analysis of the pressure response of the annulus assuming no outleakage during the positive pressure period and a depressurization time of 802 seconds (about 13.4 minutes) was calculated.

Based on our review of the applicant's analysis, we have concluded that the analysis is conservative and that the calculated depressurization time is acceptable.

In Section 15.3 of our Safety Evaluation Report we used a depressurization time of 780 seconds (13 minutes) in our calculation of doses for postulated radiological consequences of accidents. We have recalculated the radiological accident doses using the conservative depressurization time of 802 seconds (about 13.4 minutes). The recalculated doses are provided in Section 15.3 of this report. We find that

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our reevaluation of the postulated accident doses are within the 10 CFR Part 100 guidelines. On this basis we conclude that the secondary containment functional design is acceptable.

6.3 Emergency Core Cooling System

6.3.2 System Design

With regard to leak testing of the check valves in the high pressure to low pressure interface of the low pressure injection discharge line, the applicant has committed to periodically verifying valve integrity in accordance with Section 4.05 of the Technical Specifications. This surveillance requirement performed on valves CF-28, CF-29, CF-30, CF-31, DH-76 and DH-77 for at least each refueling outage is in accordance with the applicant's commitment of April 18, 1975, to staff position 6.3.2 and is acceptable.

We also requested that the applicant adopt a surveillance requirement in the technical specifications to verify that the emergency core cooling system piping is water solid to minimize the potential for water hammer. Technical Specification 3/4 5-4 provides this requirement.

6.3.3 Performance Evaluation

In Section 6.3.3 of the Final Safety Analysis Report, the applicant incorporated by reference the Babcock and Wilcox topical reports, BAW-10104, "B&W's ECCS Evaluation Model," May 1975 and BAW-10105, "ECCS Evaluation of B&W's 177-FA Raised-Loop NSS," June 1975 into its application to operate Davis Besse, Unit No. 1. Pursuant to the requirements of 10 CFR 50.46, the Babcock and Wilcox Company submitted these reports to demonstrate compliance with the emergency core cooling system Acceptance Criteria for plants using 177 fuel assemblies with raised loops. The basis for acceptance of the principal portions of the Babcock and Wilcox evaluation model were set forth in the staff's Status Report of October 1974 and the Supplement to the Status Report of November 1974. Together, the Status Report and its Supplement describe the Babcock and Wilcox emergency core cooling system evaluation model and the basis for the staff's previous acceptance of the model. BAW-10104 describes the general features of the Babcock and Wilcox emergency core cooling system evaluation model and reflects the modifications previously required by the staff. The original emergency core cooling system calculations applicable to Davis Besse 1 were submitted in BAW-10105 using the Babcock and Wilcox evaluation model described in BAW-10104. Later developments on the validity of these calculations revealed the following:

(1) The Babcock and Wilcox method for calculating fuel cladding temeratures during the blowdown phase of a loss-of-coolant accident did not conform to Appendix K because it allowed for a return to nucleate boiling after critical heat flux conditions have been reached.

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(2) A steam cooling model was used in the Davis Besse 1 emergency core cooling system calculations which had not been reviewed by the staff.

(3) Improper pin pressure assumptions were employed.

(4) Incorrect values of certain loop resistances were used.

With regard to item (1) above, the staff evaluation of a revised nucleate boiling lockout logic proposed by Babcock and Wilcox concludes that the revised logic is an appropriate change to be incorporated in the Babcock and Wilcox Evaluation Model and that the overall effect on the change on peak clad temperature would be small (about six degrees Fahrenheit).

With regard to item (2) above, the staff has concluded that the steam cooling model used by Babcock and Wilcox is acceptable. Items (3) and (4) relate to input errors and are discussed in more detail in Section 6.3.3.1 of this supplement.

Other model changes have taken place subsequent to the emergency core cooling system calculations in BAW-10105. These changes have been reviewed and accepted by the staff and their cumulative effect is not significant to the peak clad temperature which still meets the acceptance criteria of 10 CFR Part 50.46.

6.3.3.1 Emergency Core Cooling System Analyses

The background of the staff's review of the revised Babcock and Wilcox emergency core cooling system evaluation model and its application to Davis Besse Unit No. 1 is described in a letter from A. Schwencer (NRC) to K. Suhrke (B&W) dated January 8, 1976. The applicant's Final Safety Analysis Report contains documentation by reference to BAW-10105 of a generic break spectrum appropriate to Davis Besse Unit No. 1. The applicant has appropriately referenced all subsequent corrections and revisions to the Babcock and Wilcox model. In addition, we will require that the responses to questions submitted on BAW-10105 will be made a part of the revised topical report by Babcock and Wilcox. In the initial analysis, a spectrum of break sizes, configurations, and locations were performed. These analyses identified the worst break as the 8.55 square foot double-ended break at the pump discharge. Parametric computations performed in connection with the various corrections and modifications submitted since the initial computations, showed that the size and locations of the worst break is not affected.

Babcock and Wilcox responses to staff inquiries during its review of BAW-10105 determined that incorrect internal fuel pin pressures had been assumed in the emergency core cooling system calculations. Babcock and Wilcox subsequently resubmitted analyses with the corrected pin pressures. These revised analyses also included consideration of an additional flow resistance in the cold legs to account for high pressure injection pumps injecting emergency core cooling water during reflood.

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Table 6.1 summarizes the results of the revised loss-of-coolant limit analyses which determine the allowable linear heat generation rate limits as a function of elevation in the core for the worst case break:

TABLE 6.1

ALLOWABLE LINEAR HEAT GENERATION LIMITS

Elevation (feet)	LHGR Limit (Kw/ft) (Kilowatts per foot)	Peaking Cladding Temperature (Degrees Fahrenheit)	Maximum Local Oxidation (Percent)
2	16.5	2133	4.01
4	17.2	2073	3.15
6	18.4	2166	5.25
8	17.5	2164	6.56
10	17.0	2194	7.17

Subsequent to this review the applicant informed the staff that an erroneous resistance value to the reactor vessel inlet nozzle had been used in the loss-of-coolant accident analysis. To determine the effect of such an error, the applicant submitted a reevaluation of the Davis Besse Unit No. 1 loss-of-coolant accident analysis based on the corrected inlet nozzle model for the worst case break considering the loss-of-coolant accident limit at the six-foot elevation. At the same time the applicant also revised the system pressure distribution. These results showed a lower peak cladding temperature for the worst break analysis. The peak cladding temperature obtained for the reevaluation of the loss-of-coolant accident limit at the six-foot level analysis is 2133 degrees Fahrenheit, a value 33 degrees Fahrenheit lower than that previously calculated (see Table 6.1).

The reduction in overall peak cladding temperatures, despite correction of the inlet nozzle resistance value, was due to improved reflooding rates in the core resulting from the improved system pressure distribution (i.e., the new reactor coolant system total pressure drop is less than the original assumed pressure drop).

We have reviewed the proposed pressure drops, the derivation of the revised system pressure distribution and its impact on the loss-of-coolant accident limit analysis, and agree that they properly conform to the provisions of Appendix K, so that the proposed linear heat generation rates calculated on the bases of the revised system pressure distribution, will conform to the criteria of 10 CFR Part 50.46. The reevaluation of the loss-of-coolant accident at the six-foot level limit is sufficient to determine that the effect of the revised system pressure distribution on peak cladding temperature for the range of axial power distributions previously analyzed would not increase the previously calculated peak clad temperatures. As reported

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earlier, the peak cladding temperature following a loss-of-coolant accident was reduced when analyzed at the six foot elevation of the core. Similar effects would be expected at other elevations of the core.

Although the reevaluated results are less severe than those reported in BAW-10105, the applicant does not propose to increase the allowable linear heat generation rate limits for Davis Besse, Unit 1 on the basis of the most recent submittals discussed in this report. Although we conclude that the proposed linear heat generation limits are in compliance with the acceptance criteria of 10 CFR Part 50.46, because of the recent changes separately treated in various separate computations described above, we will require that the applicant quantify the actual margins that exist when all the changes are considered together and submit within six months from the date for issuance of an operating license additional analyses for a range of large break spectrum and use a model in which all modifications have been incorporated to further document the exact margins within the emergency core cooling system criteria of 10 CFR Part 50.46. The additional analyses should, as a minimum, confirm previous evaluations with regard to worst break size, configuration, and allowable linear heat generation limits as a function of elevation in the core.

Also, we will require the applicant to provide operating reactor coolant system flow data for the Davis Besse, Unit 1 which can be used to further verify the assumed total system pressure drop. The new pressure drops were based on standard calculation methods supported by operating plant pressure data and the results from scaled reactor vessel model flow tests. Babcock and Wilcox has shown that although there are some design differences between Davis Besse, Unit 1 and other Babcock and Wilcox plants from which measured data were obtained, these differences have a negligible effect on total system pressure drop. We have reviewed the flow path resistances input to the REFLOOD emergency core cooling system evaluation code for the revised system pressure distribution, and have checked several flow paths resistance values. We find the methods to be appropriate for the derivation of loop resistances and accept the reported values as being appropriate for Davis Besse, Unit 1.

Therefore, the staff concludes that the previous values quoted in Table 6.1 remain applicable to Davis Besse, Unit 1.

The maximum core-wide metal-water reaction was calculated to be 0.66 percent, a value which is below the allowable limit of one percent. As shown in the tabulation in Table 6.1, the calculated values for peak clad temperature and local metal-water reaction were below the allowable limits specified in 10 CFR Part 50.46 of 2200 degrees Fahrenheit and 17 percent, respectively. As shown in BAW-10105 core geometry remains amenable to cooling and long-term core cooling can be established.

Our review of other plant-specific assumptions discussed in the following paragraphs regarding the Davis Besse, Unit 1 analyses addressed the areas of single failure criterion, long-term boron concentration, potential submerged equipment, partial loop operation, and the containment pressure calculation.

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6.3.3.2 Single Failure Criterion

Appendix K to 10 CFR 50 of the Commission's regulations requires that the combination of emergency core cooling system subsystems to be assumed operative shall be those available after the most damaging single failure of emergency core cooling system equipment has occurred. In its analyses Babcock and Wilcox has conservatively assumed all containment cooling systems to be operating to minimize containment pressure and has assumed the loss of one diesel to minimize emergency core cooling system cooling. We stated in our Status Report of October 1974 that the application of the single failure criterion was to be confirmed during subsequent plant reviews.

The applicant has concluded that no single active failure would more severely degrade the emergency core cooling system than the previous assumptions stated above. A review of the Davis Besse, Unit 1 piping and instrumentation diagrams and emergency core cooling system motor-operated valve electrical schematics was conducted by the staff. Based on these electrical schematic reviews the staff required electrical design changes for valves in the low pressure injection discharge lines, low pressure injection-high pressure injection crossover lines, and high pressure injection mini-flow bypass lines. On the basis of the revised plant design, the staff concludes that a bounding single failure analysis has been performed for the Davis Besse, Unit 1 facility, and that no single failure will more severely degrade the emergency core cooling system than the loss of one emergency diesel.

6.3.3.3 Containment Pressure

The emergency core cooling system containment pressure calculations for Babcock and Wilcox plants with 177 fuel assemblies in a raised loop configuration were performed generically by Babcock and Wilcox as described in BAW-10105. We reviewed Babcock and Wilcox's evaluation model and published the results of this review in our Status Report and its Supplement in October 1974 and November 1974, respectively. We concluded that Babcock and Wilcox's containment pressure model was acceptable for emergency core cooling system evaluations. We required that justification of the plant-dependent input parameters used in the containment analyses be submitted for our review of each plant.

Justification for the containment input data was submitted for Davis Besse, Unit 1 by letter dated September 5, 1975. This justification allows comparison of the actual containment parameters for Davis Besse, Unit 1 with those assumed in topical report BAW-10105. The applicant has evaluated the containment net-free volume, the passive heat sinks, and operation of the containment heat removal systems with regard to the conservatism for the emergency core cooling system analysis. This evaluation was based on as-built design information. The containment heat removal systems were assumed to operate at their maximum capacities, and lowest expected values for the spray water and service water temperatures were assumed. The containment pressure analysis in BAW-10105 was demonstrated to be less than the calculated

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pressure response for the Davis Besse, Unit 1 containment following a loss-of-coolant accident and is therefc._ conservative for Davis Besse, Unit 1.

We have concluded that the plant-dependent information used for the emergency core cooling system containment pressure analysis for Davis Besse, Unit 1 is conservative and, therefore, the calculated ... .cainment pressures are in accordance with Appendix K to 10 CFR Part 50 of the Commission's regulations.

6.3.3.4 Long-Term Boron Concentration

We have reviewed the proposed procedures and the systems designed for preventing excessive boric acid buildup in the reactor vessel during the long-term cooling period after a postulated loss-of-coolant accident. The applicant has agreed to implement procedures for Davis Besse, Unit 1 which would allow adequate boron dilution during the long term and which will comply with the single failure criterion. These procedures will employ a hot leg drain and hot leg injection network similar to the concept described in BAW-10105. The hot leg drain mode will direct reactor coolant from the hot leg, down the decay heat line to the decay heat removal pump suction. This coolant draining from the hot leg would then be mixed with the diluted water being pumped from the containment sump and would then be pumped back to the reactor vessel. Should a single active component failure not allow operation of the hot leg drain mode, the operator then has the alternative of selecting the hot leg injection mode to provide boron dilution. The procedure would be to use the relatively diluted water being pumped out of the contai ent sump during the long-term recirculation mode and route a minimum of 40 gallons per minute of this sump water to the hot leg to provide dilution of the water in the upper plenum of the reactor vessel. The applicant will be required to demonstrate this minimum flow rate in each mode during preoperational testing. In addition, the applicant must install flow rate measuring devices prior to startup following the first refueling to assure that a minimum of 40 gallons per minute is continually available following a loss-of-coolant accident, and to facilitate system tests. With the addition of the flow devices and preoperational tests, this proposal is acceptable to the staff. We conclude that the acceptable results from the preoperational tests will provide reasonable assurance that the system will deliver the minimum flow if needed during the first refueling period prior to addition of the flow rate measuring devices.

6.3.3.5 Submerged Valves

The applicant has conducted a review of equipment arrangement to determine if any components inside the containment will become submerged following a loss-of-coolant accident. Based on this review, decay heat suction valves DH-11 and DH-12 were identified as being located in an area that will be flooded. The applicant subsequently enclosed these valves in a watertight compartment to ensure their operability during the long term after a loss-of-coolant accident. The staff will require that an acceptable leakage test of this enclosure be performed at each refueling outage.

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Simple visual inspection will not be sufficient. The requirements for this test are stated in Section 4.5.2 of the technical specifications.

6.3.3.6 Partial Loop Analyses

To support an operating configuration with less than four reactor coolant pumps on the line (partial loop), we require an analysis of the predicted consequences of a loss-of-coolant accident occurring during the proposed partial loop operating mode(s). The applicant submitted an analysis for partial loop operation with one idle reactor coolant pump (three pumps operating). Using a reduced power level of 77 percent of rated power, Babcock and Wilcox performed this analysis assuming the worst case break (8.55 square feet double-ended, $C_D = 1$) and maximum linear heat generation rate allowed by technical specifications for this mode of operation. Based on a sensitivity study referred to by the applicant in a letter dated May 1976, the break selected was located in the active leg of the partially idle loop. Placing the break at the discharge of the pump in an active cold leg of the partially idle loop (instead of at the discharge of the pump in an active cold leg of the fully active loop) yields the most degraded positive flow through the core during the first half of the blowdown and results in higher cladding temperatures. The maximum cladding temperature for the one-idle-pump mode of operation was 1675 degrees Fahrenheit, a value which is within the criterion of 10 CFR Part 50.46. Therefore, this analysis may be used to support the applicant's proposed operation with one idle reactor coolant pump.

Since an analysis of the emergency core cooling system cooling performance with one idle reactor coolant pump <u>in each loop</u> has not been submitted, power operation in this configuration will be limited by Technical Specifications to 24 hours.

Single loop operation (i.e., operation with two idle pumps in one loop) is prohibited by current Technical Specifications without notifying the staff. Each proposal for a scheduled single loop test will be considered on a case-by-case basis.

6.3.4 Tests and Inspections

We requested that, prior to issuance of an operating license, a test be conducted at ambient conditions to verify the capability of the emergency core cooling system to operate in the recirculation mode. The applicant has completed confirmation testing of the emergency core cooling system to operate in the recirculation mode. Head loss data gathered onsite for a flow rate from the containment sump equivalent to the maximum capability of one train were compared to predicted values. The predicted values were shown to be conservative head loss estimates. An investigation of the potential for the formation of vortices in the containment sump was conducted using a 1 to 2 scale model offsite. Additional grating on the containment sump was installed in Davis Besse, Unit 1 subsequent to the testing to provide additional assurance that unacceptable vortex formation would not occur. The staff concludes that the emergency

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core cooling system containment sump as designed will be free of unacceptable vortices and that adequate net positive suction head exists to assure that the system will operate as intended.

6.3.5 Conclusions

We have completed our review of the Davis Besse, Unit 1 emergency core cooling system performance analyses and we have concluded that:

(1) The emergency core cooling system minimum containment pressure calculations were performed in accordance with Appendix K to 10 CFR Part 50.

(2) With the modifications described herein, the single failure criterion will be satisfied.

(3) The proposed procedures for long-term cooling after a loss-of-coolant accident are acceptable to the Commission. The implementation of these procedures before startup is required to provide assurance that the emergency core cooling system can be operated in a manner which would prevent excessive boric acid concentration from occurring.

(4) The proposed mode of reactor operation with one idle reactor coolant pump is supported by a loss-of-coolant accident analysis. Operation with one idle pump in each loop is restricted to 24 hours. Requests for single loop operation will be reviewed on a case-by-case basis.

(5) Additional analyses are required within six months from the date of issuance of the operating license to further quantify existing margins.

6.4 Habitability Systems

6.4.1 Radiation Protection Provisions

We stated in our Safety Evaluation Report that our review of the control room habitability following a design basis loss-of-coolant accident indicated that the thyroid dose guidelines could be exceeded.

In Amendment No. 39 to the Final Safety Analysis Report the applicant committed to an acceptance test that will demonstrate the capability of the emergency system to maintain a minimum of one-eighth inch of water gauge positive pressure differential across the control room pressure boundary using a pressurization flow rate of 300 cubic feet per minute. We will require that the make-up flow be periodically verified to be 300 cubic feet per minute plus or minus ten percent. The technical specifications reflect this requirement.

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The applicant provided additional information regarding the airtight door located in the north wall of the control room. We have reviewed the design and have determined that this door will provide sufficient protection against inleakage.

Based on the design changes above, we have recalculated the potential dose to a control room operator after the design basis loss-of-coolant accident and find that the modified design meets the dose guideline values of General Design Criterion 19. We conclude that the radiation protection provisions of the control room are now acceptable.

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7.0 INSTRUMENTATION AND CONTROLS

7.1 General

In our Safety Evaluation Report we stated that the electrical drawings of the reactor protection system, the engineered safety features system and the Class 1E support systems that were submitted in the Final Safety Analysis Report were incomplete in part or were not presented in sufficient detail to verify that the design had been implemented adequately. We required the applicant to submit a final design package for all safety-related equipment in sufficient detail to facilitate our review. Revised final design drawing packages were submitted by the applicant with sufficient detail and permitted us to conduct an independent review. We conclude that the drawings presently docketed in the Final Safety Analysis Report are adequate for an operating license review and are acceptable. Therefore, we consider this matter resolved.

7.2 Reactor Protection System

In our Safety Evaluation Report we stated that we would report the results of our evaluation regarding modification of the reactor coolant system flow sensors in regard to the common pressure sensing line to all four differential pressure transmitters. We have now determined that the system should be modified to reduce the susceptibility of the system to false flow indication in the event of single failure (e.g., a break, leak or plugging of either the high pressure or low pressure sensing line.)

We have informed the applicant of the need to modify the system to reduce the susceptibility to false flow indication. We will review the proposed modifications when the applicant completes its assessment and determines what modifications can be made, and we will require that approved modifications be implemented during or prior to the first refueling outage.

We conclude that until this matter is satisfactorily resolved, the surveillance requirements imposed by the plant technical specifications on the reactor protection system instrumentation (Table 4.3-1) and on the reactor coolant system operational leakage (Section 3.4.6.2) provide an acceptable assurance that breaks or leaks in the sensing lines will be detected. Based on operating experience, we also conclude that for the interim period, plugging of the sensing lines is highly unlikely.

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In our Safety Evaluation Report we stated that we would require the applicant to document the modification of the power interrupt scheme for the control rod drive trip circuit. The applicant has submitted the revised schematics which implement the design as described in our Safety Evaluation Report, and we find the revised design acceptable.

During our drawing review and our site visit in October 1976, and since issuance of our Safety Evaluation Report, we determined that the applicant's separation criteria did not include separation requirements between Class 1E and non-Class 1E wiring inside the Class 1E logic cabinets and in various control panels as identified in Section 7.9.3 of the Safety Evaluation Report. As a result, the applicant was requested to verify that faults, (e.g., grounding, shorting, application of high voltage and or electromagnetic interference (noise) on non-Class IE circuits would not propagate to the safety grade circuits and degrade them below an acceptable level. The applicant agreed to submit test procedures and test results which would demonstrate that such faults would not degrade the safety systems below an acceptable level.

In response to our concern the applicant submitted tests in December 1976 which described various qualification procedures conducted by the Class 1E safety systems suppliers (the Reactor Protection System supplier and the Engineered Safety Features Actuation System supplier). Although these qualification procedures described the methodology used (by analysis and/or test) to qualify several isolation devices, and demonstrated that certain selected pieces of equipment on a component or subchannel level were immune to simulated electromagnetic interference (noise), the information and tests presented did not adequately demonstrate that faults as described above would not degrade the safety systems (as implemented and wired at the plant site) below an acceptable level. Therefore, we found the applicant's response to our concerns regarding the adequacy of the implemented design to be incomplete and unacceptable.

We have requested the applicant to reevaluate its design and verify that faults on non-Class 1E circuits would not propagate to the Class IE circuits and degrade them below an acceptable level. In addition, we requested that the applicant provide the following information for our review:

(1) Provide definition of the maximum credible voltage, current and electromagnetic disturbance that could be imposed in these circuits,

(2) Define fault duration and a description of the fault detection or fault termination devices used to limit faulted conditions (including primary and back up devices, if any),

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(3) A description of the adequacy of the cable wiring and or connectors required to sustain the above faulted conditions without degradation, or which could lead to degradation or faulted conditions in the safety channels, and

(4) Provide test procedures and test results which demonstrate that faults identified in the previous three items would not degrade the installed safety systems below acceptable levels.

In response to our requests, the applicant submitted a test proposal for our review which they will conduct on the above mentioned systems in order to demonstrate that their design as implemented will not degrade the safety systems below an acceptable level. We have reviewed the type of tests that will be performed and the type of faults that the system will be subjected to. We conclude that the proposed tests are acceptable in part. The applicant was advised that in addition to the presently proposed tests, we require that noise tests in accordance with Mil. Standard 19900, Section 4.6.11 (or equivalent) be conducted on the non-Class 1E circuits that interface with the Reactor Protection System, in order to satisfy the objectives of Section 4.6 of IEEE Std. 279-1968. In addition, we require that noise test procedures (identified above) be submitted to the staff and found acceptable four months from issuance of the operating license, and that the applicant complete these tests and submit the test results for our rev prior to startup following the first regularly scheduled refueling outage or no later than 26 months, whichever comes first.

Based on the operating experience of the Reactor Protection System and the Engineered Safety Features System on similar designs and our review of the qualification documentation presently submitted, we conclude that there is sufficient basis to allow power operation for the period stated, conditioned only on the satisfactory resolution of the noise tests requirements identified above. We will review and report the results of our evaluation of the noise test procedures and the test results when submitted.

7.3 Engineered Safety Features Actuation System

7.3.2 Engineered Safety Features Actuation/Basic Logic

We reported in the Safety Evaluation Report that the applicant identified an automatic testing system which will continuously validate operation of all trip logic combinations for all parameters every 34 seconds and annunciate a detected system fault. Subsequently, the applicant identified problem areas (i.e., noise problems and calibration problems) in the automatic testing system and requested to remove the automatic test system from the design. Since this automatic test system is not required for safety, and since manual testing requirements specified in the technical specifications can be performed with the present design, we have accepted the removal of this test feature from the Engineered Safety Features Actuation System design. We requested the applicant to document in the Final Safety Analysis Report and in the final design schematics their design modification which deletes the automatic test system from the Engineered Safety Features Actuation System. In the event the applicant wishes to reinstate this testing feature at some future date, we will require that this design feature be reevaluated and submitted to the staff for approval prior to reinstatement in the Engineered Safety Features Actuation System.

We have now reviewed the applicant's proposed method of disconnecting the automatic test system which includes (1) disconnecting the power supply circuits, (2) disconnecting the interconnecting wiring to the redundant protection channels, and (3) removing the auto test module from the circuit. We have also reviewed the final design schematics which reflect these changes. Based on our above review of the applicant's proposed methods for disconnecting the automatic test system and the revised design schematics, we conclude that the applicant's revised design satisfies our requirements and is acceptable.

7.3.4 Decay Heat Removal Low Pressure to High Pressure Isolation Valves

In the Safety Evaluation Report the applicant was requested to (a) verify that the consequences of inadvertent valve closure of valve DH11 or DH12 during the decay heat removal mode of operation would not degrade the core cooling system below an acceptable level or (b) modify the design to preclude inadvertent valve closure. The applicant proposed a procedure to remove power to these valves during this mode of operation, that is, power would be removed administratively before the decay heat pumps are allowed to start. As discussed in Section 5.5.3 of this report, the staff's position requires that power will be maintained to the decay heat removal valves in order to not compromise the isolation needed during operation between high and low pressure piping. Section 5.5.3 discussed license conditions regarding the submittal and implementation of acceptable design alternatives to preclude the possibility of damage to the decay heat removal pumps from an inadvertent closure of valves DH11 or DH12 during decay heat removal operation.

7.3.5 Core Flooding Tank Isolation Valves

We stated in the Safety Evaluation Report that the design of the core flooding tank isolation valves was acceptable conditioned only on the satisfactory verification of the implementation of the design during the site visit. We have completed the site visit and found no inconsistencies with what was stated in our Safety Evaluation Report. We therefore conclude that the implementation of this design is acceptable.

7.4 Systems Required for Safe Shutdown

7.4.1 Steam and Feedwater Line Rupture Control System

We specified in the Safety Evaluation Report those areas where the designs of the steam and feedwater line rupture control system did not conform to the applicable safety criteria.

The applicant, at our request, has modified the design and responded to our safety concerns for the steam and feedwater line rupture control system. The following items identify those areas that were found unacceptable in the Safety Evaluation Report and describe subsequent resolution to our safety concerns.

(1) Item (1), Page 7-8, of the Safety Evaluation Report required that the design of the anticipatory trip inputs to the steam and feedwater line rupture control system (e.g., inputs from the integrated control system) be modified to satisfy the requirements of the Institute of Electrical and Electronic Engineers Standard 279-1971 if they are to remain as inputs to the steam and feedwater line rupture control system or remove them if they are not required for safety. The applicant elected to delete these trip functions from the steam and feedwater line rupture control system design. We conclude that since credit is not taken for these trips in the accident analysis and since they are not required for safety, this modified design is acceptable. We have reviewed the various final design schematics that implement this change and the information documented in the Final Safety Analysis Report. We conclude that the design satisfies the staff's requirements stated in Section 7.1 of the Safety Evaluation Report and is acceptable.

(2) Item (2), Page 7-8, of the Safety Evaluation Report identified areas where the design of the steam and feedwater line rupture control system did not conform to the requirements of the Institute of Electrical and Electronic Engineers Standard, 279-1971 Sections 4.2 and 4.16. The applicant provided a modified design and revised final design schematics to demonstrate full conformance with Sections 4.2 and 4.16 of the reference standard. We have reviewed the final design drawings and conclude that the design meets the staff's requirements stated in Section 7.1 of the Safety Evaluation Report and is therefore acceptable.

(3) Item (3), Page 7-8, of the Safety Evaluation Report identified two normally open valves (i.e., HV599 and HV608) on the discharge side of the auxiliary feed pumps (one in each loop), which if failed closed would preclude adequate system function. The applicant has provided a modified design for these valves. The modified design provides automatic closure of these valves only under specific accident conditions (i.e., in the event of a steam line break the valve supplying auxiliary feedwater to the degraded steam generator would automatically close). The circuit design for these valves includes interlocks to assure that a single electrical failure would not cause closure of the valve to the intact steam generator. In addition, redundant and independent position indication for these valves to alert the operator of their status at all times have been included in the design. We have reviewed the modified final schematics and also verified the implementation of the design of these valves during our site visit in October 1976. We conclude that the design satisfies the staff's requirement stated in Section 7.1 of the Safety Evaluation Report and is acceptable.

Also, during our continuing review of the design concerning the single failure criterion as it relates to electrically-operated active and passive components, four additional valves (i.e., DH1A, DH1B, DH14A, DH14B) were identified which, if failed closed during accident conditions, would preclude adequate core cooling. The applicant was requested and submitted a modified design for these valves which conforms to the requirements stated in Branch Technical Position EICSB 18 of Appendix A in the Standard Review Plan. We have reviewed the modified final design schematics and also verified the implementation of their design during our site visit in October 1976. We require that the technical specifications identify the valves which are to remain open with power removed during reactor operations. We have concluded that this design satisfies the Commission's requirements referenced above and is acceptable.

(4) Item (4), Page 7-9, of the Safety Evaluation Report identified the staff's requirements regarding testability of safety-related "blind sensors." In Amendment 39 to the Final Safety Analysis Report, the applicant identified a proposed modification to his existing design. The modified design will replace the four level switches (i.e., blind sensors) in each steam generator inside containment with four level transmitters. These sensors will have continuous indication of the measured variable displayed inside the control room. In addition bistable trip relays will be located inside the steam and feedwater rupture control system logic to actuate the trip logic whenever the predetermined setpoints are reached. This modification will be incorporated prior to Mode 2 (Startup), to facilitate the staff's requirements on testing without necessitating plant shutdown. In the interim the applicant has committed to periodically calibrate the blind sensors every three months. Monthly checks will be conducted in accordance with the requirements stated in Section 4.9 of IEEE Std 279-1971. Based on our review of steam and feedwater rupture control system design and the applicant's commitment to comply with the staff's requirements, we conclude that the design satisfies the Commission's requirements and is acceptable.

(5) Item (5), Page 7-9, of the Safety Evaluation Report identified a commitment by the applicant to remove the override interlocks which would automatically shut and inhibit the steam inlet valves to the auxiliary feed pump turbine (i.e., HV016, HV107, HV106A, and HV107A) from opening whenever containment pressure exceeded a preselected setpoint of 38.5 pounds per square inch gauge.

Modified final design schematics for these valves were submitted which deleted these interlocks. We have reviewed selected final design drawings and conclude that this modification of the circuit design for these valves satisfies the staff's requirements identified in Section 7.1 of this report and is acceptable.

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(6) During our review the applicant was requested to address the staff's concerns regarding loss of all alternating current power to this system. In response to the staff's concern the applicant committed to modify the design and provide a diverse power source (i.e., direct current power) to selected motor operated valves (HV 106, HV 306, and HV 3870) in one redundant auxiliary feedwater train to assure that the plant can be safely shut down in the event of loss of all alternating current power. Final schematics of this design modification were submitted. The applicant has stated that this design change will be implemented prior to the second cycle of fuel operation (see Section 9.3.5 of the Safety Evaluation Report). We have reviewed these changes and the final schematics which describe how the design will be implemented. We conclude that this modification satisfies the staff requirements and is acceptable.

Our review of the unresolved items identified in the Safety Evaluation Report for the steam and feedwater line rupture control system has now been completed. Also, we have reviewed the applicant's design changes for providing diverse power in one redundant auxiliary feedwater train to assure that the plant can be safely shutdown in the event of loss of all alternating current power.

Our review of the applicant's modified steam and feedwater line rupture control system, the final design schematics, and verification for implementation of the modified design has been delineated in our evaluation as stated above.

Based on our review, we conclude that the design of the steam and feedwater line rupture control system conforms to the staff's requirements, and is acceptable.

7.5 Safety-Related Display Instrumentation

During the course of our review, and subsequent to the issuance of our Safety Evaluation Report, we indicated to the applicant that the manual initiation of system level inoperable status or bypass indication did not fully meet the objective of Regulatory Guide 1.47, Section C.4, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems," and the design was not acceptable. Specifically, manual initiation of systems level inoperable status or bypass indication for containment isolation was not provided. The applicant was requested and agreed to provide manual initiation of inoperable status indication for this subsystem and to review his design to assure that manual initiation of inoperable status or bypass indication is provided for all safety-related systems.

The applicant modified the design and provided two additional manual initiation of inoperable status or bypass indication for the following subsystems: (1) Containment Isolation and (2) Steam Generator Isolation.

The applicant revised the information in the Final Safety Analysis Report to reflect this change. The present design includes bypass status indication for the following subsystems:

(1) Auxiliary Feedwater System,
(2) Component Cooling System,
(3) Service Water System,
(4) High Pressure Injection System,
(5) Low Pressure Injection System,
(6) Containment Spray System,
(7) Core Flooding System,
(8) Emergency Ventilation System,
(9) Borated Water Storage System,
(10) Containment Air System,
(11) Containment Radiation System,
(12) Control Room System,
(13) Containment Isolation System, and
(14) Steam Generator Isolation System.

We have reviewed the modified design and conclude that the design satisfies the Commission's requirements stated in Section 7.1 of the Safety Evaluation Report and is acceptable.

7.7 Environmental Qualification

We stated in our Safety Evaluation Report that, subject to satisfactory resolution of the qualification of safety-related equipment required to be operable following a postulated main steam line break accident environment, we found the environmental qualification of the safety-related equipment to be acceptable. Our concerns and resolution regarding the main steam line break accident environment is addressed in Section 6.2.1 of this supplement.

In a recent amendment to the Final Safety Analysis Report, the applicant identified additional safety-related instrumentation inside containment (steam generator level transmitters) and outside containment (containment pressure transmitters). The applicant was requested and agreed to supplement the information in the Final Safety Analysis Report and describe the qualification tests performed on the steam generator level transmitters and the containment pressure transmitters and submit the test results and procedures used to qualify this equipment. The applicant amended the Final Safety Analysis Report and submitted the information requested. We have reviewed the equipment qualification procedures for the steam generator level transmitters and conclude that the qualification environments that the equipment was submitted to during the test was substantially in excess of the required environmental envelope as stated in the Final Safety Analysis Report and is acceptable. Also, the applicant has submitted the qualification tests for the

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containment pressure transmitters, the applicant has documented that these transmitters will be located outside containment and subjected only to a worst case environment of 120 degrees Fahrenheit at 100 percent relative humidity.

Based on the relatively low ambient requirements imposed on these sensors and our review of the qualification tests submitted, we have reasonable assurance that the equipment will perform its intended function in these environments and is acceptable. We conclude that the environmental qualification of safety-related equipment is acceptable subject to any new concerns which may be identified in our generic review of the main steam line break accident (see Section 6.2.1 of this Supplement).

7.9 Cable Separation and Identification Criteria

7.9.2 Separation Criteria Between Redundant Class 1E Circuits in Metal Conduits

7.9.2.1 Wireways

During our review of the applicant's separation criteria we determined that wireways (metal troughs with covers) were used in the cable spreading areas. Several meetings were conducted with the applicant to determine the adequacy and implementation of the wireways. The applicant documented their separation criteria for wireways routed in close proximity with other redundant Class 1E raceways (i.e., ladder type trays, wireways and conduit) and with non-Class 1E raceways (i.e., channels A, B, and C). In addition to providing thermal insulating blankets on all open type trays (as described in Section 7.9.2.2 below) the applicant identified that certain cables routed in these wireways were different from the cables tested for flame retardancy as described in Section 7.9.1 of our Safety Evaluation Report. For these wireways the applicant will inject silica gel (into) the wireway in order to encase these cables with a flame retardant material. Based on our review of the criteria established for safety circuits routed in wireways, and the additional protective measures incorporated by the applicant, we conclude that this design is acceptable.

7.9.2.2 Conduits

In our Safety Evaluation Report we identified areas where the separation between redundant circuit routed in metallic conduit was inadequate and the information was not sufficient to complete our review.

The applicant has amended the information in the Final Safety Analysis Report describing the separation criteria for routing Class 1E circuits in conduit and has documented their separation criteria for these circuits when crossing open tray type raceways. In addition to the separation distances provided, the applicant documented that all open trays (with certain justified exceptions, i.e., trays inside containment) will be covered with flame retardant insulating blankets to minimize flame propagation or ignition. The adequacy of using the specific type of thermal blankets will be reviewed as part of our post license review of the plant fire protection to meet levels recommended in Appendix A to the Standard Review Plan 9.5.1, Revision 2 as discussed in Section 9.6.1 of this report.

The applicants recently amended minimum separation criteria for redundant Class 1E circuits routed in metallic conduits allows less than one inch of free air space between redundant conduits with no provisions for barriers other than the conduit itself. In order to justify the adequacy of this amended design, the applicant committed to demonstrate by test that a single failure such as a fault (i.e., noise, voltage surge, short circuit or ground) imposed in one Class 1E circuit routed in these conduits, would not degrade the redundant Class 1E circuitry routed in the redundant conduits below an acceptable level.

The applicant submitted their final test results, analysis, and the test procedures for the tests which were conducted at our request to justify the adequacy of their design. We have reviewed these procedures, the test results, and the adequacy of the methods that were used on these circuits to simulate abnormal conditions.

Based on our review of these tests and the separation requirements established at the construction permit stage of review, we conclude that the design for redundant Class 1E cables in metallic conduit as implemented at Davis Besse, Unit 1 satisfies the objectives of General Design Criterion 22 and is acceptable.

Although the applicant's design criteria was compared to the recently established separation requirements, we do not believe that the incremental safety margins which would be achieved by these requirements warrants backfitting the Davis Besse, Unit 1 design to the new standards.

Based on our review of the separation criteria for circuits routed in metal conduit, we conclude that the design is acceptable.

7.9.3 Separation Criteria Between Redundant Class 1E and Non-Class 1E Circuits Within Enclosures

In the Safety Evaluation Report we identified concerns regarding the applicant's criteria between redundant Class 1E and non-Class 1E circuits within enclosures. We have reviewed the final test results regarding the flame tests conducted by the applicant and conducted a site visit in October 1976 to review the as-installed designs. Based on the staff's evaluation during the site visit, we could not support the adequacy of the design as implemented and requested the applicant to provide additional barriers in these installations to assure that Class 1E circuits are adequately separated.

Subsequently, additional criteria for this installation was documented by the applicant in the Final Safety Analysis Report. The modified criteria for these

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enclosures provides additional fire stops and barriers on the top of the cabinets and at intercabinet junction points. In addition the cables on the bottom of the cabinets will be coated with silicon rubber. Also, smoke detectors are installed in specified panels to detect fircs.

Based on review of the modified criteria, and the demonstrated degree of flame retardancy of the cable used in this installation, we conclude that the design satisfies the staff's requirements and is acceptable.

7.10 Electrical Penetrations

In our Safety Evaluation Report we stated that the applicant was requested to supplement the information in the Final Safety Analysis Report and provide their justification and basis to assure that the design of the electrical penetrations satisfies the requirements stated in General Design Criterion 50, "Containment Design Basis." In response, the applicant documented short circuit test results which were conducted on their medium and low voltage penetrations that demonstrate that these penetration assemblies can withstand, without loss of mechanical integrity, the maximum possible fault current versus time conditions. In addition, the applicant submitted analyses which demonstrate that the primary and back-up protective relaying used in these circuits are designed to interrupt power in sufficient time to preclude electrical penetration damage in the event of faults in these circuits. Also, during our review the applicant was requested to verify that the operation of the primary and back-up protective relaying used in these circuits would not be negated assuming a single failure in the supply power to these breakers. In response to our concern the applicant identified that only the 13.8 kilovolt breakers require power (i.e., direct current power) to isolate the reactor coolant pumps from their motor control centers, and committed to modify their design by providing independent direct current power sources to the respective protective breakers. The modified design will supply direct current power to the primary breakers from the direct current distribution panels "DAP" and "DBP" and will supply direct current power to the back-up breakers from direct current distribution panels "DAN" and DBN." Each distribution panel is supplied by an independent battery and battery charger.

Based on our review of the test results, the analysis design modifications and the various final design schematics, we conclude that the design of the electrical penetration protection is acceptable.

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8.0 ELECTRIC POWER

8.2 Offsite Power Systems

We stated in our Safety Evaluation Report that we had requested the applicant to evaluate the Davis Besse, Unit 1 design for the Class IE electrical distribution system to determine whether the operability of safety-related equipment, including associate circuitry and instrumentation, can be adversely affected by short term or long term degradation in the offsite power system as recently experienced at the Millstone Unit 2 plant.

In response to our requests for information, the applicant submitted a response which did not address all of our concerns and we have requested additional information from the applicant. A condition to the operating license for Davis Besse, Unit 1 will stipulate that within four months from the issuance of the license, Toledo Edison Company shall provide to the Commission an acceptable evaluation to assure that the facility design will provide adequate breaker coordination and isolate its onsite system in sufficient time to permit the required Class IE equipment to operate in the event of offsite grid degradation. Prior to the Commission's approval of the modification, Toledo Edison Company shall maintain the normal operating range for the grid system voltage between 98.3 percent to 102.2 percent of rated voltage with corresponding safety-related bus voltages as defined in Attachment 1 of Toledo Edison Company's letter to the Commission dated November 13, 1976. In the event the system conditions exceed these values, Toledo Edison Company shall proceed in an orderly manner to reduce load to five percent of rated power and take corrective action to stabilize the system within the values stated above.

We conclude that with the condition noted above, adequate operating procedures will be followed such that offsite power degradation during this period will not cause an adverse effect on the health and safety of the public.

9.0 AUXILIARY SYSTEMS

9.6 Other Auxiliary Systems

9.6.1 Fire Protection System

We stated in our Safety Evaluation Report that as a result of investigations and evaluations being conducted by the staff on nuclear power plant fire protection systems, additional requirements might be imposed on Davis Besse Unit 1 to further improve the capability of the fire protection system.

On February 11, 1977 the applicant submitted its Fire Hazard Analysis Report in response to our September 30, 1976 letter transmitting Appendix A to Branch Technical Position 9.5.1 and our request for a fire hazard analysis and a reevaluation of the fire protection program for Davis Besse Unit 1.

Our review of the applicant's Fire Hazard Analysis Report indicated that the submittal was not adequate for determining the fire protection program of the facility in accordance with Appendix A to Branch Technical Position 9.5.1.

We have notified the applicant that a condition to the operating license for Davis Besse Unit 1 will stipulate that: The licensee shall increase the level of fire protection in the facility to the levels recommended in Appendix A to Standard Review Plan 9.5.1, Revision 2, "Fire Protection System," or with alternatives acceptable to the staff within three (3) years from issuance of the operating license except that prior to startup following the first regularly scheduled refueling outage, the licensee shall implement Section B of Appendix A, "Administrative Procedures, Controls and Fire Brigade," and Section C of Appendix A, "Quality Assurance Program."

We reaffirm our conclusions as stated in our Safety Evaluation Report that, based upon our review of the facility fire protection design at this time, we conclude that for the interim, the facility fire protection system is acceptable.

Our review of the applicant's reevaluation of its Fire Hazard Analysis Report and any required modifications to the facility fire protection system will be reported in a future report.

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12.0 RADIATION PROTECTION

12.6 Radioactive Material Safety

We were informed by the applicant that sealed neutron startup sources (Americium-Beryllium-Copper) were exceeding the leakage test limits specified in the technical specification 3/4.7.8.1.

Although the sources exceed the leakage test limits of the technical specifications (< .005 μCi), the leakage activity was determined not to be leaking from any source itself, but from Protactinium-233, an activation product of the Thorium-232 impurity of the tungsten welding material used in the source fabrication.

Since start-up sources are placed in the reactor, it is expected that the surface contamination from the pre-reactor irradiated neutron source will be negligible when compared to the surface contamination of the post-irradiated neutron source upon removal from the reactor for radioactive waste disposal.

Based on our evaluation above, and our evaluation of the applicant's radioactive material safety program to assure the safe storage and handling of radioactive sources (see Section 12.6 of the Safety Evaluation Report), a condition to the operating license will stipulate that the applicant shall be exempted from technical specification 3/4.7.8.1 for the Americium-Beryllium-Copper startup sources until such time as the sources are replaced.

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13.0 CONDUCT OF OPERATIONS

13.6 Industrial Security

The applicant has revised the Industrial Security Plan for protection of the facility from industrial sabotage since the issuance of the Safety Evaluation Report. These revisions were identified by letter dated February 2, 1977 and by Revisions 4, 5, and 6 submitted to the Commission on November 15, 1977, February 18, 1977, and March 24, 1977, respectively.

We have reviewed the revisions to the Industrial Security Plan and have determined that these revisions do not decrease the provisions for industrial security previously reviewed and found to be acceptable as stated in our Safety Evaluation Report. On this basis we confirm our conclusion that the Industrial Security Plan is acceptable.

We will require that the applicant submit an amended physical security plan in compliance with the requirements of 10 CFR Part 73.55 (effective February 24, 1977). The applicant's submittal for an amended industrial security plan pursuant to 10 CFR Part 73.55 must be submitted on May 25, 1977.

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15.0 ACCIDENT ANALYSES

15.2 Thermal and Hydraulic Analyses

15.2.2 Accidents

We requested that the closure times of steam and feedwater isolation valves assumed in the accident analyses be periodically verified. The proposed Technical Specifications were reviewed by the staff and are acceptable.

Also, for the main steam line break and feedwater line break analyses, we requested that the applicant further examine the potential for single active component failures, such as an isolation valve failure or the opening of an atmosphere vent valve. Additional information was submitted by the applicant. We have reviewed the information and we find the modified description and analyses submitted in the Final Safety Analysis Report in conjunction with the responses to our questions provides an acceptable evaluation of these events.

15.3 Radiological Consequences of Accidents

15.3.1 Loss of Coolant Accident

We have reevaluated the radiological consequences of the postulated loss-of-coolant accident based on the revised relative concentration values (1974 to 1976) and the revised depressurization time (8J2 seconds) in the annulus space as described in Sections 2.3.4 and 6.2.3 of this supplement, respectively. The assumptions used for reevaluating this accident are provided in Table 15.1 and the resulting doses are shown in Table 15.3 of this supplement. We find the recalculated doses are within the guideline values of 10 CFR Part 100 and, therefore, are acceptable.

We have reevaluated the post-loss-of-coolant accident for hydrogen purging using the revised 4-30 day relative concentration value (8.0×10^{-7} seconds per cubic meter) at the low population zone boundary (3200 meters). The assumptions used in calculating this accident and the resulting doses are provided in Table 15.2 of this supplement. We find these doses remain unchanged from those calculated at the time our Safety Evaluation Report was issued and are well within the guideline values of 10 CFR Part 100 and, therefore, are acceptable.

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15.3.2 Steam Line Break, Steam Generator Tube Rupture, and Control Rod Ejection Accidents

For the steam line break and steam generator tube rupture accidents there are no changes in either the assumptions used (Table 15.4) or the calculated dose results (Table 15.6) as reported in our Safety Evaluation Report. Therefore, based on our conclusions as

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TABLE 15.1

ASSUMPTIONS USED TO ESTIMATE
RADIOLOGICAL CONSEQUENCES DUE TO A
POSTULATED LOSS OF COOLANT ACCIDENT
AT DAVIS BESSE UNIT 1

Power level, megawatts thermal	2772
Operating time, years	3
Primary Containment Leak Rate, percent per day	0.5 to 24 hours 0.25 greater than 24 hours
Fraction of Core Inventory Available for Leakage from Containment:	
Noble Gases	100 percent
Iodine	25 percent
Bypass Leakage Fraction, percent of Primary Containment Leak Rate	
0-802 sec.	100 percent
802 sec. to 30 days	3 percent
Primary Containment Free Volume, cubic feet	2.834×10^{6}
Iodine Form Fractions, percent	
Elemental	91
Particulate	5
Organic	4
Filter Efficiencies for Iodine Forms, percent	
Elemental	95
Particulate	90
Organic	95
Spray Removal Rates, per hour	
Elemental (Effective to 1.1 hours)	0.5
Particulate	0.2
Relative Concentrations, seconds per cubic meter	
0-2 hours at 732 meters	2.2×10^{-4}
0-8 hours at 3200 meters	8.2×10^{-6}
8-24 hours at 3200 meters	5.7×10^{-6}
24-96 hours at 3200 meters	2.6×10^{-6}
96-720 hours at 3200 meters	8.0×10^{-7}

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TABLE 15.2

HYDROGEN PURGE DOSE

The ...tions used to calculate the low population zone doses due to post-loss-of-coolant accident hydrogen purging are:

Power Level: 2772 megawatts thermal

Containment Volume: 2.83×10^6 cubic feet

Purge Time: 30 days

Holdup Time Prior to Purging: 24 days

Purge Rate: 47 cubic feet per minute

Charcoal Filter Efficiency of 95 percent and 95 percent for Elemental and Organic Iodine, respectively

X/Q Value: 4-30 days (8.0×10^{-7} seconds per cubic meter)

Estimated Consequences

Low Population Zone Boundary (3200 meters)	Doses, Rem	
	Thyroid	Whole Body
	11	< 1

TABLE 15.3

ESTIMATED LOSS OF COOLANT ACCIDENT DOSE RESULTS

	Doses, Rem	
	Thyroid	Whole Body
Exclusion Area Boundary (732 meters)		
0-802 seconds	163.6	1.73
802 seconds - 2 hours	115.8	4.68
	279.4	6.41
Low Population Zone Boundary (3200 meters)		
0-802 seconds	6.08	< 1
802 seconds - 8 hours	10.56	< 1
8-24 hours	3.76	< 1
1-4 days	3.38	< 1
4-30 days	2.74	< 1
0-30 days	26.52	< 1

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TABLE 15.5

ASSUMPTIONS FOR CONTROL ROD EJECTION ACCIDENT

Case I*

1.	Core Power Level	2772 Megawatts thermal
2.	Fuel Failed in Transient	28 percent
3.	Iodine and Noble Gas Inventory in Gap of Failed Fuel	10 percent
4.	Gap Activity in Failed Fuel Released to Containment Building	100 percent
5.	Plate-out of Radioactive Iodines	50 percent
6.	Containment Building Leak Rate for 24 hours	0.50 percent
7.	Containment Building Leak Rate after 24 hours	0.25 percent
8.	Containment Building Sprays are not Initiated	------------
9.	Filter Efficiencies for Iodine Forms for time periods greater than 802 seconds	
	Elemental	95 percent
	Particulate	90 percent
	Organic	95 percent
10.	Relative Concentrations, seconds per cubic meter	
	0-2 hours at exclusion area boundary - 732 meters	2.2×10^{-4}
	0-8 hours at low population zone boundary - 3200 meters	8.2×10^{-6}
	0-24 hours at low population zone boundary - 3200 meters	5.7×10^{-6}

Case II**

1.	Core Power Level	2772 Megawatts thermal
2.	Fuel Failed in Transient	28 percent
3.	Iodine and Noble Gas Inventory in Gap of Failed Fuel	10 percent
4.	Gap Activity in Failed Fuel Released to Reactor Coolant	100 percent
5.	Reactor Coolant to Secondary Coolant Operational Leakage	1 gallon per minute
6.	Time to Reach Reactor Coolant-Secondary Coolant Equilibrium	16 min
7.	Loss of Offsite Power so that Steam is Released from Secondary Side Relief Valve	

*Case I Releases through the containment vessel

**Case II Releases through secondary system

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TABLE 15.6

DOSE RESULTS FOR STEAM LINE BREAK, STEAM GENERATOR TUBE RUPTURE AND CONTROL ROD EJECTION ACCIDENTS

ACCIDENTS	THYROID (REM) (732 METERS)	WHOLE BODY (REM) (732 METERS)
Tube Rupture Accident	1.5	< 1.0
Tube Rupture Accident with Coincident Iodine Spike	12.0	< 1.0
Steam Line Break	< 1.0	< 1.0
Loss of Offsite Power	< 1.0	< 1.0
Loss of Offsite Power with Coincident Iodine Spike	< 1.0	< 1.0
Rod Ejection Accident		
Case 1* (Exclusion Area Boundary)	5.0	< 1.0
Case 2** (Exclusion Area Boundary)	42.0	< 1.0
	(3200 METERS)	(3200 METERS)
Case 1* (Low Population Zone Boundary)	1.0	< 1.0
Case 2** (Low Population Zone Boundary)	< 1.0	< 1.0

*Releases through the containment vessel

**Releases through the secondary system

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stated in Section 15.3.2 of the Safety Evaluation Report, we find the calculated doses for the radiological consequences of the steam line break and steam generator tube rupture accidents remain acceptable.

We have reevaluated the radiological consequences of the control rod ejection accident based on the revised relative concentration values and depressurization time in the annulus space. The revised assumptions used and the recalculated doses are shown in Tables 15.5 and 15.6, respectively, of this report. The reevaluated dose results for the control rod ejection accident remain well within the guideline values of 10 CFR Part 100, and are acceptable.

15.3.3 Radiological Consequences of a Postulated Fuel Handling Accident

We have reevaluated the fuel handling accident using the revised zero to two hour relative concentration values at the exclusion area boundary and the low population zone boundary. The revised assumptions used for reevaluating the fuel handling accident and the estimated radiological consequences of this accident are provided in Table 15.7 of this report. We find that the reevaluated doses remain unchanged from those values presented in Table 15.7 of our Safety Evaluation Report and reaffirm our conclusions stated in that report that the potential doses calculated for the fuel handling accident are well within the guideline values of 10 CFR Part 100, and are, therefore, acceptable.

15.3.4 Waste Gas Decay Tank Accident

We have reevaluated the consequences of a postulated gas decay tank accident based on the revised relative concentration values. The assumptions used and the calculated dose results are shown in Table 15.8 of this report. We find that the recalculated doses remain unchanged from those presented in Table 15.8 of our Safety Evaluation Report and reaffirm our conclusions that the doses for the waste gas decay tank accident are well within the guideline values of 10 CFR Part 100, and are acceptable.

TABLE 15.7

ASSUMPTIONS FOR AND CONSEQUENCES OF A POSTULATED FUEL HANDLING ACCIDENT

Power Level	2772 Megawatts-thermal
Power Peaking Factor	1.7
Operating Time	3 years
Number of Rods Failed	208
Number of Rods in Core	38,816
Fraction of Inventory in Gap:	
Noble Gases	10 percent
Iodines	10 percent
Effective Iodine Decontamination Factor in Pool	100 percent
Filter Efficiencies:	
Elemental Iodine	90 percent
Organic Iodine	70 percent
Iodine Fractions Leaving Pool	
Elemental	75 percent
Organic	25 percent
Shutdown Time	72 hours
Q/X Relative Concentration Values	
0 - 2 hours at 732 meters	2.2×10^{-4} seconds per cubic meter
0 - 2 hours at 3200 meters	8.2×10^{-6} seconds per cubic meter

Estimated Consequences:

	Dose, Rem Thyroid	Dose, Rem Whole Body
Exclusion Area Boundary (732 meters)	9	< 1
Low Population Zone Boundary (3200 meters)	< 1	< 1

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TABLE 15.8

ASSUMPTIONS FOR AND CONSEQUENCES OF
A POSTULATED GAS DECAY TANK ACCIDENT

Gas Decay Tank Ruptures

The assumptions used to calculate the offsite doses from a gas decay tank rupture were:

(1) Gas decay tank contains one complete primary coolant loop inventory of noble gases resulting from operation with 1 percent failed fuel (94,000 curies of noble gases).

(2) The release is complete within 2 hours.

(3) Meteorological assumptions are the same as for the loss-of-coolant accident.

	Dose, Rem	
	Thyroid	Whole Body
Estimated Consequences:		
Exclusion Area Boundary (732 meters)	Negligible	< 1
Low Population Zone (3200 meters)	Negligible	< 1

18.0 REVIEW BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

A Subcommittee of the Advisory Committee on Reactor Safeguards (Committee) considered the application for an operating license for Davis Besse, Unit 1 on December 21, 1976 in Washington, D. C. The full Committee completed its review of the application at its 201st meeting on January 6, 1977. A copy of the Committee's report dated January 14, 1977 is attached as Appendix E. The following paragraphs discuss the current status of each item on which the Committee commented or made recommendations in that report.

(1) The Committee indicated that the structures and components of Davis Besse, Unit 1, were designed for a safe shutdown earthquake (SSE) acceleration of 0.15g at the foundation level. Because of changes in the regulatory approach to selection of seismic design bases, the Committee believes that an acceleration of 0.20g would be more appropriate for the SSE acceleration at a site such as this in the Central Stable Region. The applicant presented the results of preliminary calculations concerning the safety margins of the plant for a safe shutdown earthquake acceleration of 0.20g. The Committee recommended that the NRC staff review this aspect of the design in detail and assure itself that significant margins exist in all systems required to accomplish safe shutdown of the reactor and continued shutdown heat removal, in the event of a safe shutdown earthquake at this higher level. The Committee believed that such an evaluation need not delay the start of operation of Davis Besse, Unit 1.

By letter dated January 31, 1977, the NRC staff requested that the Committee verify the correctness of the staff's interpretation to item (1) as specified above. By letter dated February 17, 1977, the Committee responded to the NRC staff request as follows:

(a) "In the Committee's opinion "currently accepted procedures for deconvolution" are not acceptable at this site for structures founded on rock. It is the Committee's opinion that Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," should be applied at the foundation level of such structures.

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(b) "The systems to be investigated should include those needed for continued shutdown heat removal as well as those "systems needed for safe shutdown."

(c) "In view of the fact that the review is being made for a plant designed to different criteria, and already constructed, the responses of the components and systems reviewed need not be strictly "within current acceptance limits."

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Instances in which those limits are exceeded may be considered acceptable on a judgment basis, with due consideration to the contribution of seismic motions to the overall response."

By letter dated February 24, 1977, the applicant submitted an alternative proposal for determining the seismic rock motion for the Davis Besse site. We have evaluated the applicant's submittal and find the applicant's proposal to be unacceptable. We are preparing a position to be issued to the applicant which will specify the scope of review required to address the Committee's recommendations for the Davis Besse, Unit 1 seismic review.

We agree with the Committee that the forthcoming seismic evaluation need not delay the start of commercial operation for Davis Besse, Unit 1.

See Section 3.7.1 of this supplement for the status on Item 1 above.

(2) The Committee stated that the performance of the emergency core cooling system has been evaluated using a Babcock and Wilcox evaluation model applicable to the raised-loop configuration. The NRC staff has reviewed these evaluations and has determined that certain assumptions regarding return to nucleate boiling do not comply strictly with the provisions of Appendix K to 10 CFR Part 50. The NRC staff is also reviewing several other areas relating to emergency core cooling system performance. These matters should be resolved in a manner satisfactory to the NRC staff.

See Section 6.3 of this supplement for the status on return to nucleate boiling.

See Section 6.3 of this supplement for the status on other areas of the emergency core cooling system.

(3) The Committee noted that in conjunction with the evaluation and assessment of the impact of routine waste releases from this plant, the Committee recommends that the NRC staff provide leadership in encouraging the development of improved environmental radiation surveillance capabilities on the part of the State of Ohio and appropriate local regulatory agencies.

The Commission's licensees have the responsibility for evaluating the impact of their facility on the environment and for assuring that individuals near the facility do not receive radiation doses in excess of design objectives and applicable limits. In some cases, state agencies conduct environmental surveillance around nuclear facilities which gives an independent assessment of a facilities offsite radiological impact. However, state surveillance, which is not generally as extensive as that of the licensees, is not a requirement for the issuance of an operating license. A licensee's program must be adequate to assure that significant pathways are being monitored.

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By letter dated December 14, 1973 the Commission contacted the State of Ohio about participating in the Commission's program. As yet, we have had no response from the State of Ohio. Although the Commission does provide limited technical support to any state to upgrade their technical capability, the Commission cannot provide sufficient aid to develop a capable surveillance program for the State of Ohio without adequate state funding and a commitment from the State of Ohio.

(4) The Committee noted that post-accident operation of the plant to maintain safe shutdown conditions may be dependent on instrumentation and electrical equipment within containment which is susceptible to ingress of steam or water if the hermetic seals are either initially defective or should become defective as a result of damage or aging. The Committee believes that appropriate test and maintenance procedures should be developed to assure continuous long-term seal capability.

By letter from M. Bender of the Committee to M. Rowden of the Commission dated February 24, 1977, the Committee added Item 4 above, to their list of generic items as Item II D-2.

This matter will be dealt with on this plant and others when a final generic solution is developed.

(5) The Committee stated that, prior to commercial power operation of Davis Besse, Unit 1, additional means for evaluating the cause and likely course of various accidents, including those of very low probability, should be in hand in order to provide improved bases for timely decisions concerning possible offsite emergency measures.

We requested that the Committee provide clarification of this matter. By letter dated February 17, 1977, the Committee stated that Item 5 above was intended to address the generic item "Instrumentation to Follow the Course of an Accident" (Item II-11 of ACRS report on Generic Items dated April 16, 1976). The Committee further stated that:

(a) "Appropriate emphasis should be given to the resolution of item II-11 so that resolution is accomplished prior to commercial operation of the Davis Besse, Unit 1 plant.

(b) "Additional attention should be given to development of procedures to be used by the plant operators in taking appropriate corrective action for incidents and accidents with very low probabilities of occurrence.

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The Committee plans to give further attention to the development of procedures to cope with operating events of intermediate probability to limit their consequences on the public health and safety. This will be done on a generic basis, however."

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Item (a) above is discussed in Section 7.5.1 of the Safety Evaluation Report. It is being considered as a generic issue (Item II-11, Status Report on Generic Items, see Appendix D of this report) and as such will be dealt with on this plant and others when a final generic solution is developed.

Item (b) above will be dealt with on this plant and others when a final generic solution is developed.

(6) The Committee noted that anticipated transients without scram remains an outstanding issue pending our review of the Babcock and Wilcox generic analyses, and recommended early resolution of this matter in a manner acceptable to us.

The discussion in Section 7.2.2 of the Safety Evaluation Report notes that considering the probability of occurrence of the event in question, we conclude that limitations on operation on this account are not necessary or appropriate until such time as any facility modifications found necessary by our generic review are finalized and can be implemented.

(7) The Committee noted that Davis Besse, Unit 1, has installed a bypass loop containing two manually operated valves around the decay heat removal system suction line isolation valves. The normally closed bypass valves would be opened in the event of a spurious closure of one of the decay heat removal system suction line isolation valves during system operation. The Committee recommended that further attention be given to the means employed for isolation of the low pressure residual heat removal system from the primary system while the latter is pressurized, and that reliable means be developed to assure such isolation. The Committee stated that this matter should be resolved in a manner satisfactory to the NRC staff.

See Section 5.5.3 of this supplement regarding the status of this matter.

(8) The Committee stated that they support the NRC staff program for evaluation of fire protection in accordance with Appendix A to Auxiliary and Power Conversion Systems Branch Technical Position 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants." The Committee recommended that the NRC staff give high priority to the completion of both owner and staff evaluations and to recommendations for Davis Besse, Unit 1, in order to maximize the opportunity for improving fire protection while areas are still accessible and changes are more feasible.

See Section 9.6.1 of this supplement regarding the status of this matter.

(9) The Committee stated that the applicant and the NRC staff should further review security provisions for Davis Besse, Unit 1, for measures that could significantly reduce the possibility and consequences of sabotage, and that such measures should be implemented where practical.

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We have further reviewed the applicant's revised Industrial Security Plan for Davis Besse, Unit 1 and find that the applicant's revised Industrial Security Plan is acceptable (see Section 13.6 of this report.)

In addition, we will require submittal of an amended physical security plan in compliance with the requirements of 10 CFR Part 73.55 (effective February 24, 1977). The applicant's submittal for an amended industrial security plan pursuant to 10 CFR Part 73.55 must be submitted by May 25, 1977.

(10) The Committee identified generic items of concern which they considered relevant to the Davis Besse, Unit 1 plant, and indicated that the generic concerns should be dealt with by the staff and the applicant as solutions are found.

Appendix D of this report notes the disposition and status of each of the indicated items.

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20.0 FINANCIAL QUALIFICATIONS

20.1 Introduction

The Commission's regulations relating to the determination of an applicant's financial qualifications for a facility operating license appear in Section 50.33(f) and Appendix C to 10 CFR Part 50. In accordance with these regulations, the Toledo Edison Company and the Cleveland Electric Illuminating Company have supplied operating and shutdown costs estimates for the Davis Besse Nuclear Power Station, Unit No. 1, as well as providing additional financial information. The following analysis summarizes our review of the financial information and addresses the financial qualifications of the Toledo Edison Company and Cleveland Electric Illuminating Company to operate and, if necessary, to permanently shut down and safely maintain the subject facility. The Toledo Edison Company and the Cleveland Electric Illuminating Company supply electricity to approximately 2.9 million customers over a 4,200 square mile service area in northeastern and northwestern Ohio. Recent financial information for each of the applicants, for the 12 months ended December 21, 1976, is presented in Table 20.1.

Table 20.1

Financial Data for the Toledo Edison Company and Cleveland Electric Illuminating Company (12 months ended December 31, 1976)

	Toledo Edison Company	Cleveland Electric Illuminating Company
Operating Revenues (millions)	$224	$ 523
Net Income (millions)	$ 39	$ 82
Total Capitalization (millions)	$780	$ 1488
Bond Rating		
(Moody's/Standard & Poor's)	Baa/A	Aa/AA

Toledo Edison Company and the Cleveland Electric Illuminating Company will share in the output of the Davis Besse 1 facility in the same proportion as its ownership percentage: Toledo Edison Company - 48.62 percent; Cleveland Electric Illuminating Company - 51.38 percent. These percentages reflect a transfer of 3.88 percent ownership interest from the Toledo Edison Company to the Cleveland Electric Illuminating Company, which has been completed and for which payment has been made.

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20.2 Estimated Operating and Shutdown Costs

For the purpose of estimating the unit's annual operating costs, the Toledo Edison Company and the Cleveland Electric Illuminating Company assumed July 1977 as the startup date for commercial operation of the facility. The estimate of the Toledo Edison Company and the Cleveland Electric Illuminating Company for the total annual cost of operating the unit during each of the first five years of operation is presented in Table 20.2. The unit costs (mills per kilowatt-hour) are based on a net electrical capacity of 906 megawatts electrical. The five year average costs were calculated by annualizing the estimated costs for 1977 in combination with the annual estimates for 1978 through 1981.

Table 20.2

Operating Cost Estimate
(First Five Years of Commercial Operation)

	Plant Capacity	Operating Cost Estimate (thousands)	Mills/Kilowatt-hour
(July-Dec.) 1977	60%	$ 68,473	28.8
1978	70%	$ 168,950	30.4
1979	62%	$ 164,940	33.5
1980	73%	$ 163,973	28.3
1981	70%	$ 163,952	29.5
5-year average	67%	$ 159,752	30.0

In estimating the costs of permanently shutting down the facility, the Toledo Edison Company and the Cleveland Electric Illuminating Company assumed that the plant would be entombed and no longer used as a commercial nuclear power facility. Expenditures for entombment are projected to be $10 million initially, with an annual surveillance expense of $90,000 thereafter. Entombment consists of sealing all remaining highly radioactive components within a biologically secure structure after having removed all fuel assemblies and radioactive fluids and waste.

20.3 Source of Funds

The Toledo Edison Company and the Cleveland Electric Illuminating Company expect to cover all operating expenses, including taxes, and interest payments through revenues generated from their system-wide sales of electricity. The applicants have consistently exhibited the ability to cover all operating expenses as evidenced by the ratio of operating revenue to operating and interest expenses as shown in Table 20.3. The staff assumes that shutdown and subsequent maintenance costs will either be expensed in the year incurred or amortized over a period of years, depending on the rate-making policy of the regulatory authorities.

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Table 20.3

Ratio of Operating Revenue to Operating and Interest Expenses

Year	Toledo Edison Company	Cleveland Electric Illuminating Company
1976	1.08	1.08
(12 Months ended June 30, 1976)		
1975	1.08	1.08
1974	1.06	1.10
1973	1.11	1.13
1972	1.14	1.16
1967 - 1971 (Average)	1.18	1.17

During 1976, the Toledo Edison Company and the Cleveland Electric Illuminating Company sold electricity for average unit prices (mills per kilowatt-hour) of 30.3 and 29.3, respectively. These unit prices experienced by the companies are above the 1977 estimated unit cost (including a 10 percent return on investment) of generating electricity from the Davis Besse 1 facility.

20.4 Conclusion

In accordance with the regulations cited above, there must be reasonable assurance that the applicant can obtain the necessary funds to cover the estimated costs of the activities contemplated under the license. Based on our analysis, we have concluded that Toledo Edison Company and Cleveland Electric Illuminating Company satisfy this reasonable assurance standard and, therefore, are financially qualified to operate and, if necessary, shut down and safely maintain the Davis Besse Nuclear Power Station, Unit No. 1. Our conclusion is supported by the following factors as discussed above: (1) the applicants' ability to earn revenues sufficient to cover all operating expenses, including taxes, and interest payments; and (2) the projected output of lower unit cost electricity from this facility, as compared with the utilities' present average price of electricity.

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22.0 CONCLUSIONS

Based on our evaluation of the application as set forth in our Safety Evaluation Report issued on December 9, 1976 and our evaluation as set forth in this supplement, we conclude that the operating license can be issued to allow power operation at full rated power (2772 megawatts-thermal) subject to license conditions noted in the Safety Evaluation Report and this report which will require further Commission approval and license amendments before the stated condition can be removed.

We conclude that the construction of the facility has been completed in accordance with the requirements of Section 50.57(a)(1) of 10 CFR Part 50, and that construction of the facility has been monitored in accordance with the inspection program of the Commission's staff.

Subsequent to the issuance of the operating license for full rated power for the Davis Besse, Unit 1, the facility may then be operated only in accordance with the Commission's regulations and the conditions of the operating license under the continuing surveillance of the Commission's staff.

We conclude that the activities authorized by the license can be conducted without endangering the health and safety of the public, and we reaffirm our conclusions as stated in our Safety Evaluation Report and this supplement.

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APPENDIX A

CONTINUATION OF CHRONOLOGY OF RADIOLOGICAL REVIEW

November 26, 1976	Staff letter regarding security clearance for review of classified Sandia Safeguards Reports.
November 29, 1976	Applicant letter on separation criteria for safety related redundant electrical cable conduits.
December 3, 1976	Applicant letter regarding separation criteria for electrical wire and conduits and response to staff request for amendment to Final Safety Analysis Report.
December 3, 1976	Applicant letter advising that Amendment 40 (Revision 24) to Final Safety Analysis Report to be submitted December 6, 1976.
December 9, 1976	Safety Evaluation Report was issued this date.
December 9, 1976	Staff letter to Advisory Committee on Reactor Safeguards transmitting outstanding issues and Safety Evaluation Report.
December 9, 1976	Applicant letter advising Amendment 40 will include the appropriate electrical diagram changes and modifications of feedwater system.
December 9, 1976	Staff letter to applicant transmitting Safety Evaluation Report and Federal Register Notice.
December 9, 1976	Amendment No. 40 (Revision 24) docketed.
December 13, 1976	Applicant letter providing verification for compliance with Appendix G-Pressure Temperature Limits during startup and shutdown.
December 16, 1976	Applicant letter providing final report on the ECCS emergency sump line testing.
December 15, 1976	Staff letter providing position on Reactor-Coolant Flow Transmitters.
December 20, 1976	Staff letter regarding compliance with 10CFR Part 50, Appendix K (Davis Besse, ECCS Reevaluation).

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December 20, 1976	Amendment No. 41 (Revision No. 1 to General Information portion of application).
December 20, 1976	Applicant letter transmitting Reactor Protection System Noise Testing Data.
December 22, 1976	Applicant letter transmitting Containment Vessel Integrated Leak Rate Tests.
December 27, 1976	Applicant letter regarding degraded grid voltage condition.
December 30, 1976	Amendment No. 42 (Revision 25) docketed.
January 5, 1977	Applicant Letter regarding Babcock and Wilcox Company's ECCS evaluation mode.
January 5, 1977	Applicant letter requesting additional 45 days to address staff position on Reactor Coolant Flow Monitoring Design modifications.
January 10, 1977	Staff letter advising temporary badging variance requested is acceptable regarding Industrial Security Plan.
January 12, 1977	Amendment No. 43 (Revision 26) docketed.
January 12, 1977	Staff meeting with applicant to discuss overpressure protection.
January 13, 1977	Staff meeting with applicant to discuss plant technical specifications.
January 14, 1977	Letter from Advisory Committee on Reactor Safeguards to Chairman Rowden-Report on Davis Besse Nuclear Power Station, Unit 1.
January 14, 1977	Staff meeting with applicant to discuss outstanding open items.
January 14, 1977	Staff letter transmitting Amendment No. 3 to Provisional Construction Permit No. CPPR-80; Federal Register Notice and Initial Decision for Antitrust.
January 14, 1977	Staff letter providing Sample Technical Specifications and Errata Sheet concerning Fire Protection Evaluation.
January 21, 1977	Staff letter on "Effects of Fuel Rod Bowing on Departure From Nucleate Boiling."
January 21, 1977	Staff letter transmitting Advisory Committee on Reactor Safeguards for Davis Besse Nuclear Power Station, Unit 1.

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January 24 1977	Staff letter requesting additional Financial Information.
January 24, 1977	Staff meeting with applicant to discuss ECCS-Appendix K.
January 25, 1977	Staff letter requesting additional information on Industrial Security Plan.
January 25, 1977	Staff letter providing Standard Technical Specifications for applicant review.
January 25, 1977	Staff letter requesting additional information for Degraded Grid Voltage Condition.
January 27, 1977	Applicant letter documenting removal of Automatic Test Inserter from Safety Features Actuation System.
January 27, 1977	Applicant letter providing information for Reactor Coolant System Leakage Detection System.
January 27, 1977	Staff meeting with applicant to discuss input error in B&W ECCS evaluation.
January 27, 1977	Applicant letter providing clarification on the impact of electrical separation criteria.
January 27, 1977	Applicant letter providing clarification on the impact of electrical separation criteria for cable tray fill exceeding tray side rails.
January 27, 1977	Applicant letter summarizing non IE to IE isolation qualifications of the reactor protection system and the safety features actuation system.
February 1, 1977	Applicant letter regarding staff request for amending application for license regarding ownership percentages.
February 2, 1977	Amendment No. 44 (Revision No. 2 to General Information portion of application).
February 3, 1977	Applicant letter providing response to request for additional information regarding industrial security plan.
February 3, 1977	Applicant letter providing response to request for additional financial information.
February 3, 1977	Applicant letter providing information on Natural Circulation Test.

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February 7, 1977	Applicant letter providing information for electrical separation criteria.
February 8, 1977	Applicant letter transmitting reports entitled, "Fuel Rod Bow Effects on Davis Besse 1 - Technical Specifications," and "Davis Besse 1 LOCA Analysis - Design Pressure Drop Evaluation."
February 9, 1977	Applicant letter providing additional information for offsite power systems.
February 11, 1977	Applicant submittal of Fire Hazard Analysis Report.
February 11, 1977	Staff meeting with applicant to discuss ECCS-Appendix K and over-pressurization protection.
February 14, 1977	Applicant letter with comments on proposed technical specifications.
February 17, 1977	Staff meeting with applicant to discuss decay hear removal system.
February 18, 1977	Staff meeting with applicant to discuss electrical qualification of reactor protection system and engineered safety features system.
February 22, 1977	Applicant submittal consisting of revised pages to Revision 5 of the industrial security plan.
February 22, 1977	Applicant letter regarding frequency of performing check presently required technical specifications.
February 23, 1977	Applicant letter requesting relief from the preoperational test program status.
February 25, 1977	Staff letter for guidance on implementing the Part 73.55 physical security plan.
February 28, 1977	Applicant letter requesting meeting to discuss reactor coolant flow monitoring.
February 28, 1977	Applicant letter requesting extension for construction permit from April 1, 1977 to November 1, 1977.
February 28, 1977	Applicant letter summarizing selection of 0.15g SSE at bedrock design.
March 2, 1977	Staff meeting with applicant to discuss ECCS-Appendix K.
March 4, 1977	Applicant letter providing additional information regarding containment vessel isolation systems, test, and inspections.

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March 4, 1977	Applicant letter providing additional information regarding reactor vessel surveillance specimen program.
March 4, 1977	Applicant letter providing information on status of process radiation monitors for preoperation testing.
March 4, 1977	Applicant letter providing information on operability of containment vessel pressure transmitters.
March 4, 1977	Applicant letter regarding relief request on the preoperational test program.
March 10, 1977	Applicant letter providing additional information on off-site power systems.
March 17, 1977	Staff meeting with applicant to discuss test procedures for verification that faults on non-class IE circuits would not propagate to class I-E circuits on the reactor protection system.
March 17, 1977	Staff letters regarding the routing of class IE wiring with non-class IE wiring within class IE logic cabinets and panels.
March 23, 1977	Staff letter to applicant transmitting ACRS letters dated January 31, 1977 and February 17, 1977.
March 23, 1977	Applicant letter documenting basis for assurance that the manual bypass valves DH21 and DH23 will not open inadvertently during power operation.
March 28, 1977	Applicant letter providing Revision 6 to industrial security plan.
March 28, 1977	Staff letter referencing forthcoming meeting on reactor coolant system flow.
March 29, 1977	Staff letter regarding fission gas releases from fuel pellets with high burnup.
March 31, 1977	Applicant letter transmitting final report on conduit separation test program.
March 31, 1977	Staff meeting with applicant to discuss separation criteria for safety related electrical conduits.
April 1, 1977	Applicant letter indicating vendor will not be able to supply qualified flow transmitters until July 1977.
April 6, 1977	Applicant letter advising FSAR and technical specifications revised to allow radiation setpoint to be established after testing.

APPENDIX B

BIBLIOGRAPHY FOR
THE DAVIS BESSE NUCLEAR POWER STATION
UNIT 1
SAFETY EVALUATION REPORT

The following additional references are provided:

Geology, Seismology, and Foundation Engineering

70. Trifunac, M. D. and A. G. Brady, "On the Correlation Se[illegible]ic Intensity Scales with Peaks of Recorded Strong Ground Motion," Bulletin of the Seismolo[illegible]al Society of America, Volume 65, pages 139-162, 1975.

Review by Advisory Committee on Reactor Safeguards

71. Letter, B. Rusche to M. Bender, dated January 31, 1977, for interpretation of Committee's comments on Davis Besse Unit 1.

72. Letter, M. Bender to B. Rusche, dated February 17, 1977, providing clarification of ACRS report on Davis Besse, Unit 1.

Generic Letters

73. Letter, D. Moeller to M. Rowden, dated April 16, 1976, regarding the status of generic items relating to light -water reactors, Report No. 4.

74. Letter, B. Rusche to M. Bender, dated January 31, 1977, regarding present status of generic items relating to light-water reactors, Report No. 4.

75. Letter, M. Bender to M. Rowden, dated February 24, 1977, regarding the status of generic items relating to light-water reactors, Report No. 5.

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APPENDIX C

ERRATA TO THE SAFETY EVALUATION REPORT FOR THE DAVIS BESSE NUCLEAR POWER STATION, UNIT 1

PAGE	LINE(S)	
1-2	29	Change "structure" to "vessel"
1-2	33	Change "structure" to "vessel"
1-2	34	Change "structure" to "vessel"
1-2	35	Change "structure" to "vessel"
1-4	8	Change end of sentence to read "supply all the essential water used by the facility"
1-4	27	Change "from" to "to"
1-4	29	Change line to read "automatically returned to the intake forebay which serves as an ultimate heat"
1-5	3	Delete "reactor building" and add "containment"
1-5	4	Delete "reactor building" and add "containment"
1-5	30 & 31	Delete "and the engineered safety features actuation system"
1-5	34	Delete "and procure"
1-7	20	Delete "moderate and"
1-8	9	Change "tank to" to "tank makeup to"
1-8	last line	Change line to read "(3) Reactor Coolant System Leakage Detection System (Section 5.2.4.)"
1-9	24	Change "(Section 7.4.2)" to read "(Section 7.9.2)"

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PAGE	LINE(S)	
2-9	9	Change "building" to "vessel"
2-12	20	Change sentence to read, "The maximum calculated wind tide was 9.3 feet due to a probable maximum meteorological event based on a procedure by Platzman (see Reference 12, Appendix B to this report)."
3-6	1	Change "containment" to "shield"
3-7	11	Delete "and resulting reactor forces"
3-7	23	Change "cannot" to "will not"
3-8	13	Delete "and moderate"
3-8	18 & 19	Delete "and moderate"
3-9	22	Change "relief" to "vent"
3-10	25	Change "containment" to "shield"
3-10	33 & 34	Delete "except for modes with closely spaced frequencies where responses were combined by the absolute sum method."
3-10	last line	Delete "Revision 1"
3-11	26	Add "vessel" after the word "containment"
3-11	31	Change "structure" to "vessel"
3-11	32	Change "structure" to "vessel"
3-12	2	Change "structure" to "vessel"
3-12	4	Change "containment" to "vessel"
3-12	9	Add "vessel" after the word "containment"
4-13	27	Add "shield" after the word "support"
5-1	25	Delete "mounted on" and add "connected to"
5-3	5 & 6	Change "located on" to "connected to"

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PAGE	LINE(S)	
5-3	8	Change line to read, "system safety valves, in conjunction with 18 code safety valves and 2 atmospheric vent valves, and"
5-3	14	Change "six" to "six and one-half"
5-3	21	Change "sodium hydroxide" to "trisodium phosphate"
5-6	2	Change line to read, "the systems are the containment vessel sump level and radiogas and air particulate radio."
5-8	25	Change line to read, "recommendations of Regulatory Guide 1.83, Revision 1, Inservice Inspection"
6-3	9	Change line to read "low level, an automatic switchover from injection to recirculation is initiated."
6-7	last line	Delete "reactor" and add "auxiliary"
6-8	8, 9 & 10	Change sentence to read, "This changeover of pump suction is accomplished automatically with manual backup from the control room."
7-11	9	Change "Table 7.9" to "Table 7.8"
7-11	35 & 36	Delete the sentence, "Except for the equipment referenced in Topical Report BAW-10003 which is discussed in Section 7.2 of this report."
7-11	36	Change "subject" to Subject"
7-13	28	Delete the word "solid"
8-1	10	Change "1071" to "1971"
8-2	8	Change line to read, "provided to prevent automatic paralleling both sources through an essential bus."
8-3	4	Change "vital" to "essential"
8-3	last line	Change "vital" to "essential"
8-4	20	Change "The batteries" to "The paired batteries"

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PAGE	LINE(S)	
9-1	10	Change "air-conditioning" to "emergency ventilation"
9-3	8 & 9	Change "seismic Category Safety Class 3" to "seismic Category I"
9-3	30	Delete "emergency diesel generators"
9-4	3 & 4	Change sentence to read, "The structure houses three 100 percent capacity service water pumps, two cooling tower makeup pumps, and a diesel driven fire pump."
9-4	12	Delete "heated main condenser circulating" and add "service"
9-4	23 & 24	Change sentence to read, "Under accident conditions both trains will be aligned to supply component cooling water only to the essential components, including the emergency diesel generators."
9-4	26	Change "the pump" to "the three pump"
9-4	27	Change "the component" to "the three component"
9-9	26 through 32	Delete lines 26, 27, 28, 29, 30 31, and 32
9-10	32	Change "200" to "300"
9-10	33	Change "3100" to "3000"
9-12	29	Delete "auxiliary building"
9-13	7	Change "exhaust" to "ventilation"
9-16	3	Change line to read, "elevation 585 feet International Great Lake Datum do not serve the diesel generators."
10-1	11	Change "dump" to "vent"
10-1	23	Change "to 2.7" to "at 2.7"
10-2	10	Change "hydraulically" to "pneumatically"
10-2	33	Change "building" to "vessel"
10-2	34 & 35	Change sentence to read, "Non-return valves downstream of the isolation valves prevent reverse flow."

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PAGE	LINE(S)	
10-2	35, 36, & 37	Delete "These valves close automatically upon closure of the main steam isolation valves. They can also be remote manually operated from the main control room."
11-1	20 & 21	Delete "from the main steam condenser air ejector"
11-9	11, 12 & 13	Change sentence to read, "Radioactive solid wastes resulting from operation of the plant will include wet solid waste concentrates from the radwaste evaporators, spent resins and spent filter cartridges; and contaminated dry solid waste such as disposable filters, clothing, equipment, and tools."
11-9	24	Change "high level wastes" to "wet solid wastes"
11-9	27	Change "high level" to "wet solid wastes" and change "low level" to "dry solid wastes"
12-3	8	Change "building" to "vessel"
12-3	33	Change line to read, "exposures will be controlled by use of supplied air masks and apparel such as plastic suits."
12-4	4	Change line to read, "The health program and responsibilities will be carried out by the Chemistry and Health Physics"
13-1	14	Change "Inspection" to read "Reliability"
13-4	28 & 29	Change sentence to read, "The services of a physician are available, as required."
15-5	11	Change "Table 15.2" to "Table 15.7"
15-6	47, 48, 49, & 50	Add, Estimated Consequences

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15-8	9, 10 & 11	Change "Building" to "Vessel"
-9	7	Change "<u>DOSES</u>" to "<u>ACCIDENT</u>"
15-9	25	Add "vessel" after the word "containment"

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PAGE	LINE(S)	
17-1	15	Change "res ponsible" to "responsible"
17-2	Figure 7.1	Show Power Engineering and Power Plant Construction Staffs reporting to the Project Engineer and Power Plant Construction Superintendent, respectively, by vertical lines.
17-5	34	Change "control" to "Company"
A-1	24	Chage "Atom" to "Atomic"
A-10	9	Change "34" to "39"

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APPENDIX D

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS-GENERIC MATTERS

The Advisory Committee on Reactor Safeguards (Committee) periodically issues a report listing various generic items applicable to large light-water reactors. These are items which we and the Committee, while finding present plant designs acceptable, believe have the potential of adding to the overall safety margin of nuclear power plants, and as such should be considered for application to the extent reasonable and practicable as solutions are found, recognizing that such solutions may occur after completion of the plant. This is consistent with our continuing efforts toward reducing still further the already small risk to the public health and safety from nuclear power plants. The Committee report concerning these generic items on which this Appendix is based was issued to the Commission on April 16, 1976 in a letter from Committee Chairman D. Moeller to Commission Chairman M. Rowden.

The status of staff efforts leading to resolution of all these generic matters is contained in our Status Report on Generic Items periodically transmitted to the Committee. The latest such Status Report is contained in a letter from B. Rusche to M. Bender dated January 31, 1977.

The Committee in its report on Davis Besse, Unit 1 dated January 14, 1977, identified the relevant generic items from the April 16, 1976 letter it considered applicable to Davis Besse, Unit 1. For many of the items so identified, we have provided in the Safety Evaluation Report specific discussions applicable to the Davis Besse, Unit 1 generic status as given in the January 31, 1977 Status Report.

These items are listed below with the appropriate section numbers of the Safety Evaluation Report and/or this supplement where such discussions are to be found. The numbering corresponds to that in the April 16, 1976 report of the Committee.

For those items applicable to Davis Besse, Unit 1 which have not yet progressed to where specific action can be initiated relevant to individual plants, our Status Report on Generic Items referred to above provides the appropriate information.

Group II

1. Turbine Missiles - Section 3.5.1 759083

2. Effective Operation of Containment Sprays in a LOCA - Sections 6.2.2 and 15.3.1

3. Possible Failure of Pressure Vessel Post-LOCA by Thermal Shock-Status Report

4. Instruments to Detect Fuel Failures - Section 5.2.4

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6. Common Mode Failures - Section 7.2.2

7. Behavior of Reactor Fuel Under Abnormal Conditions - Status Report

9. Advisability of Seismic Scram - Status Report

11. Instrumentation to Follow the Course of an Accident - Section 7.5, Section 18, Item 5, and Status Report

Group II.A

1. Pressure in Containment Following a LOCA - Section 6.2.1

4. Rupture of High Pressure Lines Outside Containment - Section 3.6.2

5. PWR Pump Overspeed During a LOCA - Section 5.5.1

7. Steam Generator Tube Failures - Section 5.5.2

8. Periodic Comprehensive 10-year Review of Operating Power Reactors - Status Report

Group II.C

1. Locking Out of ECCS Power-Operated Valves - Sections 5.5.3, 6.3.3.2, 7.3.4, and 7.3.5

2. Fire Protection - Section 9.6.1 and Section 18, Item 8

3. Design Features to Control Sabotage - Section 13.6 and Section 18, Item 9

4. Decontamination and Decommissioning of Reactors - Status Report

5. Reactor Vessel Supports (Asymmetric LOCA Loads from Sudden Subcooled Blowdown) - Section 3.9.3

6. Water Hammer - Section 6.3.2. In addition, the principal area of concern in this item has been the feed inlet to the steam generators. This has not been a problem in operating Babcock & Wilcox plants because of system and component design and is not expected to be a concern in Davis Besse, Unit 1.

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

January 14, 1977

Honorable Marcus A. Rowden
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: REPORT ON DAVIS-BESSE NUCLEAR POWER STATION, UNIT 1

Dear Mr. Rowden:

At its 201st meeting, January 6-8, 1977, the Advisory Committee on Reactor Safeguards completed its review of the application by the Toledo Edison Company and the Cleveland Electric Illuminating Company for a license to operate the Davis-Besse Nuclear Power Station, Unit 1. Members of the Committee visited the plant on May 18, 1976, and a subcommittee meeting was held in Washington, D.C. on December 21, 1976. During its review, the Committee had the benefit of discussions with representatives and consultants of the Applicant, the Babcock and Wilcox Company, the Bechtel Corporation, and the NRC Staff. The Committee also had the benefit of the documents listed. The Committee reported on the application for a construction permit for this unit on August 20, 1970.

The Davis-Besse Nuclear Power Station, Unit 1, is located on the southwestern shore of Lake Erie about midway between the cities of Toledo and Sandusky, Ohio. The minimum exclusion distance is 2400 ft. The low population zone, with a radius of two miles, included about 870 people in the 1970 census. The nearest population centers are Toledo (1970 population 383,818) and Sandusky (1970 population 32,674), both about 20 miles from the plant.

The nuclear steam supply system employs a Babcock and Wilcox pressurized water reactor similar in most respects to those first used in the Oconee Nuclear Station. This system differs from the Oconee units and several other similar units in that the steam generator loops are raised about 30 ft above the level in the original plant arrangement. Although this change was made to eliminate the need for internal vent valves, four such valves are provided because of their beneficial effect in reducing steam binding following a postulated loss-of-coolant accident.

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The proposed power level for the unit is 2772 MWt, as compared to 2633 MWt proposed at the construction permit stage. This higher power level is the same as that proposed for the Rancho Seco and Three Mile Island, Unit 2 reactors, both of which have been reviewed by the NRC Staff and the Committee and found acceptable.

The structures and components of Davis-Besse, Unit 1, were designed for a Safe Shutdown Earthquake (SSE) acceleration of 0.15g at the foundation level. Because of changes in the regulatory approach to selection of seismic design bases, the Committee believes that an acceleration of 0.20g would be more appropriate for the SSE acceleration at a site such as this in the Central Stable Region. The Applicant presented the results of preliminary calculations concerning the safety margins of the plant for an SSE acceleration of 0.20g. The Committee recommends that the NRC Staff review this aspect of the design in detail and assure itself that significant margins exist in all systems required to accomplish safe shutdown of the reactor and continued shutdown heat removal, in the event of an SSE at this higher level. The Committee believes that such an evaluation need not delay the start of operation of Davis-Besse, Unit 1. The Committee wishes to be kept informed.

The performance of the Emergency Core Cooling System (ECCS) has been evaluated using a Babcock and Wilcox evaluation model applicable to the raised-loop configuration. The NRC Staff has reviewed these evaluations and has determined that certain assumptions regarding return to nucleate boiling do not comply strictly with the provisions of Appendix K to 10 CFR Part 50. The NRC Staff is also reviewing several other areas relating to ECCS performance. These matters should be resolved in a manner satisfactory to the NRC Staff.

In conjunction with the evaluation and assessment of the impact of routine waste releases from this plant, the Committee recommends that the NRC Staff provide leadership in encouraging the development of improved environmental radiation surveillance capabilities on the part of the State of Ohio and appropriate local regulatory agencies.

The Committee notes that post-accident operation of the plant to maintain safe shutdown conditions may be dependent on instrumentation and electrical equipment within containment which is susceptible to ingress of steam or water if the hermetic seals are either initially

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defective or should become defective as a result of damage or aging. The Committee believes that appropriate test and maintenance procedures should be developed to assure continuous long-term seal capability.

The Committee recommends that, prior to commercial power operation of Davis-Besse, Unit 1, additional means for evaluating the cause and likely course of various accidents, including those of very low probability, should be in hand in order to provide improved bases for timely decisions concerning possible off-site emergency measures. The Committee wishes to be kept informed.

The question of whether the design of this plant must be modified in order to comply with the requirements of WASH-1270, "Technical Report on Anticipated Transients Without Scram (ATWS) for Water-Cooled Reactors," remains an outstanding issue pending the NRC Staff completion of its review of the Babcock and Wilcox generic analyses of ATWS. The Committee recommends that the NRC Staff, the Applicant, and the Babcock and Wilcox Company continue to strive for an early resolution of this matter in a manner acceptable to the NRC Staff. The Committee wishes to be kept informed.

Davis-Besse, Unit 1, has installed a bypass loop containing two manually operated valves around the decay heat removal system suction line isolation valves. The normally closed bypass valves would be opened in the event of a spurious closure of one of the decay heat removal system suction line isolation valves during system operation. The Committee recommends that further attention be given to the means employed for isolation of the low pressure residual heat removal system from the primary system while the latter is pressurized, and that reliable means be developed to assure such isolation. This matter should be resolved in a manner satisfactory to the NRC Staff. The Committee wishes to be kept informed.

The Committee supports the NRC Staff program for evaluation of fire protection in accordance with Appendix A to Auxiliary and Power Conversion Systems Branch Technical Position 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants." The Committee recommends that the NRC Staff give high priority to the completion of both owner and staff evaluations and to recommendations for Davis-Besse, Unit 1, and for other plants nearing completion of construction in order to maximize the opportunity for improving fire protection while areas are still accessible and changes are more feasible.

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The Committee believes that the Applicants and the NRC Staff should further review security provisions for Davis-Besse, Unit 1, for measures that could significantly reduce the possibility and consequences of sabotage, and that such measures should be implemented where practical.

Other generic problems are discussed in the Committee's report, "Status of Generic Items Relating to Light Water Reactors: Report No. 4," dated April 16, 1976 (Attached). Those problems relevant to the Davis-Besse, Unit 1, should be dealt with by the NRC Staff and the Applicant as solutions are found. The relevant items are: II-1, 2, 3, 4, 6, 7, 9, 11; II.A-1, 4, 5, 7, 8; II.C-1, 2, 3, 4, 5, 6.

The Advisory Committee on Reactor Safeguards believes that, if due regard is given to the items mentioned above, and subject to satisfactory completion of construction and pre-operational testing, there is reasonable assurance that the Davis-Besse Nuclear Power Station, Unit 1, can be operated at power levels up to 2772 MWt without undue risk to the health and safety of the public.

Sincerely yours,

M. Bender

M. Bender
Chairman

Attachment:
Status of Generic Items Relating to Light Water Reactors: Report No. 4 dated April 16, 1976

References:

1. Davis-Besse Nuclear Power Station, Unit 1, Final Safety Analysis Report (March 1973) with Revisions 1 through 24.

2. Safety Evaluation Report (NUREG-0136) in the matter of the Davis-Besse Nuclear Power Station, Unit 1 (December 1976).

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