Docket
#50-346

September 5, 1973

To All Concerned:

A copy of a letter sent to the Toledo Edison Company, dated August 28, 1973, subject: Request for Additional Information for Davis-Besse Nuclear Power Station, was sent to you on August 30, 1973. The attached enclosure was inadvertently omitted, please attach. Thank you.

Carolyn Reid

Carolyn Reid
Secretary
PWR-4, L

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ENCLOSURE 1

REQUESTS FOR ADDITIONAL INFORMATION

TOLEDO EDISON COMPANY

DAVIS-BESSE NUCLEAR POWER STATION

DOCKET NO. 50-346

AUGUST 27, 1973

2.0 SITE CHARACTERISTICS

The Site Characteristics requests relate to the following FSAR Sections:

2.3 Meteorology

2.4 Hydrology

2.5 Geology and seismology

Our principal concerns are (1) the methods used to estimate the Probable Maximum surge elevations at the plant site, (2) the method used to estimate the Probable Maximum Flood (PMF) from the Toussaint River, and (3) potentially non-conservative estimates of Probable Maximum Precipitation used to establish the PMF from the Toussaint River and to estimate the localized flooding effects on site drainage systems (including the roofs of safety-related buildings).

2.3 Meteorology

2.3.1 Verify the statement in Section 2.3.1.2.8, first paragraph, page 2-34 of the FSAR, that high air pollution potential (atmospheric stagnation) conditions occur 20 to 30% of the time at the plant site.

2.3.2 Provide a full year of onsite temperature and humidity data with a joint recovery rate of at least 90% as soon as such data become available.

2.3.3 Provide a copy of the study of t' e potential effects of the cooling towers on the environment that is referred to in Section 2.3.2.3 of the FSAR.

2.3.4 Provide evidence that the accuracy of the meteorological measurement equipment to be used in the operational onsite measurement program, especially with respect to the dewpoint measurements, meets the accuracy criteria recommended in Regulatory Guide 1.23.

2.3.5 Provide a table of annual average relative concentration (X/Q) estimates for 16 radial sectors to a distance of 50 miles from the plant.

2.4 Hydrology

2.4.1 Provide the following information:

a. Describe the meteorology, surges, and waves which occurred in the region in the fall of 1972 and the spring of 1973. Document and compare the postulated probable maximum meteorological event with the 1972 and 1973 seiche-causing storms, and discuss whether the probable maximum event is sufficiently conservative in view of the occurrence of these recent events.

b. Verify the probable maximum surge model by reconstituting at least the most severe of the two recent events.

c. Include, in your bases for wave estimates, the components attributable to waves refracted and reflected from and through offshore islands.

d. Discuss and document the potential for wave-induced resonance in the intake canal at both high and low lake levels.

2.4.2 Section 2.6, reference 32, for probable maximum precipitation (PMF), is not considered adequate. Use "Seasonal Variation of the Probable Maximum Precipitation East of the 105th Meridian for Areas from 10 to 1000 square miles and Durations of 6, 12, 24, and 48 Hours," HM #33, U.S. Weather Bureau (now NOAA). Revise your estimates of the effects of the PMP on site drainage (including the roofs of safety-related structures) and the PMF on the Toussaint River. (Also see Request 2.4.9) With respect to the PMP effects on the roofs of safety-related structures, include a discussion of the protection provided for roof penetrations to prevent interior flooding of safety-related systems and components.

2.4.3 Provide a description of all penetrations, including locations, at or below elevation 585 feet, into safety-related buildings. Include discussions of penetrations in "non-safety-related" buildings if a pathway for inleakage to safety-related buildings exists. Describe the methods to be employed for flood-proofing these penetrations.

2.4.4 Provide the data base for the reported historical maximum surge and wave heights; include the date and source of record for the maximum recorded transverse seiche. In addition, provide a revised discussion to include the recent severe surge activity.

2.4.5 Provide the design bases for, and the size of, riprap protection on the breakwater. Identify the extent of rock protection on the intake dikes and breakwater (e.g., lower elevation of riprap blanket, etc.).

2.4.6 Discuss and document the lake stillwater elevation plus wind-generated waves and runup that will necessitate taking the cooling tower out of service. Based on historical surge records, the design basis surge hydrograph (for rate of rise, etc.) wind-generated waves and runup, and the time required to bring the plant to a safe and orderly cold shutdown, establish a Tech Spec which will ensure the plant is in a cold shutdown configuration prior to any potential loss of the use of the towers due to lake flooding.

2.4.7 You have mixed datums throughout Section 2.4. Revise the text to either consistently use one datum throughout, or clearly identify the relation between the datums used.

2.4.8 Provide map(s) of suitable scale which indicate the fetch locations and lengths used in your wind-generated wave and runup calculations. Provide cross sections from low water elevation through the plant area sufficient to allow independent verification of your runup calculations. Show slopes and types of protective cover (rock, grass, etc.) of all protective structures.

2.4.9 The staff does not concur with your estimate of the PMF from the Toussaint River. Specifically, (i), the unit hydrograph shown on Figure 2-28 does not conform to generally accepted unit hydrograph theory, (ii) the rainfall (PMP) should be revised in accordance with HM #33 (see Request 2.4.2) and (iii) the time distribution of rainfall shown in Tables 2-65 and 2-66 is not considered sufficiently conservative. The time distribution of rainfall should be at least as severe as is indicated in EM 1110-2-1411, Standard Project Flood Determinations, Corps of Engineers. In support of your unit hydrograph, provide your working estimates of the following parameters as defined in your reference 36: A, L, L_{CA}, C_t, tp, tr, and t_R. Your description of water level determinations is not clear. Supplement your description, with maps or other means, to document that backwater from restrictions in or blockage of the Toussaint River will not constitute a safety hazard at the plant.

2.4.10 The experienced and projected lake levels in 1972, 1973, and 1974 were, or are expected to be, somewhat higher than normal. Provide the data base used to estimate the maximum variation in water level in Section 2.4.2.2 and compare this estimate with the expected levels for the three years above. Describe the effect, if any, on groundwater elevations at the site. Describe the hydrostatic design bases for all safety related structures in relation to high lake-induced water levels.

2.5 Geology and Seismology

2.5.1 Provide information concerning the computation of Richter magnitudes on the bases of epicentral intensity and felt area. Discuss the applicability of these relationships to the region surrounding the site.

2.5.2 Expand on the statement by Dr. E. Walter that the bifurcation of the Cincinnati Arch has caused a weakness in the rock and that regional strain is relieved in that area. Include a discussion of how this might imply crustal weaknesses associated with the Findlay Arch which could be related to strain release in the vicinity of the site.

2.5.3 For the purpose of modeling earthquake E2, the Findlay Arch is treated as a "tectonic province" in the context of the Seismic and Geologic Siting Criteria. Provide justification for not treating the Findlay Arch as a "tectonic structure."

3.0 DESIGN CRITERIA - STRUCTURES, COMPONENTS EQUIPMENT AND SYSTEMS

The Design Criteria requests are in the following areas by FSAR Section:

3.2 Classification of Structures, Components and Systems

3.3 Wind and Tornado Design Criteria

3.4 Water Level (Flood) Design Criteria

3.5 Missile Protection Criteria

3.6 Criteria for Protection Against Dynamic Effects Associated With a Loss-of-Coolant Accident

3.7 Seismic Design

3.8 Design of Seismic Class I and Class II Structures

3.9 Mechanical Systems and Components

3.10 Seismic Design of Category I Instrumentation and Electrical Equipment

3.11 Environmental Design of Mechanical and Electrical Equipment

3.2 Classification of Structures, Components and Systems

3.2.1 The Borated Water Storage Tank is identified in Table 3-2 as Group B and designed, fabricated and tested to AWWAD100 code. In current applications this component is classified AEC Quality Group B in Regulatory Guide 1.26 and is designed, fabricated and tested to ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Class 2.

Demonstrate that your proposed classification is essentially equivalent to the requirements of Regulatory Guide 1.26 by identifying the nondestructive testing and quality assurance documentation requirements that you will impose and by demonstrating that these requirements will provide a quality level at least equivalent to that associated with Quality Group B of Regulatory Guide 1.26.

3.2.2 In order to establish compliance with Section 50.55a of 10 CFR Part 50 the following information is required in addition to that provided in Table 5-10.

Provide the order or contract dates and where not already listed the component code and code class, for the following components within the reactor coolant pressure boundary: (1) reactor vessel; (2) full length CRD mechanism housing; (3) part length CRD mechanism housing; (4) steam generators, tube side and shell side; (5) reactor coolant pumps; (6) pressurizer; (7) pressurizer safety valves; (8) relief valves; (9) loop isolation valves; (10) other valves; and (11) piping and fittings.

3.2.3 Expand Table 9-1 by including where appropriate the component code class for those pressure vessels, piping, pumps and valves of the auxiliary fluid systems.

3.2.4 Authorization to use the following ASME Code Case is not granted, since the specified requirements are unacceptable to the Commission for application to components within the reactor coolant pressure boundary.

ASME Boiler and Pressure Vessel Code Case.

1358-2 - High Yield Strength Steel for Section III Construction.

3.3 Wind Tornado Design Criteria

3.3.1 The tornado criteria in Section 3.3.2.1 refer only to the shield building and the auxiliary building. Provide the corresponding information for other Category I structures.

3.3.2 The FSAR provides a description of design-basis tornado borne missiles considered for this facility. Expand the spectrum of tornado missiles considered to include the following assuming that the tornado has 300 m.p.h. rotational and wind velocity plus a 60 m.p.h. translational velocity:

(a) 4" x 12" plank x 12 ft long with a density of 50 lbs/ft^3;

(b) Utility pole 13.5" diam. x 35 ft long with a density of 43 lbs/ft^3;

(c) 1" solid steel rod 3 feet long with a density of 490 lbs/ft^3;

(d) 6" schedule 40 pipe, 15 ft long with a density of 490 lbs/ft^3.

(e) 12" schedule 40 pipe, 15 ft long with a density of 490 lbs/ft^3.

(f) 3" schedule 40 pipe, 15 ft long with a density of 490 lbs/ft^3/

Present the following information for each of the above:

(a) the maximum velocity attained.

(b) the required thickness of a reinforced concrete missile barrier to stop the missiles without their penetrating the missile barrier or creating secondary missiles, (assuming an end-on impact, i.e., the minimum impact area).

In developing the above information, assume the missiles do not tumble and are at all times oriented such as to have the maximum value of $\frac{C_d A}{W}$ while in flight. Clearly define the other analytical model and assumptions that you have used. Since the potential destructive forces that can be developed varies with the elevation difference between where the missile originated and its impact area, present the above information for missiles originating at ground level and at increasing elevations in increments of 50 feet up to the highest structural elevation on the site.

3.3.3 With the aid of site plot plans and layout drawings, identify and locate all essential systems and components that are required in order to attain a safe shutdown in the event of a tornado, including all control, sensing, power and cooling lines. Using the missile barrier thicknesses developed in response to Request 3.3.2. above, discuss the adequacy of all tornado missile barriers protecting the essential systems.

3.3.4 Provide the bases for all safety related components or systems required for safe shutdown that will not be protected by missile barriers. Provide a list of these essential components or systems.

3.3.5 Describe the method of venting the auxiliary building in order to keep the differential pressure drop within the 1.5 psi design limit. Provide the criteria and assumptions used and elaborate by means of sketches in sufficient detail to demonstrate the effectiveness of the venting techniques. Relate your response to the time function in reference to the venting technique using as a basis the 3 psi drop in 3 sec.

3.4 Water Level (Flood) Design Criteria

3.4.1 Provide a list of all seismic Category I systems and equipment that are located below elevation 583.5 feet (MSL).

3.5 Missile Protection Criteria

3.5.1 State the structures that are to be protected against missiles. Describe the analytical techniques employed to estimate the damage of the targets due to the missiles.

3.5.2 Assuming and end-on impact of the missiles identified in Tables 3 and 4 of Section 3.5.4 and 5, missiles to be considered under Section 3.3.2.1, and other potential missiles, provide additional information on the ability of the barriers provided in preventing penetration and creation of secondary missiles.

3.5.3 Aside from the missile discussion in Section 3.5 regarding measures taken to protect the containment from both internal and external missiles, it is not clear what other steps have been taken to protect the remaining parts of the plant from missiles. To clarify this point, provide the following:

a. Identify all internal and external missiles capable of being developed at the plant site from rotating equipment and pressurized containers.

b. For each of the above pos missile sources, indicate, with the aid of drawings, its location in relation to seismic Category I structures, safety-related systems and associated components.

c. Indicate the size, weight and maximum kinetic energy contained by the most destructive missile from each of the above sources.

d. For each of the above missiles, discuss the extent of the protective measures taken and design features employed to protect all essential safety-related items required for safe shutdown.

3.6 Criteria for Protection Against Dynamic Effects Associated With A Loss-of-Coolant Accident

3.6.1 Describe the functional difference between a pipe rupture restraint and a pipe whip restraint as stated in 3.6.2.5.11. Describe the configuration and design criteria of a rupture restraint.

3.6.2 (a) Provide a diagram for each of the systems identified in Table 3-5 that is postulated to rupture and for which restraint is necessary.

(b) Indicate the break locations and the location of the restraints and their constrained directions on the diagram, and provide the criteria utilized to select break locations. Part C of Regulatory Guide 1.46 contains an acceptable method of determining locations for postulated pipe breaks inside containment. The same criteria may also be applied outside containment to determine break locations. Justify the use of alternate criteria.

(c) Provide the configuration of representative pipe whip restraints.

(d) For high energy systems outside of containment, indicate the supplemental protection required to protect systems and structures necessary for the safe shutdown of the plant from the environmental effects, including jet impingement, of through wall pipe cracks.

3.6.3 Reconcile the pipe whip restraint design criteria differences which now appear in Sections 3.6.2.5.4, 3.6.2.5.5, 3.6.2.5.12, with respect to design allowable stress and strain. An increase of 10% in the specified minimum yield strength to account for strain rate effects is acceptable, but an increase of 20% is not.

3.6.4 The thrust coefficients and the analysis methods used for obtaining dynamic effects to pipe whip restraints are inadequate and apparently use different design bases inside and outside containment as presented in Section 3.6.2. An acceptable method for pipe whip analysis is provided in Attachment A. Justify the use of alternate procedures and methods.

3.6.5 Provide the basis and a more detailed description concerning the formulation of the jet impingement force acting on an adjacent object from the postulated pipe rupture, including loading distribution on the impinged surface.

3.7 Seismic Design

3.7.1 Referring to Subsection 3.7.1.2, provide the time history accelerograms that were used as basis for design response spectra. (In Table III-4 on Page 2C-52, it is stated that these accelerograms are recommended in Section III.E.6.a and b. These are not available in the FSAR.) Give details of the accelerograms such as source of seismic record, modifications, etc.

3.7.2 Specify the response spectra actually used in the seismic analysis (the "Helena Upper Average Response Spectra" or the "Recommended Response Spectra").

3.7.3 Clarify in Subsection 3.7.2.2 that the criteria for combining modal responses are on the square-root-of-the-sum-of-squares (SRSS) basis. In Subsection 3.7.3.4, modal responses for closely spaced frequencies should be combined by the absolute sum method.

3.7.4 Seismic instrumentation which provides measured data in spectrum form, such as multi-element seismoscopes, should be provided in selected locations. Such instrumentation would enable direct comparison of measured and predicted response spectra.

3.8 Design of Seismic Class I and Class II Structures ("Category" is preferred term

3.8.1 Structures Other Than Containment

a. Specify the theories of soil mechanics and the methods of their application used to compute loads due to backfill around Seismic Category I structures (Para. 3.8.1.4.4).

b. For Seismic Category I structural elements which may be subjected to the effects of high-energy line breaks outside the containment, the criteria presented in the Attachment B should be utilized in checking and evaluating the present design.

Sufficient information should be provided to establish the extent of conformance with these design criteria. Where deviations from these criteria are proposed, justification should be provided to demonstrate that your proposed criteria are equivalent with respect to the applicable safety functions.

c. Since the borated water tank is a Seismic Category I structure which may contain liquid radioactivity, justify:

(a) Its design for 50% of tornado forces,

(b) Lack of protection against missiles.

(c) Acceptability of consequences of tank failure.

Provide a description of physical features of the tank level instrumentation, foundation (Para. 3.8.1.1.4) and paths of potential leakage.

d. With aid of sketches provide a description, structural design criteria, the degree of conservatism obtained and the location with respect to other parts of the plant for the three electrical manhol s (Para. 3.8.1.1.4).

e. Specify the locations where removable slabs, block partiti ns, etc., are utilized and describe the precautions taken to prevent them from becoming missiles during Design Basis Accidents.

3.8.2 Containment Structure

a. Describe the structural criteria used for those areas of structural design of the shield building which are not covered by ACI 307-69 (Para. 3.8.2.2.3).

b. Describe, with aid of a sketch, the support of the polar crane, its connection to the concrete walls and provisions to resist the shears induced by earthquake.

c. Provide a sketch of the reactor vessel support and describe the manner in which horizontal shears and vertical loads are carried to the concrete (Para. 3.8.2.3.4).

d. Provide a statistical evaluation of tests on splicing reinforcing bars using the Cadweld Process and compare the results of the tests with the guidance of the Regulatory Guide 1.10 (Appendix 3B). Justify any differences.

e. Submit a list of computer programs that have been used in structural and seismic analyses to determine stresses and deformations of Seismic Category I structures. Include a brief description of each program and the extent of its application.

f. Describe the design control measures as required by 10 CFR Part 50 Appendix B that have been employed to demonstrate the applicability and validity of the above computer programs by any of the following criteria or procedures (or other equivalent procedures).

(1) The computer program is a recognized program in the public domain, and has had sufficient history of use to justify its applicability and validity without further demonstration. The dated program version that has been used, the software or operating system, and the computer hardware configuration must be specified to be accepted by virtue of its history of use.

(2) The computer program's solutions to a series of test problems, with accepted results, have been demonstrated to be substantially identical to those obtained by a similar, independently written program in the public domain. The test problems should be demonstrated to be similar to ours with the range of applicability for the problems analyzed by the computer program to justify acceptance of the program.

(3) The program's solutions to a series of test problems are substantially identical to those obtained by hand calculations or from accepted experimental test or analytical results published in technical literature. The test problems should be demonstrated to be similar to the problems analyzed to justify acceptance of the program.

g. Provide a summary comparison of the results obtained from each computer program with either the results derived from a similar program in the public domain, on a previously approved computer program or results from the test problems.

3.9 Mechanical Systems and Components

3.9.1 Clarify Table 3-10 with regard to the use of Code Cases for Class 2 and 3 components. The entry "none" appears in Table 3-10 in the Code Case column for several systems, but the balance of the column is blank. The use of individual Code Cases requires specific approval by the Commission in accordance with 10 CFR 50.55a (refer to (a)(2)(ii) and footnote 6 of the regulation). Complete the table indicating the cases used for each system as appropriate.

3.9.2 The design loading combinations and stress limits for the various plant operating conditions for ASME Class 2 and 3 components have not been completed in Section 3.9.2 or Table 3-10. Regulatory Guide 1.48 provides a summary of current acceptable limits. Provide these design criteria for the Davis-Besse Plant. Justify any limit which exceeds that specified in the Guide, and demonstrate the adequacy of the design safety margin selected.

3.9.3 Describe the measures to be taken which will assure that Class 1, 2 and 3 (Seismic Category I) active pumps and valves will operate under plant conditions when their safety function must be relied upon to effect a plant shutdown or to mitigate the consequences of an accident. The material in 5.2.1.16 and 3.9.2.6 is not definitive enough to be acceptable in the area of functional testing.

3.9.4 The description of the design criteria for safety/pressure relief valve stations in Sections 5.2.2 and 3.9.2.8 is not definitive enough for acceptance. Indicate how the method of analysis has included consideration of the reaction force, dynamic effects of the valve opening time, effect of the sequence of valve openings, including simultaneous discharge, to produce the highest maximum instantaneous value of stress.

3.9.5 In Section 3.9.1.3, only the topical report BAW-10051 is referenced to confirm the design adequacy of reactor internals to withstand the flow-induced vibration during normal operating. A concurrent reference to topical reports BAW-10037, 10038 and 10039 should be included for a complete verification of the valid prototype pre-operational vibration testing.

3.9.6 Justify that the seismic disturbances of the reactor internals at the Davis-Besse plant are less severe than the seismic input used in the topical report BAW-10008, Part I, Rev. 1 and BAW-10041. A comparison of response spectra at the component support locations should be provided. In addition, a list of analysis results including maximum stresses or deformations in the reactor internals due to LOCA and SSE loadings as well as their comparison to the allowable values should be provided.

3.10 Seismic Design of Category I Instrumentation and Electrical Equipment

3.10.1 Provide a summary of seismic testing results of those electrical equipments which are not included in the topical report BAW-10003. Information should include the following:

a. Describe briefly the testing facilities, including functional capability.

b. Provide a list of equipment (devices or assemblies) and supporting structures tested.

c. Identify the type of testing input, including intensity level frequency content, number of axis, input duration and time history sketches of the typical input. The validity of such testing input should be demonstrated.

d. Describe the number, type, and location of monitoring sensors on each equipment and document the maximum response recorded.

e. Identify whether devices were tested in operating condition during the testing of assemblies or supporting structures (i.e., panels and racks).

f. Identify whether devices were mounted during the testing of assemblies or supporting structures and demonstrate the validity of any tests conducted without the devices (or suitable substitutes) or with the mounted devices in inoperative condition.

g. Describe frequency finding testing, including sweep rates and amplitude used. Provide a summary of the frequency finding test results.

h. In the event analyses were used for determining the testing input, provide a description of the analytical methods and procedures, including sketches of the mathematical models used.

i. In the event testing was replaced by analyses, provide justification for assuring the proper functioning of the equipment during the DBE event.

j. Document and discuss any malfunctioning occurring during the testing.

3.10.2 Verify that the response of cabinet assemblies at various instrument or device mounting locations due to Safe Shutdown Earthquake disturbances are less than 1g, the device testing input level specified in BAW-10003.

3.10.3 Supplement the information contained in FSAR Section 3.10, Seismic Design of Class I Instrumentation and Electrical Equipment, as follows:

a. Provide a summary listing (tabulation) correlating all safety related electrical equipment, equipment locations, seismic design bases at each location, seismic qualification method used (test and/or analysis) and seismic test and/or analysis results. This should include the 4.16 Kv switchgear; 480 v switchgear, unit substations and motor control centers; 120 v a-c system components; 125 v d-c system components including batteries, battery racks, battery chargers, distribution centers, and panelboards; static inverters; process control equipment; protection and safeguards actuation racks; nuclear instrumentation; electrical penetration assemblies, diesel generators, motor operated valves, etc.

b. Identify the auxiliary equipment (local control panel, lube oil system, etc.) which is required for the operation of the emergency diesel generators and verify that this equipment has been seismically qualified. Describe the testing and/or analysis performed to seismically qualify this equipment.

c. Confirm that the seismic qualification testing demonstrated the capability to change state or operate during a SSE for all components which are required to so operate in performance of their design safety function. Provide the bases for the methods of simulation of the net effect of the design basis seismic event which were used in the qualification tests.

d. Provide a more detailed description of the seismic qualification method (test and/or analysis) used for each Class IE component. This description may be incorporated in the summary listing of (a) above.

e. In Section 3.10.2.2, explain or delete the term "SC 2.5" and identify the IEEE standard referred to by number.

3.11 Environmental Design of Mechanical and Electrical Equipment

3.11.1 Supplement and revise the information contained in Section 3.11 with regard to environmental qualification of safety related electrical equipment to include the following:

a. Provide a concise statement of the limiting DBA environmental conditions within the containment, auxiliary building, intake structure, and out-of-doors. This should include temperature, pressure, humidity, radiation, and chemical environment.

b. State the length of time from occurrence of the DBA that each safety related component located in these environments is required to operate in order to perform its design safety function.

c. Describe the environmental qualification testing performed for each component and identify the applicable test documentation. Provide this test documentation if it has not been previously submitted. Your response should (1) state whether the tests were performed on prototype equipment, (2) contain sufficient detail to permit a direct comparison between the test conditions and the limiting DBA conditions (superimposed on normal aging) for all parameters, and (3) discuss the adequacy of the environmental qualification for each component.

d. If environmental qualification is based (in whole or in part) on analyses or on use of data from tests on other than prototype equipment, describe and justify each instance of the use of these methods and identify the applicable documentation.

e. Discuss the bases and the applicability of Figure 3-18, Containment Vessel Environmental Pressure and Temperature Test Envelope.

f. Describe how the motor operators discussed in Section 3.11.2.2.3 were loaded during qualification testing to simulate actual operating conditons.

g. Justify testing electrical penetrations to only 45 psig as stated in Section 3.11.2.1.6 when the expected pressure ranges from ambient to 60 psig as stated in Section 7.4.1.4.1.

4.0 REACTOR

The Reactor Requests are in the following FSAR Subsections:

4.2 Mechanical Design

4.4 Thermal and Hydraulic Design

4.2 Mechanical Design

4.2.1 Provide the following information:

a. Expand the explanation on how the effects of burnup on fuel conductivity and melting were considered. Explain the term (K_n 0.05) on page 4-59. Provide an additional curve on Figure 4-43 to show the variation of conductivity with Temperature for the density of Davis-Besse fuel.

b. Clarify the statements on pages 4-7, 4-11, and 4-13 that the fuel rod spring spacer provides radial support for the cladding. Specifically, evaluate the potential for spring spacer binding, thereby inhibiting axial growth of the fuel column. Submit upper and lower gas plenum dimensions and type of spring spacer material.

c. Page 4-6 indicates that the fuel rods will be internally pressurized with helium. Quantitative information on the degree and distribution of fuel internal pressure as a function of power level and operating life are not presented. Radial pressure profiles across the core at full power, initially and at the end of fuel element life, are required for a full understanding of the effect of helium pressurization on clad loadings and heat transfer.

d. Page 4-12 indicates that unacceptable wear was observed in a few isolated instances during an experimental investigation of fuel assembly and fuel rod vibration. These anomalies were attributed to pretest spacer grid damage.

Provide a discussion of the damage mechanism for the spacer grid, if known, and evaluate the potential for and consequences of similar pre-loading spacer grid failures in the Davis-Besse core. Include a diagram showing the locations of the observed unacceptable wear. With regard to the overall test program results, provide a comparison of wear observed to the maximum wear which would have been considered acceptable. Extrapolate the test data to the maximum lifetime conditions in the Davis-Besse core.

e. Page 4-13 states that prepressurization of fuel rods has caused rates of collapse to decrease significantly. Submit evidence applicable to Davis-Besse to support this contention.

f. Provide a more detailed description of the planned post-shipment inspection program for fuel components (page 4-16) to reflect the level of quality assurance which has been adopted by Davis-Besse.

4.2.2 Provide the following information:

a. Expand the description of the control rod drive mechanism on page 4-22 to include supporting diagrams.

b. Provide a flow diagram for the Chemical Addition System, showing flow rates and pressures for operating and test conditions. Explain the apparent disparity between the operating margin above crystallization temperature discussion on page 9-91 (+15°F) and the same limiting condition for operation in the proposed station Technical Specifications (+10°F). Expand the description of the Chemical Addition System to include complete operational sequences during all modes of operation, clearly identifying each mode.

c. Subsection 4.3.2 includes by reference a description of the Chemical Addition System (page 4-23). However, no mention is made in this subsection of the Makeup and Purification System even though credit is taken for this system to signify conformance to AEC General Design Criterion 26 (see page 3D-25). Explain this apparent omission. Also, clarify the descriptions of both these systems which indicate that they are not required for emergencies (i.e., shutdown when control rods not available).

d. Explain the term "100% misalignment" at the bottom of page 4-24.

e. The analysis of pressure forces that could eject rods (page 4-25) is incomplete. Provide (or reference) a detailed evaluation of the consequences of this event.

f. Discuss the potential for functional failure of critical components of the reactivity control systems. Assess the sensitivity of the systems to mechanical damage as regards the capability to continuously provide reactivity control.

g. Discuss the effects of potential control rod failures and blowdown loads (e.g., LOCA or control rod ejection) on control rod channel clearances.

h. Expand the design evaluation on page 4-24 to include the remaining critical components in the reactivity control systems. Include any pertinent previous experience and developmental work with similar systems and materials.

i. Include an expanded discussion on page 4-26 of the instrumentation to be employed in connection with mechanical and chemical reactivity control systems and reactivity monitoring in terms of functional requirements.

4.2.3 Identify which of the three prototype control rod drive assemblies discussed in B & W topical BAW-10029 will be utilized for Davis-Besse, and describe any differences from the design described in the topical report.

4.2.4 Provide a list of the materials and their specifications used for safety related components of the control rod drive system. State the extent to which this listing conforms to that shown on page A-9 of BAW-10029. Discuss any differences.

4.2.5 Verify that the materials, and their specifications, used for each component of the reactor internals described in this section and Table 4-4a are as stated on pages E-2 and H-7 of Topical Report BAW-10008, Part 1, Revision 1.

4.2.6 Provide the maximum allowable .2% offset yield strength at room temperature of the cold worked Type 304 ss cladding listed in Tables 4-4c and 4-4d.

4.4 Thermal and Hydraulic Design

4.4.1 Provide reasons for the significant changes in thermal design, hydraulic design, nuclear design, core mechanical design, reactor coolant system design, and ECCS since the PSAR. For example, Section 4.4 was referenced on page 1-26 as providing a description of the changes in hydraulic and thermal design parameters and the reasons for them. No such discussion could be found in Section 4.4.

4.4.2 In June 1973, Babcock & Wilcox filed Proprietary Topical Report BAW-10054, Revision 2, "Fuel Densification Report", outlining the methods to be used to analyze B&W fuel in accordance with the guidelines contained in the Regulatory staff report of November 14, 1972, "Technical Report on Densification of Light Water Reactor Fuels".

Submit the specific results of an evaluation for determining the effects of fuel densification on normal operation, transients, and accidents for Davis-Besse. Address each event in the FSAR and show the consequences of densification on all controlling parameters.

5.0 REACTOR COOLANT SYSTEM

The Reactor Coolant System requests are in the following FSAR Sections:

5.2 Integrity of Reactor Coolant Pressure Boundary (RCPB)

5.3 Thermal-Hydraulic System Design

5.4 Reactor Vessel and Appurtenances

5.5 Component and Subsystem Design

5.2 Integrity of Reactor Coolant Pressure Boundary (RCPB)

5.2.1 Paragraph 5.2.2.1 on page 5-14 indicates that all pressure-relieving devices for the RC system, secondary system, and systems connected with the RC system are shown in Figure 5-3. Expand Figure 5-3 to make it consistent with this statement of a consolidation of all pressure-relieving devices.

5.2.2 Describe the design and installation details for the mounting of the pressure-relieving devices within the reactor coolant pressure boundary. In particular, specify the design basis which will be used to take into account full discharge loads imposed on valves and on connected piping in the event all the valves are required to discharge. Indicate the provisions made to accommodate these loads.

5.2.3 Clarify Table 5-2, "Tabulation of Reactor Coolant System Settings"; specifically, what do the "Code Relief Valves" relate to in comparison to Table 5-1b? None of the settings in Table 5-2 agree with Technical Specification 2.2 for pressurizer electromatic relief valves. Why?

5.2.4 Paragraph 5.2.2.3 references B&W Topical Report BAW-10043 to fulfill the requirement of the ASME Boiler and Pressure Vessel Code to submit an Overpressure Report. Is the plant analyzed in BAW-10043 the same size as Davis-Besse? If not, submit a complete safety/relief valve sizing analysis for Davis-Besse. In addition, expand the discussion in Subsection 5.2.2 to include safety and relief valve numbers, sizes, capacities, setpoints (of each valve, with tolerances), design descriptions, valve design diagrams, and all autoclave test and operating data to support the Davis-Besse safety and relief valve design.

5.2.5 Provide a description of symbols (such as valves) listed on all drawings to facilitate interpretation of information.

5.2.6 Sections 5.2.3.5 through 5.2.3.8 and Figure 5-1 appear to us to be internally inconsistent, and may not be consistent with current AEC fracture toughness regulations. Although the heatup and cool-down limits, given in Figures 3.1-7 and 3.1-8 of the Technical Specifications, appear to be satisfactory, the bases for them and the criticality limits are not clear. An indication should be provided of the degree of compliance with Appendix G of 10 CFR 50 and Appendix G of Section III of the ASME Boiler and Pressure Vessel Code, Summer, 1972, Addenda.

5.2.7 The design criteria for setting stress levels of Class 1 active and non-active valves (differentiate between standard design rated and design by analysis types, as appropriate) and active pumps associated with the loading combinations of the emergency and faulted operating conditions requires more definition than given in Section 5.2.1.7 of the FSAR. The levels of maximum stress specified in your design requirements for these components to cover these combined loads should be given in the FSAR. A summary of the currently accepted limits appears in Sections C2 through C5 of Regulatory Guide 1.48. Provide justification for exceeding any of the limits specified in the Guide, and demonstrate the adequacy of the design margins selected.

5.2.8 (a) Specify which of the three faulted stress limit alternatives listed in Section 5.2.1.5 is being used in the design of Reactor Coolant Pressure Boundary components.

(b) Specify and justify the values selected of ultimate material strength at temperature used in the faulted limit analysis.

Reconcile the statement regarding the non-use of plastic instability methods with elastic system dynamic analysis stated in Section 5.2.1.9, and the plastic instability limits specified in Table 5-12. Indicate which of the alternate limit criteria in the table is used for what specific analysis.

5.3 Thermal-Hydraulic System Design

5.3.1 Provide the basis for the reactor coolant pump operational restriction in Section 5.3.4 to preclude operation below the minimum required NPSH for the pump. Justify the omission of this limiting condition of operation from the Davis-Besse proposed Technical Specifications (Chapter 16.0). Show that the consequences of reduced NPSH (i.e., cavitation and reduced flow) could not contribute toward an event which would affect the health and safety of the public.

5.3.2 Relate the temperature-power operating map discussed in Section 5.3.6 to the core safety limit in Chapter 16.0.

5.4 Reactor Vessel and Appurtenances

5.4.1 For all austenitic stainless steel used for components that are part of

(1) the reactor coolant pressure boundary,

(2) systems required for reactor shutdown,

(3) systems required for emergency core cooling,

(4) reactor vessel internals required for emergency core cooling, and

(5) reactor vessel internals relied on to permit adequate core cooling for any mode of normal operation or under postulated accident conditions,

the following information should be provided, including the degree of conformance with Regulatory Guides 1.31, Control of Austenitic Stainless Steel Welding, and 1.44, Control of the Use of Sensitized Stainless Steel.

a. Cleaning and Contamination Protection Procedures

Describe the procedures that were used and will be used to ensure that the material was and will be suitably cleaned and protected against contaminants capable of causing stress corrosion cracking throughout the fabrication, shipment, storage, construction, testing, and operation of components and systems.

b. Avoidance of Sensitization

Provide a description of materials, processes, inspections, and tests that were used to ensure freedom from the increased susceptibility to intergranular stress corrosion caused by sensitization. This should include the following:

1. If special processing or fabrication methods were used that subjected the material to temperatures between 800 and 1500°F, or that involved slow cooling from temperatures over 1500°F, provide justification that such treatments did not cause increased susceptibility to intergranular stress corrosion.

2. If the presence of delta ferrite was relied on to prevent sensitization of welds or castings, describe the methods that were used to ensure the presence of at least 5% delta ferrite.

c. Welding of Austenitic Stainless Steel

Describe the procedures and requirements that were employed to avoid hot cracking of austenitic stainless steel welds, especially pertaining to filler metal compositions, welding procedure qualifications, and methods for ensuring adequate delta ferrite content of production welds.

5.4.2 Provide an explanation of the difference between the maximum allowable emergency cooldown rate of 100F/hr. and the reactor coolant cooldown rate limit of 235 F/hr.

5.5 Component and Subsystem Design

5.5.1 Main steam line flow restrictors have been adopted in other steam supply systems to limit the discharge rate of steam following a postulated steam line break to acceptable values. Discuss the alternative design features of the Davis-Besse plant which are provided to similarly limit the rate of discharge of steam to acceptable values.

5.5.2 Provide a plot similar to Figure 5A-4 for the calculated overspeed condition which would result in exceeding reactor coolant pump flywheel fracture toughness. Use all assumptions upon which Figure 5A-4 was based (e.g., maximum end-of-life defect = 1.5034" and fracture toughness $\geq$ 85 ksi $\sqrt{in}$).

5.5.3 Provide a description with supporting diagrams of check valve internals and indicate the variety of check valve designs utilized in the Davis-Besse ECCS. Include a discussion of limitations, if any, imposed or modes of ch κ valve failure as applied to the FSAR single failure assumptions and provide an evaluation of the most probable failure mechani .

5.5.4 In light of recent operating experience at a PWR in which an oil spill from a reactor coolant pump resulted in a fire and substantial equipment damage, what steps will be taken to minimize the potential for a similar occurrence at Davis-Besse?

5.5.5 Provide information that steam generator tube fouling, as described in BAW 10027, pgs. A-20-3 or B-26-9 will not lead to deterioration of the steam generator tubing by intergranular stress corrosion or wastage. Provide information that the cleaning procedures (described in Section B-6 of BAW 10027) will not cause attack at the tube - tube sheet crevices, and the precautions taken after cleaning that assure complete rinsing of the cleaning solution from these crevice areas.

5.5.6 Table 5-20: Indicate whether these feedwater quality specifications are for the reactor coolant ake-up water or for the steam generator coolant or both.

6.0 ENGINEERED SAFETY FEATURES

The Engineered Safety Features concerns are in
by FSAR Sections:

6.1 General

6.2 Containment Systems

6.3 Emergency Core Cooling System

The following comments are based on our review:

6.1 General

6.1.1 Provide sufficient information about your proposed ESF inservice inspection program to indicate that the program will be at least as conservative as the program outlined in Regulatory Guide 1.51, "Inservice Inspection of ASME Code Class 2 and 3 Nuclear Power Plant Components," issued May 1973.

6.2 Containment Systems

6.2.1 The FSAR indicates that cold leg, pump suction and pump discharge breaks have not been analyzed, and it is not apparent that the 3 ft^2 hot leg break results in the highest calculated containment pressure;

therefore, provide the results of containment pressure response analyses for a spectrum of break areas for a cold leg (pump suction) pipe and a cold leg (pump discharge) pipe to identify the break size and location that results in the highest containment pressure. Include the following information for each case analyzed: break area, break location, peak containment pressure, time of peak pressure, and energy released to the containment up to the time of peak pressure. For the loss-of-coolant accident at each of the assumed break locations, i.e., the hot leg and cold leg, pump suction and pump discharge pipes, that results in the highest calculated containment pressure, provide a table of mass release rate (lb_m/hr) and enthalpy (Btu/lb_m) as a function of time (hr) throughout the blowdown and core reflood phases of the accidents.

6.2.2 Provide an analysis of the containment pressure response for a spectrum of steam generator, steam line and feedwater pipe ruptures. Specify the postulated break sizes and locations and initial plant conditions. Provide justification for the assumed initial plant conditions. Describe the analytical model used in the analysis. Discuss the conservatism in the analysis with regard to maximizing the energy release to the containment. Provide a table of mass release rate (lb_m/hr) and enthalpy (Btu/lb_m) as a function of time (hr) for the secondary system pipe rupture that results in the highest containment pressure.

6.2.3 The FLASH computer code is used to predict the mass and energy release to the containment during blowdown. Discuss the assumptions made to obtain conservatively high energy release rates from the core for containment evaluation studies. Discuss the criterion used to establish the time to DNB considering that a conservative approach would be to delay DNB until the core was voided by steam.

6.2.4 During blowdown, energy may be transferred from the steam generators to the reactor coolant by conduction through the tube walls. Discuss the heat transfer correlations used for both sides of the steam generator during blowdown. Give additional energy that could be released to the containment if DNB was delayed on the reactor coolant side of the steam generator tubes.

6.2.5 Provide a description of the core reflood model. Discuss the conservatism in the model with respect to maximizing the energy release to the containment. Include the following in your discussion of the core reflood model:

a. Discuss the assumptions made regarding the water remaining in the reactor vessel at the end of blowdown. We believe a conservative approach for containment analyses would be to assume that the water remaining in the reactor vessel is saturated and at the bottom of the core.

b. Discuss the assumptions made regarding the core flooding rate. We believe a conservative approach for containment analyses would be to assume full ECCS operation.

c. Discuss the assumptions made regarding the core quench height and carryout fraction. We believe a conservative approach for containment analyses would be to assume a carryout fraction of 0.8 and that the core would be quenched at the 10-foot level.

d. Provide a tabulation of the system resistances used in the reflood analysis. If these resistances were determined for normal system operating conditions, describe the method used to extrapolate them to reflood conditions.

6.2.6 After the core has been recovered with water following a pump suction break, boiling will occur to cool the core, and a two-phase mixture of steam and water will be generated. Provide an analysis showing the height that the two-phase mixture will rise above the core. If any water is calculated to enter the steam generators, provide the energy release rate to the containment as a function of time.

6.2.7 With respect to the heat sinks listed in Table 6-1 of the FSAR, identify the heat sinks that are exposed to the containment atmosphere on both sides, and specify whether the exposed surface areas represent the surface area of one side or both sides. Also, provide the exposed surface area of the miscellaneous sheathed concrete (item 7 in Table 6-1).

6.2.8 Discuss the method(s) and the accuracy of the method(s) used to determine the free containment volume. Provide a sensitivity study of the effect of the uncertainty in calculating the full volume on the containment vessel pressure response under loss-of-coolant accident conditions. Discuss how the containment full volume will be verified.

6.2.9 For the subcompartment analyses, provide assurance that there are no flow restrictions within a subcompartment that could cause pressure differences. Discuss the difference between the orifice flow area and the miscellaneous flow area that are given for each subcompartment, and how the areas are treated by the computer code COPRA.

6.2.10 The arrangement drawings of the plant indicate that the containment emergency sump is not at the lowest elevation in the plant, and that a significant amount of water could be retained below the elevation at which water would begin to overflow into the emergency sump. The reactor vessel cavity, normal sump, refueling canal, incore instrumentation tunnel, pipe tunnel, and value pit are some of the areas that lie below the emergency sump. Also, Figure 6-17 indicates that the refueling canal drains to the reactor vessel cavity which drains to the reactor vessel cavity which drains to the normal sump, and the emergency sump also drains to the normal sump. Specify the water volume below the sump following a LOCA assuming the containment volume below the elevation of the emergency sump is uniformly filled with water. Discuss the adequacy of available NPSH to the containment spray pumps in the context of Safety Guide 1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps."

6.2.11 The intake screen installed over the containment emergency sump does not appear to be structurally adequate. For example, only a single, completely exposed wire mesh screen is provided, and if the screen was damaged debris might enter both recirculation lines. Provide the following information:

a. a more detailed drawing of the intake screen which shows how the screen is attached to the containment vessel wall and floor,

b. assurance that the failure of a portion of the intake screen will not negate the effectiveness of the entire screen, and

c. assurance that the screen cannot be readily damaged by a missile or large debris that could be carried in the water following a LOCA.

6.2.12 Specify the manufacturer of the containment air cooler units. Describe the qualification test program that was conducted to determine the performance capability of an air cooler unit. Provide a curve of air cooler performance showing energy removal rate as a function of containment atmosphere temperature. Since lake water will be circulated through the air coolers and since the air coolers will be used under both normal and accident conditions, discuss how fouling of the secondary side of the cooling coils was factored into the analysis of the heat removal capability of an air cooler. Specify the service water (lake water) temperature used in the analysis, and provide a table of the maximum and minimum, and monthly average temperature of the lake water at the service water system intake.

6.2.13 Identify the ductwork of the containment air cooling system that must remain intact following a loss-of-coolant accident to assure that the functional capability of the system is not impaired. Discuss the design provisions to assure that the air cooler unit housings and system ductwork can withstand the differential pressures resulting from a loss-of-coolant accident.

6.2.14 Describe how the fusible dropout register(s) associated with the containment air cooling system (as shown on Figure 9-12A) will function.

6.2.15 Provide the following information in Table 6-8, Containment Vessel Isolation Valve Arrangements:

a. the type of valve and valve operator,

b. the valve location with respect to the containment vessel,

c. the method(s) of valve actuation,

d. the valve operator power source,

e. the valve position on motive process failure,

f. the line size, and

g. the FSAR figure on which the isolation valve arrangement is shown.

The penetration numbers listed in Figure 6-12 as spares do not correspond to those listed in Table 6-8 as spares: provide clarification.

6.2.16 The containment vessel penetrations that are exceptions to General Design Criterion 56 are listed on page 6-46 of the FSAR. With respect to items 6 and 7; i.e., the isolation valve arrangements for the containment vessel hydrogen dilution and purge system, and the containment vessel air sample inlet and outlet lines, the rationale for exempting them from the requirements of GDC 56 was not presented. Therefore, discuss why these penetrations are being considered to be exempted from the requirements of GDC 56. Include the containment vessel spray lines in the discussion.

6.2.17 Table 6-8 in the FSAR indicates that the core flooding tank sample and vent lines are each provided with a single isolation valve outside containment. The core flooding tanks are not considered closed systems inside containment and, therefore, General Design Criterion 57 does not apply to these lines. Discuss any other basis that you may have which would demonstrate that the valve arrangement meets the intent of the GDC.

6.2.18 Table 7-5, SFAS Actuation Summary, indicates that the containment valves are grouped into three systems. Provide a tabulation of the isolation valves in each system and specify the trip setpoints.

6.2.19 Describe the qualification test program that was conducted to assure the operability of containment isolation valves, valve drives, position indicators, sensing elements, cables, etc. following a LOCA or steam line break accident. Identify the equipment that was tested. Graphically show the environmental test conditions as a function of time.

6.2.20 Identify all lines penetrating the containment that do not terminate within areas served by the emergency ventilation system. Provide an estimate of the total amount of containment leakage which can bypass the areas served by the emergency ventilation system.

6.2.21 Provide the following information with respect to the plant combustible gas control systems, i.e., the hydrogen dilution system, the hydrogen purge system, and the containment air recirculation system:

a. Provide an analysis of the differential pressures that may occur following a LOCA for the fan housings and ductwork of the containment air recirculation system.

b. On page 3-3 of the FSAR, the hydrogen purge - dilution system is identified as being Seismic Category I. However, the purge line is not Seismic Category I (as indicated on Figure 9-12A), and is subject to a single active failure. Since the proposed method of hydrogen control for the plant involves repressurizing the containment, the purge line should be designed to engineered safety feature standards to assure that continuous hydrogen control capability will exist. Therefore, provide a hydrogen purge system that meets the design criteria for an engineered safety feature.

c. Specify the maximum allowable pressure that the containment will be repressurized to using the hydrogen dilution system before hydrogen purge system operation becomes necessary.

d. Specify the power source for each isolation valve in the hydrogen dilution system (HV 5064, HV 5065, HV 5090, and HV 5091) to assure that the hydrogen dilution system is not subject to a single active failure.

6.2.22 Provide a P and I drawing of the containment gas monitoring system. Discuss the accuracy of the hydrogen analyzer.

6.2.23 Identify any leakage paths which could bypass the volumes treated by the Emergency Ventilation System following a design basis loss-of-coolant accident. Consider isolation valve leakage and leakage through guard pipe welds. Indicate where lines which could be open to containment atmosphere following a LOCA terminate assuming a concurrent seismic event. List the specific leakage paths identified and the Technical Specification commitment you are able to meet for each path. Provide the total leakage specification fo· leakage to untreated areas. This Technical Specification must be met assuming a single active failure.

6.2.24 Describe in detail the tests, and their sensitivity, which will be performed to determine the ability of the EVS to pull down the annulus to negative pressure and maintain it at a maximum pressure of -0.25 inch water gage at all points within the boundaries treated by the EVS.

6.2.25 Analyze each engineered safety feature air filtration system (Control Room, Fuel Handliug Building, Annulus Ventilation Filtration System) as to the positions in Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants."

6.2.26 The following information is required for iodine removal credit for containment spray in accident computations. The required information corresponds to Sections 6.2.3 and 15.1.x.2 of the Standard Format and Content of Safety Analysis reports for Nuclear Power Plants, Rev. 1, 1972.

a. Containment Air Purification and Cleanup Systems: Description of the iodine removal function of the Containment Spray System.

b. Design Basis (for iodine removal function)

c. System Design (as affected by iodine removal function). Description of systems and components employed to carry out the containment cleanup function of the spray system, including the method of additive injection (if any) and delivery to the containment. Detailed information should be provided in section 6.2.3.2 concerning:

1. Methods and equipment used to insure adequate delivery and mixing of the spray additive (where applicable).

2. Source of water supply during all phases of spray system operation.

3. Spray header design, including the number of nozzles per header, nozzle spacing and orientation (a plan view of the spray headers, showing nozzle location and orientation should be included).

4. Spray nozzle design, including the drop size spectrum produced by the nozzle. Source of the data method of measurement and expected accuracy should be discussed.

d. A description of the operating modes of the system should be given including the time of system initiation, time of first delivery through the nozzles, length of injection period, time of initiation of recirculation, and length of recirculation operation. Flow rates should be supplied for each period of operation, assuming minimum and maximum spray operation coincident with minimum and maximum safety injection flow rates.

e. Evaluation of iodine removal function of the containment spray system. The system should be evaluated for fully effective and minimum safeguards operation. In Section 6.2.3.3 specific attention should be given to the evaluation of the effects of spray solution chemistry, spray and sump pH, drop size spectrum, drop coalescence, steam condensation, drop saturation, iodine partition coefficient, containment coverage unsprayed volumes, wall effects, and mixing in the sump.

f. Description in Section 6.2.3.4 of provisions made for testing all essential functions required for the iodine removal effectiveness of the system. Where appropriate, reference may be made to Section 6.2.2.4, in order to avoid duplication.

g. Description in Section 6.2.3.5 of any additional instrumentation of the spray required for actuation and monitoring of the iodine removal function of the system.

h. Discuss in Section 6.2.3.6 the chemical composition, susceptibility to radiolytic or other decomposition, corrosion properties, etc. of the spray additive (if any), the spray solution, and the containment sump solution.

6.3 Emergency Core Cooling System

6.3.1 Compare the amount of reactivity required to maintain the core subcritical after the worst-case LOCA to the amount required after the worst-case main steam line break.

6.3.2 Does the capacity for HPI pump requirements (GPM/ft) in Table 6-11 agree with Figure 6-10?

6.3.3 Page 6-71 indicates that one HPI pump, one DH pump, one DH cooler, and both core flooding tanks are required to protect against the full spectrum of pipe breaks. How is this requirement met with a postulated double-ended break of one of the two 14-inch lines which connect a core flooding tank (CFT) to the reactor vessel? Assuming no offsite power and a single active failure (such as in one of the buses supplying emergency power, or failure of a valve to open), submit a complete analysis of this event for Davis-Besse. Also, evaluate the applicability of BAW-10045 for this break (referenced on page 6-81; specifically, deviation 4).

6.3.4 Discuss the possible consequences of the single failure in Table 6-16 which resulted in a loss of accumulator nitrogen pressure.

6.3.5 Item 6.3.2.11 of the "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants" (Revision 1, October 1972) indicates the need to distinguish between true redundancy incorporated in a system and multiple components. To complement the FSAR discussions in this regard, provide a summary of a systematic core cooling functional analysis of components required over the complete range of coolant pipe break sizes inside the containment. The summary should be shown in the form of simple block diagrams beginning with the event (pipe break), branching out to the various possible sequences for the different size breaks, continuing through initial core cooling and ending with extended or long-term core cooling. When complete, the diagram should clearly identify each safety system required to function to cool the core for all coolant pipe breaks inside the containment during any plant operating state. The attached Figure 6.3.E-1 is provided as a guide.

6.3.6 For each engineered safety feature identified in 6.3.5 above, list the auxiliaries required for its operation.

6.3.7 In light of recent operating experience in a PWR in which four out of a total of sixteen solenoid-operated air cylinder exhaust valves were discovered failed in main steam line isolation valves (one isolation valve rendered inoperable), discuss those features which minimize the potential for a similar common mode failure at Davis-Besse.

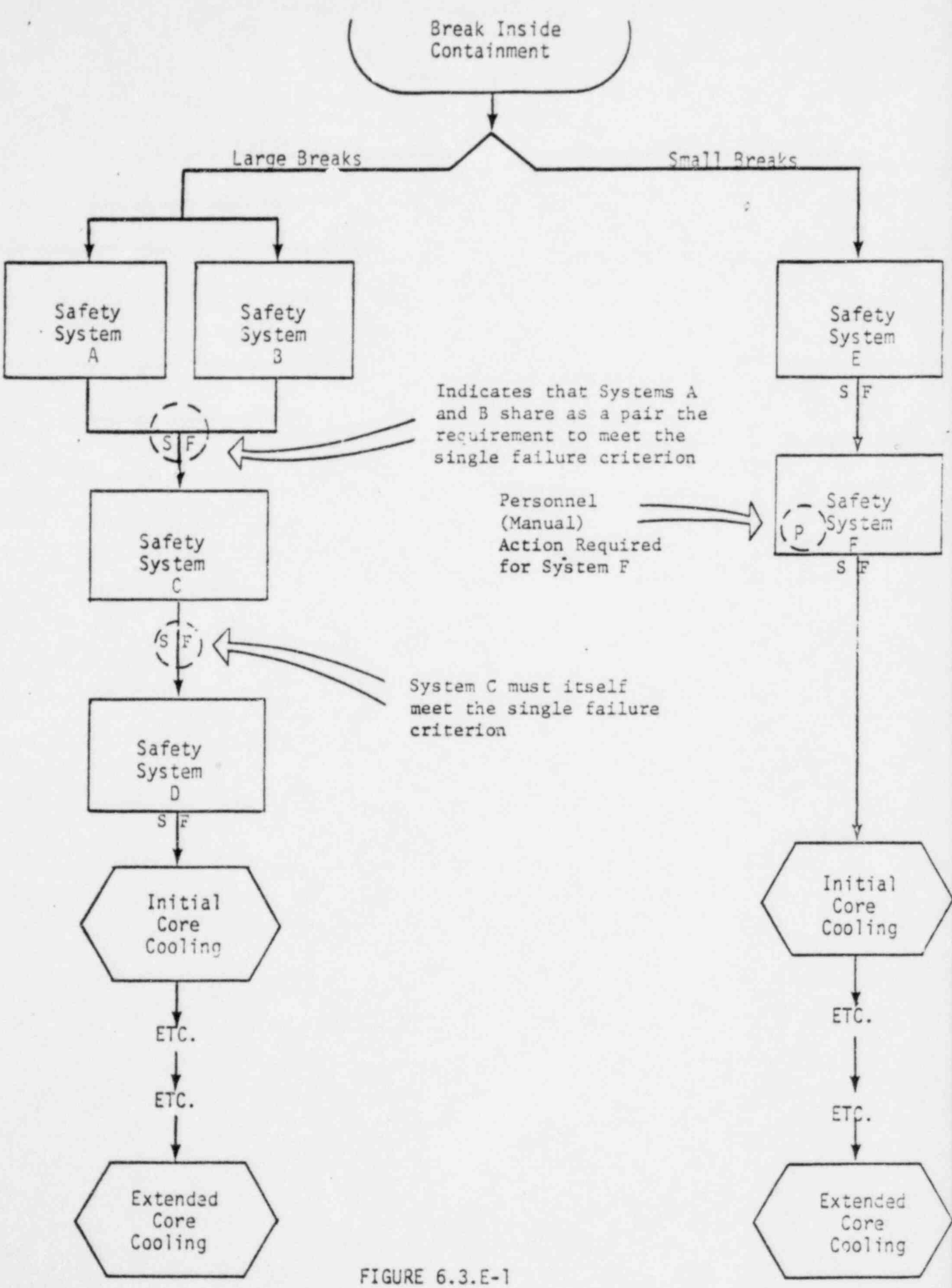
Break Inside Containment
Large Breaks
Small Breaks
Safety System A
Safety System B
Safety System E
S F
Indicates that Systems A and B share as a pair the requirement to meet the single failure criterion
S F
Safety System C
Personnel (Manual) Action Required for System F
P
Safety System F
S F
S F
System C must itself meet the single failure criterion
Safety System D
S F
Initial Core Cooling
Initial Core Cooling
ETC.
ETC.
ETC.
ETC.
Extended Core Cooling
Extended Core Cooling

FIGURE 6.3.E-1

6.3.8 A recent check of a core flooding tank (CFT) isolation valve in a PWR revealed that the valve was not in the fully open position, although all indications in the control room showed the valves to be open. What steps will be taken at Davis-Besse to provide a more reliable indication of ECCS valve positions?

6.3.9 Describe the provisions to protect the ECCS (including connections to the reactor coolant system or other connecting systems) against damage that might result from:

a. Thermal stresses

b. Seismic loads

c. LOCA loads

6.3.10 Identify all manual actions required to be taken by an operator in order for the ECCS to operate properly. Discuss the information available to the operator, the time delay during which his failure to act properly will have no unsafe consequences, and the consequences if the action is not performed at all.

6.3.11 Identify all process information available to the operator in the control room to assist in assessing post-LOCA conditions. Briefly discuss how an operator in the control room would immediately differentiate a spurious ECCS injection signal from a real need for cooling water.

6.3.12 Submit the small break analysis results for Davis-Besse that is committed in paragraph 6.3.3.1.3. The guideline of item 6.3.3.3 of the Standard Format and Content of Safety Analysis Reports For Nuclear Power Plants (Revision 1 - October 1972) should be followed.

6.3.13 Paragraph 6.3.3.2 indicates that control rod poison material (silver-indium-cadmium alloy) becomes molten at about 1470°F. Provide an estimate of the peak poison material temperature during the design basis LOCA. Compare this temperature to the peak fuel rod clad temperature and discuss the potential for and the consequences of the control rod poison material becoming molten.

6.3.14 Include data on the burnable poison rod assemblies (BPRA-$Al_2O_3B_4C$). similar to the data on page 6-82 of the FSAR.

6.3.15 Paragraph 6.3.3.1.2 indicates that the large break analysis for Davis-Besse signifying conformance to the AEC Interim Acceptance Criteria is given in B&W Topical Report, BAW-10053. Since the Regulatory Staff's records at the time of FSAR submittal showed that no such report had been received, a complete evaluation of postulated loss-of-coolant accidents for Davis-Besse should be submitted. Specifically, the guidelines of items 6.3.3.1 through 6.3.3.4 of the Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (Revision 1 - October 1972) should be followed.

6.3.16 Discuss the extent to which components or portions of other systems are required for operation of the ECCS.

6.3.17 Describe the results of analyses and tests performed to determine the effects of ECCS operation on the reactor coolant system.

6.3.18 Reconcile the apparent inconsistency of stating a containment design temperature of 264°F (page 6-4) when the calculated peak containment temperature is 266.5°F for the 14.1 ft^2 pipe break (page 6-8).

6.3.19 Provide a basis for the statement on page 6-71 that "It was considered incredible that valves would change to the opposite position by accident if they were in the required position when the accident occurred," especially in light of recent experience at a PWR in which, after a spurious signal to inject emergency core coolant, a logic malfunction improperly isolated a coolant injection valve, thereby closing a flow path to an injection header...and to the core.

6.3.20 Clarify paragraph 6.3.2.4 and Table 6-11 on the ECCS materials, indicating the stainless steels and the grades of carbon steel that were used. The term CS clad SS is misleading.

7.0 INSTRUMENTATION AND CONTROL

The Instrumentation and Control requests are FSAR Sections:

7.2 Reactor Protection System (RPS)

7.3 Safety Features Actuation System (SFAS)

7.4 System Required for Safe Shutdown

7.5 Safety Related Display Information

7.7 Control Systems

7.2 Reactor Protection System (RPS)

7.2.1 With regard to the qualification testing of protection system instrumentation as described in BAW-10003, the following additional information should be provided for further evaluation of the method used. (Items a through n below refer to pages and Tables in BAW-10003.)

a. On page 2-11, Table 2-3, include the following in the SFAS Design Basis or a discussion for their omission: (1) d-c supply ripple and relative humidity under "Reference Operating Conditions," and (2) a-c supply frequency, ambient temperature and relative humidity under "Design Range Operating Conditions."

b. Discuss the difference in power supply ripple design basis between the SFAS and NI/RPS requirements.

c. On page 2-10, Table 2-2, the d-c supply ripple requirements of 10 max p-p from 40 to 20 H_Z under "Absolute Calibration Condition" appears in error. Correct this deficiency r discuss the basis for your requirements.

d. With regard to page 2-27, Table 2-30, discuss the acceptability of the d-c supply voltage results when they do not meet the requirements.

e. With regard to page 2-38, Table 2-31:

   1. Discuss not qualifying the input line current requirements.

   2. Include the maximum loading and induction load current requirements for the tripping contact and remote trip indication.

f. With regard to page 2-29, Table 2-32:

   1. Discuss not qualifying the input d-c supply current and test signal voltage requirements.

   2. Include the "On-Test" Form A maximum loading VA requirement.

g. With regard to page 3-1 and 3-2:

   1. Describe and provide tests or discuss not qualifying all type sensors for linearity, absence of drift, resolution and repeatability.

h. On page 3-2, it is not clear whether wells for temperature detectors are utilized in the plant design. Describe response time qualification if wells are used.

i. With regard to page 3-3, provide qualification testing of the amplifier portion of the Motorola pressure transmitter to the same conditions as the pressure head or discuss this position.

j. With regard to page 5-2, include environmental qualification testing of pressure switches or discuss this exclusion.

k. With regard to page 5-3, discuss excluding temperature detector heads from abnormal environmental qualification testing.

l. With regard to page 5-4:

   1. Provide qualification radiation testing for an integrated dose to equal a plant life of 40 years plus a loss-of-coolant accident radiation environment or discuss the adequacy of 2×10^4R which the instruments have been qualified for.

   2. It is not clear whether credit was taken for air conditioning when the cabinet cooling fan was disabled.

   3. Identify the system cabinets that were included in the system cabinet environmental tests. Include a list of the equipment mounted in these cabinets during these tests.

m. With regard to page 5-7, Table 5-2:

   1. Include the amount of calibration shift for the Bailey Transmitters at accumulated radiation of 4×10^7R.

   2. The maximum inaccuracies listed are greater than the performance requirements listed on pages 3-3 and 4. Provide a discussion of these inconsistencies.

3. The type of connecting cable and connectors included in the testing of the instruments listed on this table should be identified in Section 5.

n. A section or table should be provided which identifies by type the sensors used for each parameter input and permissive input of the NI/RPS and SFAS. Identify all parameter sensors, permissive sensors and interlocks required by this design and not included in this report.

7.2.2 Table 7-4, RPS Operating Requirements, does not include ambient radiation for equipment inside the containment. Provide this information. Also, verify that Table 7-4 includes the requirements imposed by the high energy line break environment outside the containment.

7.2.3 Table 7-6, SFAS Operating Requirements, does not include ambient radiation for equipment inside or outside the containment. Provide this information. Also, verify that Table 7-6 includes the requirements imposed by the high energy line break environment outside the containment.

7.2.4 Supplement or correct the information contained in Section 7.0 with regard to the Reactor Protection System (RPS) as follows:

a. Provide all the information specified by Section 7.1.1, Identification of Safety Related Systems, of the Standard

Format and Content of Safety Analysis Reports For Nuclear Power Plants, Revision 1, dated October 1972.

b. The comparison of the Davis-Besse RPS design with that of Rancho Seco in Section 7.2.1.5 does not reflect the change in CRDCS trip circuits as described in Sections 7.4.1.1.2 and 7.4.1.1.5. Correct this inconsistency.

c. The Davis-Besse CRDCS trip circuit design utilizes solid state switches (SCR's) in the rod motor power supplies and in the motor return circuits as one means of interrupting power to the Control Rod Drive Mechanisms. The Rancho Seco design utilizes a combination of d-c breakers and SCR's to perform this function. Provide an analysis verifying that your design will give an equivalent degree of reliability as the Rancho Seco design. Your response should state if the SCR's in the motor return circuits can by themselves effect a reactor trip, i.e., without the other SCR's reverting to the open state.

d. It is noted that in Figure 7.1 the outputs of Protection Channels 3 and 4 are indicated as being "Breakers 3 and 4" respectively. These breakers are not part of your RPS design as shown in Figure 7.9(a). Also, the Protection Channels are identified as "1, 2, 3, and 4" in Figure 7.1 and as "A, B, C, and D" in Figure 7.9(a). Correct Figure 7.1 and 7.9(a) as required to reflect your CRDCS trip design, and for consistency.

e. The information presented in Sections 7.2.1.2.4, 7.2.2.1(10), and 7.2.2.4 is not sufficient to demonstrate conformance of your RPS design with all the regulatory positions of Regulatory Guide 1.22 (Safety Guide 22), particularly Positions 1(a), 1(b), and 3(a). Provide this information in the detail required for this evaluation. Your response should address conformance of your design with all the positions of Regulatory Guide 1.22, and particularly including a description of the interlocks which prevent bypassing more than one RPS channel (reference Section 7.2.1.2.4), and a description of the testability provided for all elements of the CRDCS trip circuitry. Also, your response should clarify the reference to "breaker trip lights" in Section 7.2.2.1(10). These lights are not shown on Figure 7.1.

f. Sections 7.2.1.7 and 7.2.2.2 state that the RPS does not comply with IEEE Std 338-1971, Trial Use Criteria for Periodic Testing of Nuclear Power Generating Station Protection Systems, because this standard was issued subsequent to equipment procurement. This standard is primarily concerned with periodic testing not with system design. Define the degree of conformance of the test and surveillance program for the RPS and SFAS embodied in the Technical Specifications (Section 16.0) with the provisions of IEEE Std 338-1971. Identify any system design features that preclude testing in conformance with this standard.

g. Provide a detailed description of the RPS manual trip switches including their installation. Your response should address those features of the design which implement the separation and independence requirements for redundant safety criteria.

7.2.5 The design bases for the RPS [Section 7.2.1.1(8)] and for the SFAS [Section 7.3.1.4(8)] do not identify the "accidents" referred to in the introduction or address the vulnerability of these systems to any DBA. Correct this deficiency. Your response should specifically address LOCA and the high energy line break outside the containment.

7.3 Safety Features Actuation System (SFAS)

7.3.1 Supplement or correct the information contained in Section 7.3 with regard to the Safety Features Actuation Systems (SFAS) and the associated actuation devices and actuated equipment as follows:

a. Provide all the information specified by Section 7.1.1, Identification of Safety Related Systems, of the Standard Format and Content of Safety Analysis Reports For Nuclear Power Plants, Revision 1, dated October 1972. Your response should include a comparison of your design with that of the Rancho Seco Plant.

b. Provide a functional block diagram of the SFAS which identifies and clearly shows the relationship between the sensor channels, logic channels, actuation channels, and actuated equipment.

c. Section 7.3 does not contain a sufficiently detailed description of the testability of the SFAS to permit evaluation of the conformance of your design to all the regulatory positions of Regulatory Guide 1.22 (formerly Safety Guide 22). Provide this information. Your response should (1) specifically address conformance of your design to each regulatory position on a system by system basis, (2) identify those components including actuated equipment which cannot be tested without adversely affecting the safety or operability of the plant, and (3) specifically address the testability of the diesel generator start logic and circuitry and the emergency loads sequencing logic and circuitry, through the actuation devices.

d. The "Sequence Logic" shown in Figure 7-4A does not appear to be operable. Describe the operation of this portion of your design. Also, "Figure 1-Actuation Channel" should include a reference to the drawings in Appendix 7B which show the actuation contacts applied to the various motor and solenoid control circuits.

e. Table 7-1 does not list AEC Regulatory Guide 1.29 (formerly Safety Guide 29) or AEC General Design Criteria 1, 3, 4 and 15 as being applicable to the SFAS. These criteria are applicable. Correct this omission.

f. The RPS and the SFAS incorporate means for generating signals to test and calibrate the systems (Section 7.3.2.3). Discuss the maintenance and calibrations that are performed on these test function generators.

7.3.2 The design and Technical Specification for the motor-operated isolation valves for the core flooding tanks (as described in Section 6.3.2.15) do not provide sufficient assurance that these valves will be open when required. An acceptable variation of your design would include the following features:

a. Valve position visual indication (open and closed) for each valve which is not dependent on power being available to the valve controller.

b. Visual and audible alarms for each valve when the valve is not fully open and reactor coolant pressure is above a preset value. These alarms shall be actuated by redundant and independent sensor sensing actual valve position, and by redundant and independent pressure signals.

c. A Technical Specification requirement that the reactor shall not be made critical or shall be shut down unless the motor operated isolation valve in the discharge line of each core flooding tank is open and tagged (the description states only that the reactor shall not be made critical...etc.).

Please indicate your plans and schedule to modify the design of the isolation valving to include the preferred features or to conform to other criteria that provide equivalent assurance that these valves will be open when required.

7.4 Systems Required for Safe Shutdown

7.4.1 We have concluded that the Auxiliary Feedwater System (AFS) described in Sections 7.4.1.3, 7.4.2.3, 9.2.7, and 15.2.8 is essential to plant safety and must meet the single failure criterion. We will require that the instrumentation, control, and electrical subsystems associated with the AFS (for both the automatic and manual control modes of operation) be designed to conform to IEEE Std 279-1971 and IEEE Std 308-1971. Therefore,

a. Modify your design of these subsystems to conform to these standards and criteria or justify it on some other defined basis.

b. If your design is modified, provide a sufficiently detailed description, including revised design bases and supporting analyses, to enable evaluation of the new design for conformance with the stated standards and criteria.

c. If justification of your present design is contemplated, your response should: (1) include the analyses made to determine the required AFS response time and to verify that automatic control (ICS) is not required for safety, (2) identify the means by which the operator is informed that the ICS is not operative and that manual control of the AFS is required, (3) describe the operations performed by the operator (including time required) to manually initiate AFS operation, and (4) describe the AFS instrumentation and control subsystems required for safety (including indications and alarms) in sufficient detail to enable evaluation of conformance with the stated standards and criteria.

d. Resolve the inconsistency between the AFS response time requirements given in Section 7.4.1.3.1 as 60 seconds, and in Table 16.2.8-2 as 40 seconds.

7.4.2 Supplement or correct the information contained in Sections 7.4.1.6 and 7.4.2.5 on the Auxiliary Shutdown Panel (ASP) as follows:

a. Correct the definition of the protective action (Section 7.4.2.5.1) performed from the ASP to include "establishing" as well as maintaining hot shutdown.

b. Describe the process of establishing and maintaining hot shutdown from the ASP. Your response should confirm the availability and redundancy of all the instrumentation and controls required by the operator.

c. Elaborate on the possibility of the operator being unable to trip the reactor prior to control room evacuation as required to Section 7.4.1.6.3(3).

7.4.3 Table 7-1, Safety Criteria Used in the Design of Safety Related

Instrument Systems, is not complete with regard to either the listing of safety related systems or to the applicable AEC General Design Criteria. Also, the title of the table should be revised to include "control" as well as instrumentation systems. Correct these deficiencies. Justify any differences in applicable criteria between these systems (other than RPS) and the SFAS. Correct Section 7.0 as required to be consistent with the complete tabulation of safety related systems and applicable criteria.

7.5 Safety Related Display Information

7.5.1 Provide a plan layout of the control room and adjacent equipment areas showing the location of all control consoles, control boards, cabinets and racks. Identify all safety and non-safety related components including redundant portions of the protection systems. Identify all components in the control room complex that are not Seismic Category I.

7.5.2 Define the degree of conformance of the safety systems to Regulatory Guides 1.40, 1.41, 1.47, and 1.53. With regard to Regulatory Guide 1.47:

a. The conditions of positions 3(b) and 3(c) are interpreted to include bypasses that result from manipulation of permanently installed electrical control devices located in any accessible area of the plant: and

b. The design criteria for the indication systems should reflect the importance of both providing accurate information for the operator and reducing the possibility for the indicating equipment to affect adversely the monitored safety systems. In discussing the Davis-Besse design criteria, the following should be considered:

1. The bypass indicators should be arranged to enable the operator to assess readily the operating status of each safety system and determine whether continued reactor operation is permissible.
2. Means by which the operator can cancel erroneous bypass indications, if provided, should be justified by demonstrating that the postulated causes of erroneous indications cannot be eliminated by another practical design.
3. Unless the indication system is designed in conformance with criteria established for safety systems, it should not be used to perform functions that are essential to the health and safety of the public. Neither should administrative procedures require immediate operator action based solely on the bypass indications.

4. The indication system should be designed and installed in a manner which precludes the possibility of adverse effects on the plant's safety systems. Failure or bypass of a protection function should not be a credible consequence of failures occurring in the indication equipment and the bypass indication should not reduce the required independence between redundant safety systems.

5. The indication system should include a capability of assuring its operable status during normal plant operation to the extent that the indicating and/or annunciating function can be verified.

7.5.3 Justify the exclusion of General Design Criterion 1 from the list (Section 7.5.2.1) of the criteria applicable to the surveillance systems required for safety.

7.7 Control Systems

7.7.1 Two incidents have occurred at a nuclear power plant that indicate a deficiency in the control circuit design that warrants a review of the control circuits to assure that these types of deficiencies do not exist or are corrected if they do exist. Both incidents involved the inadvertent disabling of a component by racking out the circuit breaker for a different component.

As a result of these occurrences, we request that you perform a review of the control circuits of all safety related equipment at the plant to assure that disabling of one component does not, through incorporation in other interlocking or sequencing controls, render other components inoperable. All modes of test, operation, and failure must be considered. It appears that in the cases cited above, the racked out position of breakers had not been included in the failure mode analysis of those control circuits.

Also, your procedures should be reviewed to ensure they provide that, whenever part of a redundant system is removed from service, the portion remaining in service is functionally tested immediately after the disabling of the affected portion and, if possible, before disabling of the affected portion.

8.0 ELECTRIC POWER

The Electric Power requests are in

FSAR Sections:

8.2 Offsite Power System

8.3 Onsite Power Systems

8.2 Offsite Power System

8.2.1 Provide the results of your power system stability analyses with respect to loss of (1) the nuclear unit, (2) the largest unit in the system, and (3) the most critical transmission line.

8.2.2 The information on the offsite power system design is not complete enough for evaluation of conformance with the requirements of General Design Criterion 17. Submit the following additional information:

a. Provide a scaled plan layout of the site showing all switchyards, transmission lines and associated rights-of-way, the location of switchgear and power transformers, and the routing of control and power circuits to remote structures. Identify all overhead and underground circuits. It is noted that this information request is only partially satisfied by Figure 8-3.

b. Identify any transmission line crossovers, onsite or offsite, which could jeopardize the availability of offsite power. Verify that structural failure of any one line would not result in the failure of all other offsite power.

c. Identify and describe any structures (such as microwave towers) or sources of fire, explosion or missiles located onsite or offsite which, if postulated to fail in the worst possible manner, could result in damage which precludes meeting the requirements of General Design Criterion 17.

d. Identify and justify any aspect of your design that does not meet the requirements of General Design Criterion 17.

8.3 Onsite Power Systems

8.3.1 The information contained in this section is not in sufficient detail to perform an evaluation of the Diesel Generator and associated safety related systems. Please expand this area and provide the design details, diagrams and other pertinent information that justifies that the design meets criteria that are stated.

a. Diesel jacket water cooling is provided by the component cooling water system. In the event of loss of power, there will be an interruption in the power supply to the component cooling water pumps coincidental with diesel start. Considering component cooling water pump flow coast down and restart of the pumps, discuss the effect on the cooling water flow, and what effect (if any) it will have on the performance of the diesel generators. What procedures will be used to maintain the diesel generators on-line during this interval?

b. Provide a discussion of the installed protective type devices that are incorporated in the design to protect the diesel generators from exceeding operating limits or otherwise prevent them from performing their intended function during a DBA. What measures will be taken to minimize the possibility of the above devices from needlessly preventing the diesel from operating when required?

c. Describe the measures taken to assure fast and reliable starting of the diesel engines, with respect to maintaining minimum jacket cooling water and lubricating oil temperatures.

d. The description and physical arrangement of essential subsystems for the diesels have not been adequately described. Provide a description of the design including arrangement drawings and diagrams for the following subsystems:

(a) The air intake structure and filtering system.

(b) Lubrication and its filtering system,

(c) cooling water and its sources,

(d) the fuel oil filtering system, and

(e) the batteries and starting systems.

e. The FSAR states the fuel oil storage tank and the fuel oil transfer system are not designed as Class IE structures within the meaning of IEEE-308. Please clarify this statement. Explain how you propose to maintain the diesel generator in operation for a minimum of 7 days to assure safe shutdown and maintenance of post-shutdown or post-accident station security.

f. Describe the Seismic Category I auxiliary systems which supply the diesel generator building. The response should specifically address (a) freeze protection upon loss of auxiliary boiler heating service in winter, and (b) ventilation to prevent overheating or loss of power to the fans serving the compartments during diesel operation in summer.

g. The FSAR states the diesel engine day tank capacity is sufficient for approximately 24 hours at 110% full load operation. Provide a discussion of the factors considered in arriving at this capacity. Include in the discussion the range of malfunction considered, and the time interval between the low level alarm and when the day tank will be empty. Relate the time period required to carry out the various remedial actions to the time period available.

h. With the aid of general arrangement drawings of that part of the auxiliary building housing the diesel generators, and their associated auxiliary systems, provide the results of a failure mode and effects analysis for each individual diesel generator auxiliary system. The results of the analysis should demonstrate that it is not possible for one single event to disable more than one diesel generator. Include in your discussion and analysis events such as fires, flooding, external and internal missiles (i.e., crankcase door missile created by a crankcase explosion or a failure in one of the air receiver systems).

8.3.2 Describe the switchyard batteries installation in more detail (Section 8.3.2.1.8). Discuss the independence of these power supplies.

8.3.3 Provide a listing cross-referencing the FSAR "figure numbers" with the corresponding "drawing numbers" which appear in parentheses on each figure. Such a cross-reference is required, particularly for the drawing review, because the references on all figures are in terms of "drawing number" not "figure number."

8.3.4 The design of the fuel oil storage and transfer system for the standby diesel generators (Sections 8.3.1.1.4 and 9.5.4) does not provide the redundancy and independence required for systems essential to safety; nor does it meet the criteria of IEEE Std 308-1971 with regard to seismic design (Paragraph 4.2) and fuel storage capacity (Paragraph 5.2.4(6)). The features of concern in the present design include:

a. The lack of seismic qualification of the single bulk fuel oil storage tank.

b. The vulnerability to single failure of the common suction header to the two fuel oil transfer pumps.

c. The vulnerability to single failure resulting from the interconnection of non-safety fuel oil systems (auxiliary boiler and fire pump diesel) with the standby diesels fuel oil systems.

We will require fuel oil storage and transfer systems for the standby diesel generators that are in full conformance with the above criteria. Modify the design of these systems accordingly or justify your present design on some other defined basis.

8.3.5 In Figure 8-4B, sheet 2, some breakers are identified (see note 2) as being "without tripping device." Provide a detailed description of these breakers and their design function. Justify the use of this type of breaker in the design of Class IE electric power systems.

8.3.6 Describe the design of the Class IE electric power systems downstream of 480 V Switchgear Buses "E1" and "F1" (Figure 8-4B) in sufficient detail to permit evaluation of conformance with the requirements of IEEE Std 308-1971 and Regulatory Guide 1.6 (formerly Safety Guide 6). Your response should (1) define the degree of conformance of your design with Regulatory Positions 4(a) through 4(c) of Regulatory Guide 1.6, (2) describe the switches, breakers and associated interlocks in those circuits which permit crossconnection of redundant safety buses, and (3) describe the alarms and indications provided to alart the operator that the electrical independence of redundant safety loads has been voided by means of these crossconnect circuits.

8.3.7 With reference to Section 8.3.1.18(i), identify the redundant circuits that are routed through the same penetration room and justify this design.

8.3.8 The design criteria that establish minimum requirements for preserving the independence of redundant safety cables are presented in Sections 8.3.1.2.14 through 8.3.1.2.28. Discuss the administrative responsibility and control provided to assure compliance with these criteria during design and installation of the cabling systems.

8.3.9 Provide the design bases used in sizing the batteries. Identify all d-c loads on each battery, including operating requirements during the limiting design basis event. Your response should state the capability of your design for carrying the worst case accident load assuming the unavailability of a-c power.

9.0 Auxiliary Systems

The Auxiliary Systems requests are in the following areas by FSAR Section:

9.1 Fuel Storage and Handling

9.2 Water Systems

9.3 Process Auxiliaries

9.4 Air Conditioning, Heating, Cooling and Ventilating Systems

9.5 Other Auxiliary Systems

In responding to the following requests the applicant should provide sufficient description matter and details to allow an understanding of the various systems and the capability to function without compromising directly or indirectly the nuclear safety of the plant under both normal operation or transient conditions. Emphasis should be placed on those aspects of design and operation that affect the reactor and its safety features or contribute toward the control of radioactivity and that all pertinent criteria are met.

9.0.1 In regard to potential failures or malfunctions occurring due to freezing, icing, and other adverse environmental conditions for those components not housed within temperature controlled areas and which are essential in attaining and maintaining a safe shutdown, identify and discuss the protective measures taken to assure their operation.

9.0.2 Provide a tabulation of all valves in the reactor pressure boundary and in other Seismic Category I systems, as recommended in Regulatory Guide 1.29, e.g., safety valves, relief valves, stop valves, stop-check valves, control valves whose operation is relied upon either to assure safe plant shutdown or to mitigate the consequences of a transient or accident. The tabulation should identify the system in which it is installed, the type and size of valves, the actuation type(s), and the environmental design criteria to which the valves are qualified, as stated in the design specifications.

9.0.3 For all vessels that contain gas under pressure (such as nitrogen, chlorine, hydrogen, oxygen, air and CO_2 tanks) provide the following: (a) The design and operating pressure, (b) the maximum pressure of the gas supply, (c) the location of the vessel, (d) the total energy released if the largest pipe connected to the vessel should rupture, (e) the protective measures taken to prevent the loss of function of adjacent equipment essential for a safe and maintained reactor shutdown, (f) for each vessel identify, discuss and supply the basis for any exceptions or deviation that will be taken to the positions set forth in the Occupational Safety and Health Administration OSHA 29 CFR 1910.

9.1 Fuel Storage and Handling

9.1.1 Provide the following additional information regarding the new fuel storage pit:

a. An evaluation of design loading that includes all external loads and forces, including handling.

b. Additional description of the storage pit including materials of construction, design codes and standards, seismic classification and the effect of adjacent equipment failure.

9.1.2 Provide a more detailed description of the spent fuel operations involved during the transfer of fuel from pool to the cask and the cask to loading area in the pool. Also the arrangement of the fuel pool unloading area that prohibits the fuel cask from being moved over the spent fuel pool as stated in Section 9.1.2.3.

9.1.3 Provide the following for the spent fuel pool cooling and cleanup system:

a. The spent fuel pool water quality requirements including the maximum allowable corrosion and fission products and the bases for determining when the use of the cleanup demineralizer is needed and a description of the operations needed to bring it on and off line including isolation capabilities of the system.

9.1.4 The spent fuel cooling system is designed to maintain the borated spent fuel pool water at approximately 100°F for a heat load based on the decay heat generated from 1/3 of the core fuel assemblies which have undergone infinite irradiation and have been cooled in the reactor for 150 hours prior to being transferred to the pool. The total storage capacity of the spent fuel pool is, however, designed for 1-1/3 core plus 24 spare locations. The cooling capacity for this additional core must be provided by the decay heat removal system. In view of the system design and the nature of the engineered safety functions of the decay heat removal system, please clarify and/or provide information on the following items:

a. Clarify the bases for the temporary connections between the spent fuel cooling system and the decay heat removal system; also provide information to justify the position for using single isolation valves between these two systems.

b. Present information to demonstrate that a power failure during refueling or spent fuel handling operations will not create hazardous condition.

c. Indicate the seismic and safety design classification for this system and discuss the possibility of complete loss of cooling. In light of common suction line and discharge line for both the spent fuel cooling pumps, the occurrence of such incident cannot be considered as remote.

d. Specify operating restrictions which will be imposed on the reactor when the RHR system is interconnected with and performing the cooling function for the spent fuel cooling system.

e. Describe the instrumentation and controls provided for the spent fuel pool, specifically for radiation, water level and component failure. Provide the level at which alarms are actuated and describe the action taken for each should they alarm.

f. Describe the procedure and the associated pumps, piping and valves used to supply the spent fuel pool with a Seismic Category I makeup source from the borated water storage tank and/or other emergency supply.

g. Describe, with the aid of Figure 9-3A and other details, the spent fuel storage racks and their arrangement in the pool. Include the design basis and the ability of the racks to withstand external loads, including seismic loads and impact forces due to dropped objects (indicate the largest object to be handled over the spent fuel pool).

h. In addition to Figures 1-6, 1-7, 9-26 and 9-27 provide additional plans and elevations showing dimensioned details of the fuel storage (new and spent fuel) and cask loading pools.

i. Provide a list of all major tools and servicing equipment necessary to perform the various reactor vessel servicing and refueling functions and indicate whether each is designed to Seismic Category I requirements or their storage locations are designed to these requirements.

j. Identify all the applicable codes and standards used in the design, fabrication, installation and testing of crane, rails, supporting structures, bridge, trolley, hoists, cables, lifting hooks, special handling fixtures and slings.

k. Provide the tensile properties for the hook and eye and discuss the margin of safety in terms of yield strength. Also provide data on sheave size and wire rope performance, and discuss redundancy (if provided) for hoist, motors, controls, brakes and other features of the cranes. Describe in detail the cab and pendant control features.

l. For each crane, list its design load rating preoperation test load, maximum operating loads and the test loads that will be used throughout the life of the facility.

m. Describe the modes of failure that were considered in the design of the spent fuel cask crane and reactor polar crane such as breaking of cables, lifting slings, sheared shafts, keys, stripped gear teeth, and brake failures. Also discuss the limitations and control that will exist in handling objects over and opened reactor vessel.

n. Since the coolant loop arrangement represents a departure from previous B & W design, provide an analysis of the consequences of dropping the following objects from their maximum drop heights:

a. The reactor vessel head onto an opened reactor vessel.

b. The upper core barrel assembly in an opened reactor vessel.

The evaluation should consider the maximum lift point required to remove or install the components cited above. Provide drawings and sequences of lifting operations to illustrate the evaluation. Evaluate the yield and shear strengths of the vessel support for the postulated head drop.

o. Describe and discuss the operating practices, qualifications and training of the people who will operate and/or direct the operation of the reactor and turbine building cranes. As a guide, use the Chapter 2-3.1 Operation - Overhead and Gantry Cranes USAS - B-30.2-1967 as developed by the American National Standard Safety Code for Cranes, Derricks, Hoists, Jacks and Slings.

p. What are the geometric changes of load position that may occur in the event of malfunction or failure in the hoisting system (the hoisting system includes the load and all items of mechanical and structural support on the bridge trolley)? Provide an evaluation of the effects of these geometric changes on the fuel handling and storage area and any other safety related equipment.

q. Discuss the degree of compliance of the reactor building polar crane and cask crane with OSHA Subpart N Materials Handling and Storage of 29 CFR 1910, Section 1910.179. Identify, discuss and provide a basis for any exceptions and/or deviations taken.

r. Describe and discuss the plans and means provided to absorb the resulting impact should the spent fuel cask be dropped in the spent fuel pool or cask pool. The discussion and analysis should include:

   a. An outline drawing of the cask, cask dimensions, and center of gravity.

   b. The cask weight, assumed drop height, deceleration distance, deceleration force versus stroke, velocity at impact considering deceleration caused by the pool water.

   c. The maximum possible drop height.

   d. The means, aside from administrative control, to limit the drop height to that assumed in the analysis.

   e. A description of an energy absorbing device (if used) and the vendor identification should it be commercially available.

   f. The possible modes of failure of the energy absorbing device and the inspections and surveillance to be carried out prior to each time a potential for a cask drop exists.

   g. Information which demonstrates that the cask cannot be tipped before being dropped and/or that the energy absorbing system is adequate even if it is dropped in the tipped condition.

h. The individual and combined static and dynamic concrete and reinforcing steel stresses of the fuel pool structure when the pool is subjected to its maximum normal anticipated loads as well as those experienced during impact. Also, the dynamic properties of the pool structure that are essential in establishing the dynamic stresses should ' included in this discussion.

s. Provide an evaluation of how the Regulatory Positions set forth in Regulatory Guide 1.13 "Fuel Storage Facility Design Basis" were implemented. Indicate the areas of agreement with the guide and in the cases of differences, provide justification regarding acceptability of the proposed design.

t. Prove an outline of the cask handling procedure including a sketch or drawing which shows the routing of the spent fuel handling cask from receipt to the pool for loading with spent fuel to its return to the transporting car ready for shipment from the nuclear plant.

9.2 Water Systems

9.2.1 Provide plan elevation and section drawing(s) of service water pump room. On the elevation and section drawing, provide the arrangement of the pumps, important dimensions and the minimum and extreme high water levels including the probable maximum flood.

9.2.2 Figure 9-2 indicates a single 30" service water return header through the service water tunnel. The diagram also indicates the cooling water outlets from the containment air coolers are manifolded into a single 8" return line connecting to the 30" return header. Similarly the cooling water outlets from the component cooling water heat exchangers are manifolded into a single 18" return header also connecting to the 30" return header. What are the consequences if a line break occurs immediately after the point of manifold in the 8" or 18" return lines mentioned above, or a break in the 30" service water return header inside the service water tunnel? With the aid of drawings or diagrams, discuss which essential systems would be rendered inoperable due to flooding. Include in your discussion the consideration given to passageways, pipe chases, cableways and all other possible flow paths joining the flooded space or other spaces containing essential systems and components. Discuss the effect of flooding waters on all submerged essential (electrical/mechanical) systems and components. Discuss what provisions have been made in the design to alert the control room operator in the event of system leakage or rupture. Consider instrument failures.

9.2.3 Figure 9-4 indicates a single failure in the component cooling water supply line to the reactor coolant pumps can deprive the pumps of cooling water. Provide the following information:

a. How long could the reactor coolant pumps operate at power without seizure of all pumps occurring after loss of coolant as postulated above?

b. How long would they operate at power before the sensing devices would cut power to the pump motors?

c. How long would they continue to rotate once power has been cut off?

d. Describe the sensing and associated circuitry to cope with the above situation in sufficient detail to form a valid basis of its acceptability and to assure that power to the reactor coolant pumps will be cut-off; also describe the design features which assure that it will remain functionally operable even when experiencing a single failure.

e. Assuming the loss of cooling water and that the power to the reactor coolant pumps was not cut off, provide a discussion and description of the most adverse situation that could follow the seizure of all reactor coolant pumps.

9.2.4 Identify all components that have a single barrier between the component cooling water system and the reactor coolant system e.g., RHR heat exchangers.

a. Indicate the design pressure and temperature requirements of the barriers confining the reactor coolant in the above components.

b. Indicate the operating range of the reactor coolant temperature and pressure in the above components.

c. In those cases where the pressure and temperature design requirements of the barriers in the above components are less than reactor coolant operating pressure and temperature:

1. indicate the operating modes during which these components are in use and the range of pressures and temperatures of the reactor coolant during these modes;

2. describe the controls and interlocks provided in detail for the isolation valving between the reactor coolant system and the RHR system; and,

3. for each of the above components, assume a complete failure of the barrier and describe the consequences to the component cooling water system.

9.2.5 Demonstrate that in the event of a system leak or rupture, the component cooling surge tank capacity is adequate to assure a continuous supply of component cooling water to equipment required for safe shutdown until the leak can be isolated. Describe any automatic devices provided to mitigate the effects of system leakage or rupture.

9.2.6 In view of the safety related function of the CCWS, discuss, with the aid of drawings and diagrams, the Seismic Category I source of make-up to the component cooling water system (CCWS).

9.2.7 State the design provisions made that preclude the contamination of the plant drinking water and sanitary water from radioactive sources.

9.2.8 This section of the FSAR contains design parameters and heat loads utilized in the design of the "Ultimate Heat Sink". On what basis have the heat loads been calculated? Further, a staff review of available information does not support your conclusions that the service water system meets the suggested criteria of Regulatory Guide 1.27 "Ultimate Heat Sink". Your response should provide the following:

a. The results of an analysis supporting your conclusions, in sufficient detail to permit an independent review;

b. a discussion of how the Regulatory positions set forth in Safety Guide 1.27 were implemented. Identify each exception taken and provide the bases,

c. a tabulation and plot spanning a thirty-day period of (1) the total heat rejected, (2) sensible heat rejected, (3) station auxiliary system heat rejected, and (4) decay heat from radioactive material. Use the methods set forth in the October 1971 draft Proposed ANS Standard Decay Energy Release Rates Following Shutdown of Uranium - Fueled Thermal Reactors to establish the heat input due to the decay of radioactive material. Assume an equilibrium fuel cycle* and increase the calculated heat inputs as follows:

1. For the time interval 0 to 10^3 seconds, add 20 percent to the heat released by the fission products to cover the uncertainty in their nuclear properties.
2. For the time interval 10^3 to 10^7 seconds, add 10 percent to the heat released by the fission products to cover the uncertainty in their nuclear properties.
3. For the time interval 0 to 10^7 seconds, calculate the heat released by the heavy elements (using the best estimate of the production rate for each unit) and add 10 percent to cover the uncertainties in their nuclear properties.

In submitting the results of the analysis requested, include the following information in both the tabular and graphical presentations:

a. The heat rate and total integrated heat rejected due to the fission product decay heat.

b. The heat rate and total integrated heat rejected due to the heat released by the heavy elements.

c. The heat rate and total integrated heat rejected by the Station Auxiliary Systems.

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* In this regard use the ANS formulation for finite operating time.

d. The heat rate and total integrated heat rejected due to sensible heat.

e. The maximum allowable plant inlet water temperature taking into account: (i) the rate at which the heat must be removed; (ii) the water flow rate, and (iii) the capabilities of the respective heat exchangers.

f. The required NPSH for the water pumps (taking the required water flow rates and temperatures into account).

g. The maximum available NPSH for the water pumps.

9.2.9 Section 9.2.5.1 states the design of the cooling water system incorporates a Seismic Category I return line from the service water system to the intake canal Seismic Category I area forebay. Provide the necessary diagrams and design detail drawings including plan and sections of the intake canal and return line and any additional information necessary to permit an independent evaluation of this portion of the ultimate heat sink. Provide similar drawings and other detail information for the intake structure.

9.2.10 Provide a legible plot plan of the facility indicating an identifying all essential lines (cooling, power, sensing and control) that pass between Seismic Category I structures. Discuss the measures taken to prevent the loss of those lines required to attain and maintain a safe shutdown due to seismic events, missiles from rotating equipment and tornadoes, fires, floods and the collapse of non-seismic structures.

9.2.11 In the event of an earthquake, it is assumed that condensate storage and all other Class II water systems including the seismic Class I portion of the intake forebay and all Class II structures are not functional. Under this condition, to assure safe plant shutdown, the service water system which is Seismic Category I must supply the total heat sink for plant shutdown. The total quantity of water available for this purpose will be that contained within the confines of the intake canal Seismic Category I area forebay. This volume of water must suffice for emergency feed to the steam generators and for reactor cooldown to shutdown condition and maintain this condition for a 30-day period.

a. Provide the results of an analysis to substantiate the volume of water entrapped in the Seismic Category I area of the forebay is adequate to accomplish the above function and bring the reactor to a safe shutdown condition.

b. What effect will the unclarified and untreated Lake Erie water have on the operation of the steam generators when used to cool down the reactor to 280°F? Relate the response to the development program performed for the once through steam generator.

9.2.12 Considering that steam generator (E-24-2) is effected by a main steam line break accident and assuming either normally closed valve HV 608 or HV 106 fails to open (single failure), provide a description of the procedure or a description of the modifications necessary to bring the plant down to a safe condition. Indicate the time interval required to accomplish this function and compare it to the conditions in the primary system.

9.3 Process Auxiliaries

9.3.1 In view of the fact that the compressed air system is not a safety class system and is not seismically designed, demonstrate that failure of the compressed air system will not render any safety class system components or their functions inoperable. List all air operated valves whose malfunction can affect plant safe shutdown, provide their failure mode and demonstrate that their failure mode will not compromise safe shutdown of the plant.

9.3.2 Have alternate paths been provided to obtain a sample from the reactor system or containment during accident conditions?

9.3.3 Provide a list of codes and standards used for that portion of the sampling system that interconnects to Seismic Category I systems.

9.3.4 For all components needed for safe shutdown and accident prevention or mitigation, provide a discussion on the floor drainage system serving the area where the equipment is located. The response should include sufficient plan and elevation drawings to disclose the elevation of drains and discharge points as well as the proposed routing of pertinent piping systems. Identify potential sources of water for which a single failure could cause flooding of the areas and, in this event, what effect it would have on the safe shutdown of the plant. Discuss the precautions taken to prevent flooding by the above mentioned sources. Identify the means provided by which the operator will be alerted that water is entering the area, room or component and the methods available for corrective action.

9.3.5 Provide additional explanation and assumptions used for the evaluation of the auxiliary building lower elevation drainage system sumps and sump pumps as to their capability to collect excess liquid due to an emergency flood condition.

9.3.6 Section 9.3.4.3.3 states each of the reactor coolant pumps seal injection lines contain a solenoid valve designed to fail closed upon loss of air supply. An air accumulator is provided to keep the valve open in the event of failure of the air supply. How long will the accumulator keep the valve open? What operator action is required upon loss of air supply to avoid reactor coolant pump damage?

9.3.7 Provide the maximum allowable temperature for the make-up purification demineralizers mixed bed and cation bed resin and the consequences of exceeding this temperature.

9.3.8 In addition to the normal demineralization and filteration system, the make-up flow can be diverted through a pre-filter. Provide the bases for determining when the operator will use this pre-filter.

9.3.9 The letdown temperature in the letdown line downstream of the coolers is alarmed and provides an interlock for isolation to protect the purification system. Is the letdown temperature always indicative of the pressure associated with the letdown system? Discuss the effects that the interlock failure would have on the purification system. How would excessive temperature and pressure otherwise be detected?

9.3.10 Letdown flow rates are controlled by a fixed block orifice, a parallel remotely operated valve, and a second manually positioned valve also parallel with the block orifice. Discuss the operation of these valves and describe the associated conditions required for operation. Consider also the effects on letdown flow and system pressure with either one of the valves open and with both valves open.

9.3.11 In addition to the normal make-up line, two alternate paths for adding boron to the reactor coolant system are identified. Determine the limiting condition for boration and provide the margin associated with the alterna` injection method to maintain subcriticality during reactor cooldown.

9.3.12 During cooldown of reactor coolant from 280°F to 140°F, the pressurizer is cooled by spray from the decay heat removal system. Discuss the effects on the cooldown of a single failure in the single spray line indicated in Figure 6-16.

9.3.13 The borated water storage tank is located outside the reactor and auxiliary buildings. In light of the safety function of this tank, provide the following additional information:

a. Discuss the effects of the tank heater failure.

b. Describe heat tracing requirements for the system.

c. Provide the limits of radioactivity concentration in the tank.

d. Describe the requirement for leak detection and leakage control.

9.3.14 Provide an analysis for the chemical addition system to determine the effects of system malfunction or failure on safety related equipment to control the reactor coolant chemistry and shutdown margin.

9.4 Air Conditioning, Heating, Cooling and Ventilating Systems

9.4.1 Provide a description of the ability of the control room ventilation system to detect air-borne contaminants, specifically smoke and radiation, and preclude their admission to the control room. Include in the description the detection methods, closure times of isolation valves, and time required to expedite the discharge of contaminants from the control room.

9.4.2 Describe the heating, ventilating and air conditioning system and controls of the control room and other areas shown in Figure 9-10. Describe the effects of emergency isolation of the control room on the air supply and return systems. Discuss the extent to which this system can operate with any single failure.

9.4.3 Provide a description of the smoke detectors used in the ventilation systems and indicate where they will be located with respect to air intakes.

9.4.4 State the design ventilation capacities required for the control room, equipment and cable room ventilation systems. This should include flow rates, cooling and heating requirements. Consider failures.

9.4.5 Describe the administrative controls necessary to insure that all entrance ways and other openings to the control room are normally closed. Indicate any additional steps required to assure that the pressure differential within the control room is maintained during emergencies.

9.4.6 In the isolated mode, estimate the infiltration rate into the control room assuming 1/8" Wg pressure differential across all leak paths and the maximum operation pressure differential across dampers upstream of active fans. Substantiate the leak rate by providing experimental and manufacturer's data.

9.4.7 Provide a failure mode and effect analysis for the fuel handling area ventilation system, including the effects of the inability to maintain preferred air flow patterns.

9.4.8 For the electrical penetration room, laboratories, and health physics monitor areas, provide a discussion that delineates the anticipated heat loads and their effects on radwaste ventilation system operation.

9.4.9 Provide the leakage assumptions that were used in areas housing ECCS equipment and how leakages exceeding the assumed values, up to a rupture of a pipe, will be handled for post-accident conditions.

9.4.10 Assuming a radwaste tank rupture (or any other pressure vessel containing radioactive materials) discuss the effects of:

a. the pressure pulse on the auxiliary building exhaust system;

b. this accident on the capability of the exhaust system to handle this situation and provisions made in the design to prevent contaminants from being delivered to other areas of the auxiliary building by the ventilation systems.

9.4.11 Turbine Building

Provide a description of the monitoring instrumentation, isolation capabilities, inspection and testing requirements for the turbine building.

9.5 Other Auxiliary Systems

9.5.1 Describe the potential fire hazards in each plant area and fire protection requirements and discuss the fire risk evaluation utilized in the design of the fire protection system.

9.5.2 Provide the results of a failure mode and effects analysis for the fire protection system, including an analysis of potential adverse effects caused by operation of the system. Also provide a discussion relating to the reliability of the fire detection equipment in terms of sensitivity, mean time between failures, and other operational experiences.

9.5.3 Discuss how the design assures that failure of any part of the fire protection system not Seismic Category I will not damage or prevent fire protection to a Category I structure, system or component.

9.5.4 Describe, with the aid of drawings, the fire detection and protection system provided in the circulating water pump house, turbine room, auxiliary building, transformer areas, diesel generator rooms and all other areas where fire protection is required for the safe shutdown of the plant. For the above areas provide the following information:

a. Describe the principles of operation, calibration and set point of the sensing devices that will detect the fire and automatically actuate the fire dampers in all ducts containing this equipment. Indicate if the operator has the ability to override the automatic controls actuating the fire dampers.

b. Indicate the location and distance between detectors and relate the accuracy and sensitivity of the detectors to the maximum possible size of an undetected fire assuming the flow of ventilation air in the area carries the combustion products away from the detector.

9.5.5 Discuss the potential of a fire protection system storage tank rupture and the effects upon safety related systems.

9.5.6 Demonstrate with elevation drawings that the fire pump locations are compatible with minimum and maximum supply source levels. State the required and available NPSH at minimum supply levels.

9.5.7 Provide a discussion of the precautionary measures taken to prevent the buildup of explosive mixtures of hydrogen and oxygen given off by the batteries.

9.5.8 Provide a description, including drawings, in the FSAR for the sections listed below. These sections are included in Revision 1 of the "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants" issued October 1972. In the event the information is included in other sections of the FSAR, provide references where the information may be found.

9.5.5 Diesel Generator Cooling Water System

9.5.6 Diesel Generator Starting System

9.5.7 Diesel Generator Lubrication System

10.0 Steam and Power Conversion Systems

10.1 Summary Description

10.1.1 Provide the criteria and bases for the various steam and condensate instrumentation systems. The FSAR should differentiate between operating and required safety instrumentation.

10.2 Turbine Generator

10.2.1 Describe, with the aid of drawings, the bulk hydrogen storage facility including its location and distribution system. Include the protective measures taken to prevent fires and explosions during operations, such as purging the generator, as well as during normal operation.

10.2.2 With regard to Emergency Control Operations:

a. identify all monitored parameters, whose signals are utilized in providing assurance that a turbine overspeed condition will either be prevented or will be controlled within acceptable limits;

b. identify and describe the associated components that function upon receipt of its signal in order to prevent or limit turbine overspeed to within acceptable limits;

c. for each of the above monitored parameters and associated components, describe and discuss the degree of compliance with each of the items of Section 4 of IEEE-279, Nuclear Power Plant Protection Systems;

d. identify and provide the bases for each exception to or deviation from IEEE-279.

10.3 Main Steam Supply

10.3.1 Describe the consequences of the reactor system transient expected to occur assuming all power operated relief valves in the secondary system fail to open.

10.3.2 Describe the preoperational and periodic functional tests that will be performed to demonstrate that the main steam line isolation, check, relief and bypass valves will function in accordance with design. Provide similar information for the high pressure feedwater valves.

10.3.3 Discuss the basis for the steam line isolation and non-return valve design, leakage rates and acceptance criteria for shop and inplant tests.

10.3.4 Figure 10-1 shows isolation valves FV-100 and FV-101 in the main steam lines. Assuming a failure of the main steam line upstream of the isolation valve and the non-return valve fails to close:

a. Will the isolation valve maintain the required tightness under this condition?

b. What leakages may be expected?

c. What leakages may be expected through the isolation valve if the main steam line failure occurred downstream of the isolation valve?

10.3.5 Provide a description and design evaluation of the main steam line non-return valves and the inspection and test provisions incorporated into the design. Include in the discussion what effect the failure of this valve has on the mass and energy release to the auxiliary building due to a rupture of that steam line in which it is installed.

10.3.6 Provide the criteria and basis of design that has been used to preclude the consequences of postulated high energy piping system ruptures as a result of design basis breaks outside the primary containment from having an adverse effect on safety related structures, systems or components necessary for safe shutdown. Include in the discussion a failure mode and effect analysis of the auxiliary feedwater system during the accident.

10.3.7 Describe the location, physical separation, or protective barriers provided the main and auxiliary feedwater pumps to ensure their operation if flooding or gross failure of adjacent components or structures were to occur.

10.4 Steam and Power Conversion Subsystems

10.4.1 Provide the location of all safety related equipment located within the turbine building on plan and elevation drawings.

10.4.2 Provide elevation drawings showing the water level in the turbine building at various times after a complete rupture of the main condenser circulating water rubber expansion joint. For each time increment discuss which, if any, essential system and components could be rendered inoperable. Include in your discussion the consideration given to passageways, pipe chases, cableways, and all other possible flow paths joining the flooded space to other spaces containing essential systems and components. Discuss the effect of the flood waters on all submerged essential electrical systems and components.

10.4.3 Describe the means provided to detect a failure in the circulating water system and how and in what time interval flow will be stopped, considering all factors, e.g., operator reaction time, drop-out time for control circuitry and coastdown.

10.4.4 Indicate the actuation time for all valves in the circulating water system. For each valve indicate the maximum possible closure time assuming the most likely condition to effect this. Indicate the maximum pressure peak that could be experienced due to this failure and relate this to the design pressure of the circulating water system barrier.

10.4.5 Provide a description of the chlorine treatment system for the circulating water system, with the aid of drawings:

a. the location and maximum inventory of chlorine that will be kept at the site;

b. the means of transportation and size of the incoming shipment of chlorine;

c. the range of adverse conditions that were considered during the design of the chlorine transportation, storage and utilization system;

d. the precautionary measures taken to prevent accidental release of chlorine;

e. the means provided to detect the escape of chlorine;

f. the sensitivity of the detection means in relation to the maximum continuous acceptable concentration of chlorine for operating personnel.

10.4.6 Section 10.4.4.5 states turbine bypass valves can be tested during plant operation. Aside from those opening and closing tests made during the initial startup and shutdown, describe the extent of the tests and the frequency of tests that will be performed during plant operation.

10.4.7 Provide a description and discussion of the potential and consequences of a condensate line rupture in the turbine building or other structures housing portions of the system. The discussion should include the applicable portions of requests 10.4.1/10.4.2 above as it relates to the rupture of condensate lines and its effect on safety related systems to prevent safe shutdown of the reactor.

10.4.8 Paragraph 10.4.7.2 in the FSAR states amonia and hydraxine chemicals will be used for oxygen scavenging and pH control. Discuss the handling precautions that will be taken, the location where these chemicals are stored, the maximum inventory of each that will be kept at the site, and the precautionary methods taken to protect plant personnel against adverse effects from these chemicals should failures occur in the headers or connecting piping.

10.4.9 Provide typical analysis of the lake water and the cooling tower - condenser system. Provide information that water of this quality will not lead to stress corrosion cracking of the Type 304 stainless steel condenser tubing.

10.4.10 Identify the demineralizer effluent impurity limits above which the demineralizer will be removed from service and regenerated. Include (under 10.4.6.4) the hot well instrumentation that gives warning of condenser inleakage of cooling tower water. Identify the pH of the steam generator feedwater.

11.0 RADIOACTIVE WASTE MANAGEMENT

11.6 Offsite Radiological Monitoring Program

11.6.1 TLD will be used to measure offsite radiation levels at 18 locations. the applicant states that there are three dosimeters at each location which will be changed monthly, quarterly and annually. This implies the use of one dosimeter read out at each station for each specified time interval. The Staff's position is that 2 or 3 dosimeters should be used and read out for each station at each interval to provide more reliable data for statistical anslysis.

The milk sampling frequency should be changed from monthly to weekly during the seasons that milking animals are on pasture. Also, the limits of sensitivity for ^{131}I should be at 0.5 pCi/liter at the time of sampling, instead of 2.0 pCi/liter as indicated in Table 11-57.

As pointed out in the FES the Environmental Monitoring Program omitted aquatic plants that are part of the food chain. The staff recommended monitoring the smartweed in marsh area and the applicant was advised of this requirement. State the reason for the omission in the FSAR.

12.0 RADIATION PROTECTION

12.1 Shielding

12.1.1 The FSAR states that the shielding is designed to ensure that during normal operations the exposure to operations personnel will not exceed the limits of 10 CFR 20. Section 12.3.3 also alludes to the fact that station personnel will be monitored to assure that they do not exceed the limits of 10 CFR 20. Although Table 12-4A shows that the expected annual man-rem based on operating plant data will in fact be less than the limits of 10 CFR 20, the applicant should state its management policies regarding as low as practicable doses as specific in 10 CFR 20 Section 21.0(c), methods of achieving these doses, and the persons responsible for their implementation and enforcement.

12.1.2 Fig. 12-11, "Isometric at Control Room", shows a 2-foot concrete shield. What is the shield thickness on the roof of the control room? Table 12-2 shows the principle nuclides in process equipment. The maximum total activity in the Miscellaneous Waste Evaporator Storate Tank is listed as greater than 5000 curies of high energy gamma radiation. A detailed description and drawings of the shielding around this compartment as well as other compartments containing high levels of high energy gamma radiation should be included in the FSAR. Fig. 12-1 is not of sufficient detail to determine the adequacy of shielding in pertinent areas. If shield design in an area is based on access requirements in that area, state the parameters (i. e., source strength, dose rate at point of interest, occupancy time, shield material and thickness, etc.) used in the design of specific shields to achieve the desired accessibility in the area.

12.1.3 Recording of background dose rates with the area monitors allows one to note an inadvertant increase in radiation levels below the alarm point. It also provides a permanent record of radiation levels in the area of interest. Explain why there are no automatic recording functions on any of the area radiation instrumentation. Relate to post accident monitoring conditions.

12.1.4 The applicant should either identify the reactors labeled A, B, C, D.... in Table 12-5A, or provide a reference for the tabulated data.

12.2 Ventilation

12.2.1 The applicant states that the maximum expected concentration of radioactivity on the station site will be within the limits of

10 CFR 50 Appendix I "for all areas outside the station structure but within the site boundary". Does this mean that the station will maintain ^{131}I concentration of 10^{-15} μCi/cc and ^{85}KR concentrations of 10^{-10} μCi/cc within the site boundary?

12.2.2 In the airborne radioactivity monitoring program, the applicant will monitor the fuel handling and radwaste areas, and the penetration and control rooms in the ventilation system. The staff therefore requires a diagram that shows the location of the sampling heads in the ventilation system with respect to each area being monitored. Also, describe the airborne radioactivity monitoring system that will detect particulate matter and ^{131}I at levels of 1×10^{-11} μCi/cc in a background of 10 mr/hr (see table 11-50).

Provide information on frequency of sample changing of filter papers and charcoal cartridges at each sample location.

12.2.3 What is the frequency of collection of hi-voi grab samples for area surveillance for alpha, beta, gamma activity analysis to establish the levels of airborne contamination.

12.2.4 Explain why tritium is not monitored on a continuous basis at Davis-Besse. The applicant states that when the tritium concentration exceeds 2×10^{-5} μCi/cc, supplied air masks are worn. What areas are monitored? What method is used for tritium monitoring? Describe the bioassay program for tritium uptake during normal operations and anticipated operational occurrences.

12.2.5 A statement such as "No significant dose is expected from iodine" is ambiguous. What is a significant dose? What records will be kept on airborne radioactivity measurements including noble gases and tritium?

12.3 Health Physics

12.3.1 Who is responsible for writing radiation safety procedures? What level of management reviews and signs off on these procedures?

12.3.2 Describe the procedure for calibrating the neutron survey meters for fast neutrons.

12.3.3 Describe the test facilities and fitting procedures for respirator equipment. What procedures will be used for decontamination of respirators after use in a contamination incident.

13.0 CONDUCT OF OPERATIONS

The Conduct of Operations requests are in FSAR Sections:

13.1 Organizational Structure

13.2 Training Program

13.4 Review and Audit

13.6 Station Records

A review of Revision 1 (dated 6/8/73) to the Davis-Besse FSAR revealed that you have not been responsive to our concerns transmitted to you on January 18, 1973. Only 2 of the twelve concerns were answered completely and satisfactorily (13.3.3 and 13.3.10). Provide adequate responses to the remainder of these concerns.

13.1 Organizational Structure

13.1.1 Regarding Section 13.1.3, Qualification Requirements of Nuclear Facility Personnel:

a. Which member of the plant staff meets the qualifications of the Reactor Engineer as specified in ANSI N18.1, Section 4.4.1?

b. Provide information to indicate the education and background of the assigned Shift Foremen.

13.2 Training Program

13.2.1 Section 13.2.2 details the requalification program for the licensed operators. Describe the program for the balance of the plant staff (see ANSI N18.1, §5.3-5.5).

13.4 Review and Audit

13.4.1 Section 13.4.2 lists the functions of the Company Nuclear Review Board. Expand this section to include:

a. the meeting frequency during the period of initial operation

b. a commitment to the preparation of a written chapter containing the provisions listed in ANSI N18.7, Section 4.2.1

c. the audit of all areas of plant operation every 2 years

d. a review of the audit program at least semiannually

e. a commitment that the Board will collectively have competence in each of the areas listed in ANSI N18.7, Section .2.2.2

f. a complete listing of the subjects requiring independent review by the Board (see ANSI N18.7, Section 4.3)

13.6 Station Records

13.6.1 Section 13.6.5 provides for the retention of training records of licensed personnel. Provide the same assurance for records of the balance of the plant staff, e.g., professionals, technicians, repairmen, and unlicensed operators.

14.0 INITIAL TESTS AND OPERATION

(See note 13.0.1)

Provide adequate responses to the concern transmittal to you on January 18, 1973 (14.1 and 14.2).

15.0 ACCIDENT ANALYSIS

15.1 General

15.1.1 Identify and summarize the digital computer program or analog simulation used for each event. A detailed description of mathematical models and digital computer programs or listings could be included by reference.

15.1.2 If Davis-Besse will operate at power with one or more of the reactor coolant pumps in an idle condition, provide a detailed evaluation of this operating state in regard to its effect on normal operation, transients, and accidents. Include predicted values of controlling parameters.

15.1.3 For each transient and accident in Chapter 15.0, provide a summary table of the time in core life (BOL or EOL) during which each controlling parameter would be at its worse.

15.1.4 For the reactor containment system provide:

a. Estimated course of events, as related to actuation of the containment clean-up function of the spray system.

b. Mathematical model employed to perform the analysis of iodine removal by spray, (unless this model is described in Chapter 6) and the model used to calculate the reduced doses with the spray system in operation. All assumptions made in this calculation should be specified. (e.g., if it is assumed that all fission products are uniformly distributed throughout the containment, or that the spray removal function is effective throughout the containment volume, or that the removal effectiveness is constant for a period of time, these assumptions should be stated.)

c. Identification of any computer programs used in the analysis.

d. Fission product concentrations in the containment atmosphere and the sump solution (as a function of time) used in the spray iodine removal analysis, particularly their effect on the iodine partition coefficient.

e. Justification of assumptions used, with reference to experimental data.

f. System interdependency, particularly the interdependency of containment spray and other engineered safety systems, su as filtration systems, secondary containment systems, ice condenser iodine removal, etc. on the dose reduction factor claimed for each system.

g. Results of analysis of iodine removal by sprays, and the margin of protection provided.

15.2 Class 1 - Events Leading to No Radioactive Release at Exclusion Area Boundary

15.2.1 Include in Table 15.2.1-2 the maximum numerical values for the uncontrolled control rod group withdrawal from a subcritical condition (similar to Table 15.2.4-2).

15.2.2 Clarify criterion (1) for reactor protection (page 15-37) which states, "For the failure of all four reactor pumps, the minimum DNB ratio will be less than 1.3."

15.2.3 Evaluate the consequences of a break in the 6-inch steam line to the auxiliary feed pump turbine. Provide (or reference) a piping diagram showing all pertinent valves (include both 6-inch runs from each steam generator). Examine the dynamic effects of such a pipe break on critical structures and equipment in the area. Include breaks in any headers common to both 6-inch steam lines and estimate the resulting maximum temperatures near any manual valve remote operators required to mitigate the consequences of this event.

15.2.4 Include in Paragraph 15.2.3.2.3 the maximum numerical values for the control rod misalignment (similar to Table 15.2.4-2).

15.2.5 What was the initial power (MWt) at which the loss of flow transients were evaluated?

15.2.6 The results of the pump startup transient analysis in Table 15.2.6-2 do not appear to agree with Figure 15.2.6-1 (maximum thermal and neutron power). Are they correct?

15.2.7 Explain why the event was initiated at 60% power. Discuss this same event with only one reactor coolant pump idle.

15.2.8 For the loss of external electrical load, provide plots of controlling parameters (DNBR, temperature, pressure, power) versus time throughout the event. Include specific numerical maximums and relate these peak values to appropriate criteria. Also, show variations of these parameters with beginning-of-life and end-of-life conditions.

15.2.9 Re-submit Figure 15.2.5-5 considering the effects of densification.

15.2.10 Provide a position summary for Davis-Besse of means for preventing common mode failures from negating reactor scram action, and of design features to make tolerable the consequences of failure to scram during anticipated transients.

15.2.11 Evaluate the potential for flooding of critical equipment due to ECCS line leaks or breaks. For example, consider pump room flooding potential for an ECCS line break with isolation valve failure to close.

15.2.12 Evaluate a main feedwater line break downstream of the check valve discussed in Paragraph 15.2.8.2.3 on page 15-53. Consider such a single failure as loss of an auxiliary feed pump (from either steam generator) or loss of a piping valve. Specify the peak clad temperature with the worse single failure and provide a parametric analysis of temperature, pressure, power, and DNBR versus time. Examine isolation valve closure times relative to blowdown of both steam generators and specify secondary system heat removal capabilities during this accident.

Analyze this event with a) No loss of offsite power, b) A loss of offsite power at the beginning of the accident, and c) A loss of offsite power at the time of reactor trip.

15.2.13 In reference to overpressurization of critical components designed for low pressure operation relative to the reactor coolant system or high pressure side of the secondary system, identify all critical systems or components that have a single barrier against these substantially higher operating pressures. For each of these systems or components, assume a complete failure of the barrier and evaluate the consequences.

15.2.14 With regard to the analysis of loss of all offsite power to the station auxiliaries presented in Subsection 15.2.9 of the FSAR, provide plots of DNBR versus time and reactor coolant system pressure versus time. Include the numerical minimum and maximum, respectively.

15.2.15 With regard to the analysis of excessive heat removal due to feedwater system malfunction presented in Subsection 15.2.10 of the FSAR:

a. What is the mechanism for reducing feedwater flow to prevent flooding of the steam generator? What are the consequences of such an occurrence should this mechanism fail?

b. Table 15.2.10-1 shows a high flux trip delay time of 40 seconds. Figure 15.2.10-1 shows neutron power peaking out and decreasing at 30 seconds. Provide a complete sequential discussion of the nuclear system response to each parameter in Figure 15.2.10-1 and clarify the reason for the decrease in neutron power 10 seconds prior to the quoted 40-second-delay trip. Specify the numerical maximums and DNBR minimum shown in Figure 15.2.10-1.

c. For the increase in feedwater flow at rated power, provide numerical maximums for the same parameters examined in Figure 15.2.10-1. Include the minimum DNBR.

d. Provide the minimum hot channel DNBR for the increase in feedwater flow at no load conditions and maximum numerical values of the parameters in Figure 15.2.10-2.

15.2.16 With regard to the analysis of excessive load increase (secondary side steam relief) presented in Subsection 15.2.11, provide a more detailed analysis including plots of controlling parameters and compare the results to a similar evaluation of inadvertent opening of a pressurizer safety valve (depressurization of the reactor coolant system).

15.2.17 With regard to the postulated failure of regulating instrumentation presented in Subsection 15.2.13, provide the complete analysis to confirm the contentions on page 15-67.

15.3 Class 2 - Events Leading to Small to Moderate Radioactive Releases at Exclusion Area Boundary

15.3.1 With regard to the non-radiological aspects of a loss of reactor coolant from small ruptured pipes or from cracks in large pipes presented in 15.3.1, provide a more detailed evaluation following the guideline of Section 15.1 of the "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants" (Revision 1, October 1972). Provide the results for Davis-Besse of a complete spectrum of reactor coolant pipe breaks. Include a plot of peak clad temperature versus break size up to and including a 0.5 ft^2 break. Consider breaks or cracks in all lines included in and connected to the reactor coolant system, such as hot leg, cold leg, core flooding tank injection line, and lines connected to the pressurizer.

Also, provide a discussion of the smallest break which will actuate ECCS.

15.3.2 With regard to the inadvertent loading of a fuel assembly into an improper position presented in Section 15.3.3, support the contention on page 4-43 that the operator would be able to discover such an error by monitoring the incore detectors. Compare anticipated readings with and without the fuel misloading.

15.4 Class 3 - Design Basis Accidents

15.4.1 With regard to the double-ended main steam line break analysis on page 15-105:

a. Discuss the potential for and consequences of an operator error allowing auxiliary feedwater to be admitted to the affected steam generator. Consider a parametric analysis similar to Figure 15.4.4-1 and include plots of time of manual override versus maximum reactivity and power level.

b. Clarify the reason for showing the results of a 24-inch steam line break in Figure 15.4.4-1, especially since the largest steam line in Davis-Besse is shown in Figure 10-1 as 36-inches.

c. Include a plot of DNBR and pressure versus time.

d. Provide a comparison of controlling parameters with and without offsite power.

e. The division between a minor and major steam line break (Class 2 versus Class 3) is not clear. Define Subsection 15.3.2 and 15.4.4 in terms of line size.

f. Provide the time which was assumed for the operator to switch the feedwater valve controller to manual in Figure 15.4.4-1.

g. Compare the consequences of a double-ended 36-inch main steam line break both inside and outside the containment vessel.

h. Justify taking credit for turbine stop valve closure, especially in light of page 3-3 which shows Seismic Category I piping only out to and including the main steam isolation valve.

i. Provide a summary table of calculated maximum pressure drops across a main steam line check valve during a main steam line rupture.

Also include an estimate of the pressure drop during normal operation. Use 50%, 75%, and 100% power as the imposed loading conditions and state the maximum pressure drop allowed.

15.4.2 With regard to the break in instrument lines or lines from the reactor coolant system that penetrate the containment vessel (subsection 15.4.5):

a. Why was this event considered a Class 3 Design Basis Accident while a "Loss of Reactor Coolant From Small Ruptured Pipes or From Cracks in Large Pipes Which Actuates Emergency Core Cooling" (subsection 15.3.1) was considered Class 2?

b. Please provide the paragraph number in Chapter 6 of the detailed analysis of potential leakage from engineered safety features referred to on page 15-118.

15.4.3 Evaluate the inadvertent operation of ECCS during no-load and power operation.

15.4.4 Item b-1 indicates only a continuous chlorine release was assumed. Assume a failure which results in an instantaneous release of 25% of the chlorine with subsequent boil-off of the balance of the chlorine. Infiltration into the control room should be taken into account in the computation.

15.4.5 Confirm the accuracy of the 10,960 CFM flow rate given in the assumptions in Sections 15.4.1.2.2(b) and Section 15.4.6.4(d).

15.4.6 A control room inleakage of 1 CFM during isolation is unrealistic. (See comment on Section 9.4.1.3 requiring an infiltration analysis). Recalculate the control room operator doses based on (1) a 25 CFM and (2) a 100 CFM infiltration rate assumption.

16.0 Technical Specifications

16.0.1 Reactor Coolant Chemistry

For the Technical Specifications regarding reactor coolant chemistry, provide information that demonstrates that the impurity levels and maximum time allowed before action is taken (items 3.15-1, 2, 3, 4, and 5) can be satisfactorily met by sampling three times weekly, as specified in Table 4.1-3 of the Technical Specifications. Reference (4), page 3.1-15 of the Technical Specifications is not an acceptable basis for the assumption that "the oxygen and halogen limits specified are at least an order of magnitude below concentrations which could result in damage to materials ... if maintained for an extended period of time," since the data referenced were obtained at 500°F with pH 10.6 water containing 50 ppm phosphate, and are therefore not applicable to neutral or pH 4-8.5, phosphate-free water containing H_3BO_3.

16.0.2 Table 4.1-3, Item 5: We recommend that the secondary also be sampled weekly for pH and conductivity measurements.

16.0.3 Provide the Technical Specifications on the Radiation and Respiratory Protection Program as required in Table 16-1, VI H of the Standard Format and Content of SAR Reports.

17.0 QUALITY ASSURANCE

17.2 Quality Assurance Program For Station Operation

17.2.1 The description of the Quality Assurance (QA) Program for the operational phase of the Davis-Besse Nuclear Power Station (Davis-Besse) is not adequate. Section 17.2 Quality Assurance Program for Station Operation of the FSAR does not contain complete organizational charts, an adequate description of a QA Program, and clear implementation descriptions for criteria three through eighteen of 10 CFR 50, Appendix "B". The implementation of Safety Guide 1.33 is not described.

To complete our evaluation, we require Toledo Edison (TE) to provide the following information relative to their QA Program for Operations for Davis-Besse.

a. Provide organizational charts and descriptions of the organizational structure which show the company and plant organizational positions, individuals, and groups responsible for performing the QA functions as defined in 10 CFR 50, Appendix "B" for operation, maintenance, repair, modification, and fueling of the Davis-Besse Plant.

b. Provide a description of the responsibilities, authority and independence of those individual positions or groups within TE responsible for formulating, establishing and implementing QA related policies, procedures, and instructions for the operational phase. Include the activities of operation, maintenance, repair, and modification. Identify the organizational positions responsible for originating, reviewing, and approving the QA program policies procedures, and instructions.

c. Define your qualification requirements necessary to fulfill positions that manage or supervise the offsite and onsite QA activities.

d. Describe the duties, authority, and independence for the organizational positions and groups which perform review, inspection, and auditing activities.

e. Provide a description of the major attributes, including purpose and scope, of those QA procedures which assure that the activities of operating, maintaining, repairing, and modifying the Davis-Besse Nuclear Power Station comply with the criteria of 10 CFR 50, Appendix "B". Include a cross index chart or a listing which shows each QA Program procedure with the applicable criteria of 10 CFR 50, Appendix "B". Identify the responsible individual or group who originates, who reviews, and who approves each procedure.

f. Describe those provisions within the QA program which demonstrate compliance with the guidelines presented in Safety Guide 1.33, Quality Assurance Program Requirements for Operations". Identify any exceptions and justify alternate proposals.

g. Describe the formal indoctrination and training program which has been or will be established for all those personnel performing QA related activities which will assure proficiency of implementation of QA policies, procedures, and requirements. If the program does not already exist, provide a date for implementation.

h. Identify those individual positions or groups responsible for reviewing and approving the QA programs and QA manuals for contractors and vendors.

i. Describe the administrative controls which assure that the QA program policies, procedures, and instructions, including changes thereto, are distributed and implemented in a timely manner by the responsible individuals or groups.

j. Briefly describe those procedures which identify quality related records to be retained, the retention period, the storage location and the assigned responsibility.

k. Identify and describe those audits performed by company management which confirm independent assurance and evaluation of the QA program policies, activities, and procedures. The purpose of these audits is to assure effective, meaningful compliance with company policy and 10 CFR 50, Appendix "B" on a periodic, scheduled basis. Identify the organizational positions which will perform the audit, list report distribution, and state audit schedule.

l. Identify and describe those independent, scheduled audits which provide a comprehensive verification and evaluation of all phases of the QA Program activities and procedures to confirm on a continuous basis that a meaningful and effective QA Program is in effect. Identify the organizational positions which will perform the audit, state audit schedule, and list report distribution.

Attachment A

PIPE WHIP ANALYSIS

Analyses are required to assure that pipe motion caused by the dynamic effects of postulated design basis breaks will not impact or overstress any structures, systems or components important to safety to the extent that their safety function is impaired or precluded. The analysis methods used should be adequate to determine the resulting loadings in terms of:

a. the kinetic energy or momentum induced by the impact of the whipping pipe, if unrestrained, on a protective barrier or a component important to safety,

b. the dynamic response of the restraints induced by the impact and rebound if any, of the ruptured pipe.

The basis used to determine the magnitude of jet thrust force as required in dynamic analysis should be provided.

The methods of dynamic analysis specified in II and III are acceptable provided the following associated criteria are met:

I. Pipe Whip Dynamic Analysis Criteria

a. An analysis of the pipe run or branch should be performed for each longitudinal and circumferential postulated rupture at the design basis break locations.

b. The loading condition of a pipe run or branch prior to postulated rupture in terms of internal pressure, temperature, and stress state should be those conditions associated with reactor operating condition (normal and upset).

c. For a circumferential rupture, pipe whip dynamic analysis need only be performed for that end (or ends) of the pipe or branch which is connected to a contained fluid energy reservoir having a sufficient capacity to develop a jet stream.

d. Dynamic analysis methods used for calculating the piping or piping/restraint system response to the jet thrust developed following postulated rupture should adequately account for the effects of:

   (1) mass inertia and stiffness properties of the system,

   (2) impact and rebound (if any) effects as permitted by gaps between piping and restraint,

   (3) elastic and inelastic deformation of piping and/or restraint and

   (4) limiting boundary conditions.

e. The allowable design strain limit for the restraint should not exceed 0.5 ultimate uniform strain of the materials of the restraints. The method of dynamic analysis used should be capable of determining the inelastic behavior of piping-restraint system response within these design limits.

f. A 10% increase of minimum specified design yield strength (Sy) may be used in the analysis to account for strain rate effects.

g. Dynamic analysis methods and procedures should consist of:

(1) a representative mathematical model of the piping system or piping/restraint system,

(2) the analytical method of solution selected,

(3) solutions for the most severe response among the design basis breaks analyzed,

(4) solutions with demonstrable accuracy or justifiable conservatism.

h. The extent of mathematical modeling and analysis should be governed by the method of analysis selected among those specified by these criteria.

II. Acceptable Dynamic Analysis for Restrained Piping Systems

a. Acceptable Models for Analysis for ASME Class 1, 2 and 3 piping systems are:

(1) Lumped-Parameter Analysis Model; Lumped mass points are interconnected by springs to take into account inertia and stiffness effects of the system, and time histories of responses are computed by numerical integration to account for gaps and inelastic effects.

(2) Energy-Balance Analysis Model; Kinetic energy generated during the first quarter cycle movement of the ruptured

pipe as imparted to the piping/restraint system through impact is converted into equivalent strain energy. Deformations of the pipe and the restraint are compatible with the level of absorbed energy. For applications where pipe rebound may occur upon impact on the restraint an additional amplification factor of 1.5 should be used to establish the magnitude of the forcing function in order to determine the maximum reaction force of the restraint after the first quarter cycle of response. Amplification factors other than 1.5 may be used if justified by more detailed dynamic analysis.

(3) Static Analysis Model - The jet thrust force is represented by a conservatively amplified static loading, and the ruptured system is analyzed statically. The amplification factor used to establish the magnitude of the forcing function should be based on selection of a conservative value as obtained by comparison with the factors derived from detailed dynamic analysis performed on comparable systems.

III. Acceptable Dynamic Analysis for Unrestrained Pipe Whip

a. Lumped-Parameter Analysis Model as stated in II.a(1) is acceptable.

b. Energy-Balance Analysis Model as stated in II.a(2) is acceptable. The energy absorbed by the pipe deformation may be deducted from the total energy imparted to the system.

c. The assumptions used to guide the mechanism of pipe movement should be justified to be conservative.

d. The results of analysis should be expressed in terms compatible with the approach used for verifying the design adequacy of the impacted structure.

IV. <u>Flow Thrust Force</u>

a. The time function of the thrust force induced by jet flow at the design basis pipe break location should consider: (1) the initial pulse, (2) the thrust dip, and (3) the transient function.

b. A steady state forcing function can be used when conditions as specified in e below are met. The function should have a magnitude not less than

$$T = K_p A$$

where

p = system pressure prior to pipe break

A = pipe break area, and

K = thrust coefficient.

Acceptable K values should not be less than the following:

(a) 1.26 for saturated steam, water and steam/water mixture

(b) 2.00 for subcooled water-nonflashing.

c. A pulse rise time not exceeding one millisecond should be used for the initial pulse, unless longer crack propagation times or rupture opening times, can be substantiated by experimental data or analytical theory.

d. The transient function should be provided and justified. The shape of the transient function, IV a.(3) above, should be related to the capacity of the upstream energy reservoir, including source pressure, fluid enthalpy, and the capability of the reservoir to supply high energy flow stream to the break area for a significant interval. The shape of the transient function may be modified by considering the break area and the system flow conditions, the piping friction losses, the flow directional changes, and the application of flow limiting devices.

e. The jet thrust force may be represented by a steady state function, b above, provided the following conditions are met:

   (1) The transient function, IV a.(3) above, is monotonically diminishing.

   (2) The energy balance model or the static model is used in the analysis. In the former case, a step function amplified to the magnitude as indicated in II.a(2) is acceptable.

   (3) The energy approach is used for the impact effects of the unrestrained piping.

ATTACHMENT B

STRUCTURAL DESIGN CRITERIA FOR EVALUATING THE
EFFECTS OF HIGH-ENERGY PIPE BREAKS ON
CATEGORY I STRUCTURES OUTSIDE THE CONTAINMENT

CONTENTS

A. INTRODUCTION

General Design Criterion 4 of Appendix A to 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants," necessitates that structures important to safety, classified as Category I structures, shall be designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing and postulated accidents. These structures shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids associated with postulated high-energy pipe rupture accidents and from events and conditions outside the nuclear power unit.

This document presents a set of acceptable criteria for evaluating and assuring the required protection. It is assumed that the following steps, which are not structural in nature and are thus not within the scope of this document, have already been performed and the necessary design parameters already defined:

1) Systems in which pipe breaks are postulated and for which protection against the effects of such breaks should be provided, have been defined,
2) Locations of postulated breaks and type and orientation of each break, guillotine or longitudinal, have been determined,

3) Protection criteria for each postulated break have been established. This should identify the structures, systems and components to be protected from the effects of the break, and

4) All induced loadings for each postulated break are defined, including:

   a) Differential pressure across compartments, if any, as a function of time,

   b) Jet impingement force, if any, on a protective barrier, as a function of time, and

   c) Whipping pipe impact parameters, if any, on a protective barrier or a pipe restraint, including the equivalent mass, impact area and impact velocity.

B. LOADS, DEFINITION OF TERMS AND NOMENCLATURE

The following nomenclature and definition of terms will apply to all the criteria that follow in this document.

All the major loads to be encountered and/or to be postulated during a high-energy pipe rupture event are listed. All the loads listed, however, are not necessarily applicable to all the structures and their elements in a plant. Loads and the applicable load combinations for which each structure has to be checked and evaluated will depend on the conditions to which that particular structure could be subjected.

B.1 NORMAL LOADS

Normal loads are those loads to be encountered during normal plant operation. They include the following:

D ---- Dead loads and their related moments and forces, including any permanent equipment loads, and prestressing loads, if any.

L ---- Live loads, present during the pipe rupture event, and their related moments and forces.

T_o ---- Thermal loads during normal operating conditions.

R_o ---- Pipe reactions during normal operating conditions.

B.2 SEVERE ENVIRONMENTAL LOADS

Severe environmental loads are those loads that could infrequently be encountered during the plant life. Included in this category are:

Feqo - Loads g nerated by the Operating Basis Earthquake or, if an OBE is not specified, loads generated by half the Safe Shutdown Earthquake. If both are specified, they shall be the largest of the two.

B.3 EXTREME ENVIRONMENTAL LOADS

Extreme environmental loads are those loads which are credit are highly improbable. They include:

Feqs - Loads generated by the Safe Shutdown Earthquake.

B.4 ABNORMAL LOADS

Abnormal loads are the loads generated by a postulated high-energy pipe break accident within a building and/or compartment thereof. Included in this category are the following:

P_a ---- Pressure equivalent static load within or across a compartment and/or building, generated by a postulated break, and including an appropriate dynamic factor to account for the dynamic nature of the load.

T_a ---- Thermal loads under thermal conditions generated by a postulated break and including T_o.

R_a ---- Pipe reactions under thermal conditions generated by a postulated break and including R_o.

Y_r ---- Equivalent static load on a structure generated by the reaction on the broken high-energy pipe during a postulated break, and including an appropriate dynamic factor to account for the dynamic nature of the load.

Y_j ---- Jet impingement equivalent static load on a structure generated by a postulated break, and including an appropriate dynamic factor to account for the dynamic nature of the load.

Y_m ---- Missile impact equivalent static load on a structure generated by or during a postulated break, like pipe whipping, and including an appropriate dynamic factor to account for the dynamic nature of the load.

In determining an appropriate equivalent static load for P_a, Y_r, Y_j and Y_m, elasto-plastic behavior may be assumed with appropriate ductility ratios and as long as excessive deflections will not result in loss of function.

B.5 OTHER DEFINITIONS

S ---- For structural steel, S is the required section strength based on the elastic design methods and the allowable stresses defined in Part 1 of the AISC "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," February 12, 1969.

U ---- For concrete structures, U is the section strength required to resist design loads and based on methods described in ACI 318-71.

Y ---- For structural steel, Y is the section strength required to resist design loads and based on plastic design methods described in Part 2 of AISC "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," February 12, 1969.

C. LOAD COMBINATIONS AND ACCEPTANCE CRITERIA FOR CATEGORY I CONCRETE STRUCTURES

The following presents an acceptable set of load combinations and allowable limits to be used in evaluating and checking Category I concrete structures outside the containment for the effects of high-energy pipe breaks. Concrete barriers, used to provide a shield against the effects of high-energy pipe breaks, will have to maintain their structural integrity under all credible loading conditions. To assure that the structural integrity will be maintained, limits on the required strength capacities are recommended.

C.1 LOAD COMBINATIONS

The following load combinations should be satisfied:

1) $U = D + L + T_a + R_a + 1.5\ P_a$

2) $U = D + L + T_a + R_a + 1.25\ P_a + 1.0\ (Y_r + Y_j + Y_m) + 1.25\ \text{Feqo}$

3) $U = D + L + T_a + R_a + 1.0\ P_a + 1.0\ (Y_r + Y_j + Y_m) + 1.0\ \text{Feqs}$

The maximum values of P_a, T_a, R_a, Y_j, Y_r and Y_m, including an appropriate dynamic factor, shall be used unless a time-history analysis is performed to justify otherwise.

Both cases of L having its full value, possibly present during the pipe rupture event, or being completely absent should be checked for.

For combinations (2) and (3), local stresses due to the concentrated loads Y_r, Y_j and Y_m, may exceed the allowables provided there will be no loss of function of any safety-related system.

Existing structures will have to be checked and evaluated for the above three combinations. The failure capacity of concrete structures may be checked by using the "Yield Line Theory." The combined loads should not exceed 90% of the calculated failure capacity. In such situations, however, it should be verified that neither excessive deflections nor excessive cracking, will result in the loss of function of any safety-related system.

D. LOAD COMBINATIONS AND ACCEPTANCE CRITERIA FOR CATEGORY I STEEL STRUCTURES

Category I steel structures outside the containment, whose function is to provide protection against the effects of high-energy pipe breaks, will have to maintain their structural integrity under all credible loading conditions. To assure this, limits on resulting stresses or required strength capacities are recommended.

D.1 LOAD COMBINATIONS

a) If elastic working stress design methods are used:

1) $1.6\ S = D + L + T_a + R_a + P_a$

2) $1.6\ S = D + L + T_a + R_a + P_a + 1.0\ (Y_j + Y_r + Y_m) + Feqo$

3) $1.6\ S = D + L + T_a + R_a + P_a + 1.0\ (Y_j + Y_r + Y_m) + Feqs$

b) If plastic design methods are used:

1) $.90\ Y = D + L + T_a + R_a + 1.5\ P_a$

2) $.90\ Y = D + L + T_a + R_a + 1.25\ P_a + 1.0\ (Y_j + Y_r + Y_m) + 1.25\ Feqo$

3) $.90\ Y = D + L + T_a + R_a + 1.0\ P_a + 1.0\ (Y_j + Y_r + Y_m) + 1.0\ Feqs$

In combinations D.1(a) and (b) thermal loads can be neglected when it can be shown that they are second[illegible] and self-limiting in nature and where the material is ductile.

In combinations (1), (2) and (3), the maximum values of P_a, T_a, R_a, Y_j, Y_r and Y_m, including an appropriate dynamic factor, shall be used unless a time-history analysis is performed to justify otherwise.

Both cases of L having its full value, possibly present during the pipe rupture event, or being completely absent should be checked for.

For combinations (2) and (3), local stresses due to the concentrated loads Y_r, Y_j and Y_m may exceed the allowables provided there will be no loss of function. Furthermore, in computing the required section strength, S, the plastic section modulus of steel shapes may be used.

Existing structures will have to be checked and evaluated for the above three combinations. The 0.90 reduction factor applied on the required section strength, Y, can be increased to 1.0. In such situations, however, it should be verified that excessive deflections will not result in the loss of function of any safety-related system.

E. ACCEPTABLE PROCEDURES FOR DETERMINATION OF THE EFFECT OF AN IMPACTING WHIPPING PIPE ON CONCRETE AND STEEL STRUCTURES

If pipe whipping is permitted and if the whipping pipe can impact a barrier whose structural integrity has to be maintained during and after the event of the pipe rupture, the barrier will have to be designed to resist that impact. Essentially, the impacting pipe can be considered as a missile for which the following parameters can be defined: impa velocity, impact equivalent area and equivalent mass of the missile. Procedures used in determining these parameters are outside the scope of this document. Missile barriers, whether concrete or steel, should have sufficient strength to stop the postulated missile. To accomplish this objective, prediction of local and overall damage due to missile impact is necessary.

Local damage prediction, in the immediate vicinity of the impacted area, includes estimation of the depth of penetration and whether secondary missiles might be generated by spalling in case of concrete targets. Overall damage prediction includes estimation of the structural response of the target to the missile impact, including structural stability and deformations.

E.1 LOCAL DAMAGE PREDICTION

a) In Concrete

There are several empirical equations available to estimate missile penetration into concrete targets. The most commonly used is the modified Petry equation, as given by A. Amirikian in "Design of Protective Structures," Bureau of Yards and Docks, NP-3726 (1950). This equation, having been widely used, is presently acceptable. Should other equations, however, be used, the level of conservatism in these equations should be comparable to that of the modified Petry equation. Actual testing for determining penetration in concrete is acceptable.

b) In Steel

Extensive series of tests were conducted by the Stanford Research Institute on penetration of missiles into steel plates. The results of these tests were summarized by W. B. Cottrell and A. W. Savolainen in Chapter 6 of Vol. 1 of U. S. Reactor Containment Technology, ORNL-NSIC-5. Equations for penetration of missiles into steel plates presented in this chapter, having been widely used, are presently acceptable. Should other equations, however, be used, the level of conservatism in these equations shall be comparable to that of those mentioned above. Actual testing for determining penetration in steel is acceptable.

E.2 OVERALL DAMAGE PREDICTION

The response of a structure to a missile impact depends largely on the location of impact, e.g., midspan of a slab or near the support, on the dynamic properties of the target and missile and on the kinetic energy of the missile. In general, it will be conservative to absorb all the missile kinetic energy into structural strain energy in the target. However, energy losses due to missile deformation and local penetration may be accounted for.

After a check has been made on whether the missile will penetrate the barrier or not, an equivalent static load can be determined from which the structural response, in conjunction with other loads that might be present, can then be evaluated using conventional methods. An acceptable procedure for such an analysis is presented in a paper by Williamson and Alvy, of Holmes and Narver, Inc. entitled "Impact Effects of Fragments Striking Structural Elements," NP-6515 (1957).

Should other methods be used, however, the level of conservatism in these methods should be comparable to that of those mentioned above.

F. ACCEPTABLE PROCEDURES FOR DESIGN OF STRUCTURAL PIPE RESTRAINTS

Protection of Category I structures, systems and components from the dynamic effects of postulated high-energy pipe ruptures can be accomplished in some situations by providing pipe restraints in critical locations on the piping systems. These restraints should function mainly by preventing the ruptured pipe, or portions thereof, from becoming a missile that might impact and damage other critical systems, and by preventing the ruptured pipe from whipping and impacting critical systems not capable of resisting such an impact. The restraints may be independent of dead and live load supports and of seismic restraints. However, should a pipe whip restraint be intended to function also as an operating dead load and/or seismic restraint, all applicable loads should be considered in the design of the restraint.

F.1 ANALYSIS METHOD

The structural analysis of pipe restraints may consist of an energy-balance approach, where a potential collapse mechanism is first established. The displacement of this mechanism will reach its limit, by conservation of energy principles, when the external work available equals the internal work done on the restraint.

External work expressions may include kinetic expressions where mass and velocity of the ruptured pipe are known. Internal work expressions are graphically represented by the area under a resisting force-displacement curve.

F.2 ALLOWABLE YIELD STRENGTH

Due to the high rate of strain that the structural restraint would experience after pipe rupture, and partly due to the strain-hardening effects, the static yield strength of the material used may be increased by 15%.

F.3 ALLOWABLE STRAINS

In general, strains of up to 50% of ultimate strain are acceptable, provided there is no loss of function. Where buckling is critical in compression members, the load on the members should be limited to 90% of the buckling load.

F.4 GAP EFFECT

Where gaps are provided between pipes and restraints, the kinetic energy of the pipe impacting the restraint may be critical and should not be ignored. Moreover, the kinetic energy of the pipe after rebound may be more critical and should also be considered.

F.5 ANCHOR DESIGN

Pipe restraints should be anchored in concrete and/or steel structures. Strains and/or stresses induced in the structure by loading the restraint should be considered and the design of the structure should be checked in accordance with criteria already presented in this document.