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DRAWINGS CITED IN THIS CHAPTER*

*The listed drawings are included as "General References" only, i.e., refer to the drawings to obtain additional detail or to obtain background information. These drawings are not part of the UFSAR. They are controlled by the Controlled Documents Program.

DRAWING*	SUBJECT
M-2	General Arrangement, Main Floor Plan
M-3	General Arrangement, Mezzanine Floor Plan
M-4	General Arrangement, Ground Floor Plan
M-5	General Arrangement, Basement Floor Plan
M-7	General Arrangement, Sections "A-A" and "B-B"
M-10	General Arrangement, Crib House

3.0 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

This chapter describes the principal architectural and engineering design aspects of Dresden Station. These include the structural and environmental design requirements imposed by consideration of natural phenomena and plant accidents.

3.1 CONFORMANCE WITH NRC GENERAL DESIGN CRITERIA

Section 3.1.1 contains an evaluation of the design basis of Dresden Station with respect to the first draft of the 70 proposed "General Design Criteria for Nuclear Power Plant Construction Permits" issued by the Atomic Energy Commission in July 1967.

The design basis of Unit 2 was later evaluated against the final "General Design Criteria for Nuclear Power Plants," published as 10 CFR 50, Appendix A in July 1971. This evaluation is presented in Section 3.1.2.

3.1.1 Compliance with Draft Design Criteria

This section presents the proposed General Design Criteria (issued July 1967) which were used by the AEC to evaluate the original design of Dresden Station. Changes to the plant design are completed in accordance with the requirements of the EGC quality assurance program. This program assures compliance with 10 CFR 50, Appendix B; the ASME Code; and plant Technical Specifications. It should be noted that no attempt has been made to update this report to reflect current NRC design criteria. This section, which originated as FSAR Appendix B, was transferred to the updated FSAR intact, as a reference to the original plant design criteria. It has since been incorporated into this section of the UFSAR.

The following changes should be kept in mind:

A. Unit 1 is no longer licensed and operating. It was decommissioned in August of 1984. Some systems, such as fire protection are still functioning and provide services to Units 2 and 3.

B. The proposed criteria that are presented in this section have subsequently been finalized and adopted by the NRC and can be found in 10 CFR 50, Appendix A.

It should be recognized that these proposed criteria were issued in July 1967 on a draft basis in order to secure comments from the industry. These proposed criteria were not adopted as regulatory requirements at the time Dresden was built. This first draft of the criteria contained many aspects which required modification or clarification prior to adoption of the 64 criteria in 10 CFR 50, Appendix A.

There was no attempt here to comment on the proposed criteria wording. Rather the draft was used as a basis for conducting a reference audit by subject matter.

Contained herein is an evaluation of the design basis of Dresden Station relative to each of the nine groups of proposed criteria. (The draft criteria are separated into nine groups by subject matter.) In each group a statement of the applicant's interpretation of the intent of the criteria of that group is made. A discussion of the plant design conformance to this interpretation of intent is presented. The text of each of the 70 draft general design criteria is provided. A complete list of references indicates where the subject material of the individual criterion is found in the FSAR. UFSAR references provide a convenient reference to the corresponding text in the current UFSAR. However, it should be noted that the CECo response was based on the referenced FSAR sections.

Based on the applicant's understanding of the intent of the proposed criteria, it was felt that Dresden Station fully satisfies the intent of the criteria.

3.1.1.1 Group 1 - Overall Plant Requirements

The intent of the proposed criteria for this group is to identify, record, and justify the adequacy of the quality control and assurance programs, the applicable codes or standards, the standards of design, fabrication, erection, and performance to protect against environmental phenomena, the test procedures, and inspection acceptance levels of the reactor facility's essential components and systems. The influence of the sharing of common reactor facility components and systems along with the fire and explosion protection for all equipment is also to be established.

It is concluded that the design of this plant is in conformance with the criteria of Group 1 based on our interpretation of the intent of these criteria.

The text of each criterion, CECo's response, and references to applicable sections of the UFSAR and FSAR are given in the following for the criteria in this group.

3.1.1.1.1 Criterion 1 - Quality Standards

Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes or standards on design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary. Quality assurance programs, test procedures, and inspection acceptance levels to be used shall be identified. A showing of sufficiency and applicability of codes, standards, quality assurance programs, test procedures, and inspection acceptance levels is required.

Response

The reactor facility's essential components and systems are designed, fabricated, erected, and perform in accordance with the specified quality standards which are, as a minimum, in accordance with applicable codes and regulations. These components and systems, as well as applicable codes and standards, have been identified in the report. Specific sections are included in the reference list following this group's discussion. Where component or system design exceeds code requirements, it has been noted. A quality assurance program has been established to assure compliance with acceptable quality control specifications and procedures. These programs, as well as applicable tests and inspections, have been identified. Specific sections are included in the reference list. In planning and executing the quality assurance programs, particular attention is being given to the quality control specifications and to the compliance by those systems, components, and structures which are important to the plant safety.

Systems or Components	Sections Which Identify Applicable Codes or Standards
Reactor coolant system	Applicable FSAR Sections: 3.4, 3.6, 4.1, 4.2, 4.3, 4.4, and 12.1 Applicable UFSAR Sections: 3.9.3.1, 3.9.5.2, 3.9.5.3, 4.2.3.1, 5.1, 5.2.2.1, 5.2.2.2, 5.3.1.1, and 5.4.1
Containment system	Applicable FSAR Sections: 5.2 and 12.1 Applicable UFSAR Sections: 3.8.2.1, 6.2.1.1, 6.2.1.2, 6.2.1.3, 6.2.4.1, and 6.2.4.5
Emergency core cooling systems(ECCS)	Applicable FSAR Sections: 6.2 and 12.1 Applicable UFSAR Sections: 3.9.3.1 and 6.3
Diesel generator	Applicable FSAR Section: 12.1 Applicable UFSAR Sections: 8.3 and 9.5
Radwaste	Applicable FSAR Sections: 1.2, 1.3, 1.5, 9.1, 9.2, 9.3, 9.4, 9.5, and 12.1 Applicable UFSAR Sections: 1.2.1.6, 1.2.2.1, 1.2.2.13, 1.2.4.2, 11.2, 11.3, 11.4, and 12.3.2.1

3.1.1.1.2 Criterion 2 - Performance Standards

Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be designed, fabricated, and erected to performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces that might be imposed by natural phenomena such as earthquakes, tornadoes, flooding conditions, winds, ice, and other local site effects. The design bases so established shall reflect: (a) appropriate consideration of the most severe of these natural phenomena that have been recorded for the site and the surrounding area and (b) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design.

Response

The plant equipment which is important to safety is designed to permit safe plant operation and to accommodate all design basis accidents for all appropriate environmental phenomena at the site without loss of their capability, taking into consideration historical data and suitable margins for uncertainties.

Applicable FSAR Sections: 1.2, 1.3, 2.4, 2.5, 2.6, 2.7, 5.2, 5.3, and 12.1
Applicable UFSAR Sections: 1.2.1, 1.2.2, 3.4, 3.5, and 3.7

3.1.1.1.3 Criterion 3 - Fire Protection

The reactor facility shall be designed (1) to minimize the probability of events such as fires and explosions and (2) to minimize the potential effects of such events to safety. Noncombustible and fire resistant materials shall be used whenever practical through the facility, particularly in areas containing critical portions of the facility such as containment, control room, and components of engineered safety features.

Response

Design allowances are provided to minimize the occurrence of fire and explosions and their effects by the use of noncombustible and fire resistant materials throughout the plant.

Applicable FSAR Sections: 5.2, 5.3, and 10.7
Applicable UFSAR Section: 9.5.1

3.1.1.1.4 Criterion 4 - Sharing of Systems

Reactor facilities shall not share systems or components unless it is shown safety is not impaired by the sharing.

Response

The Dresden Units 1, 2, and 3 share some systems and components as specified in the sections listed below. This sharing does not result in undue risk to the health and safety of the public.

This reactor facility consists of two BWR generating units located on a common site. The design criteria and performance objectives for systems and components located on a single unit site are equally applicable to the systems and components shared between two units on a common site. Additional design criteria have been used in the design of Units 2 and 3. These stipulate that:

A. Equipment and facilities are shared only when it can be done without compromising or interfering with the independent operation of Units 2 and 3;

B. For unshared equipment, the equipment and its controls will be physically separated and identified;

C. Operation or safe shutdown of either Unit 2 or 3 will not be precluded as a result of reactor operator error or equipment malfunction in the other unit; and

D. Operation or safe shutdown of either Unit 2 or 3 after a postulated design basis accident in the other unit will not be precluded because of the shared equipment or facilities.

Applicable FSAR Sections: 1.5, 5.3, and 10.8
Applicable UFSAR Sections: 1.2.4 and 6.2.3

3.1.1.1.5 Criterion 5 - Record Requirements

Records of the design, fabrication, and construction of essential components of the plant shall be maintained by the reactor operator or under its control throughout the life of the reactor.

Response

Records of design, fabrication, and construction for this facility are to be stored or maintained either under the applicant's control or available to the applicant for inspection.

Applicable FSAR Sections: 4.2 and as required by the specific codes
Applicable UFSAR Section: 5.3

3.1.1.2 Group 2 - Protection by Multiple Fission Product Barriers

The intent of the proposed criteria for this group assures, through proper design, that the plant has been provided with multiple barriers against the release or mitigation of fission products to the environment and that these barriers will remain intact under all operational transients caused by a single operator error or equipment malfunction. It is the further intent of this group that proper barriers are made available for the design basis accidents.

It is concluded that design of this plant is in conformance with the criteria of Group 2 based on our interpretation of the intent of these criteria.

The text of each criterion, CECo's response, and reference to applicable sections of the UFSAR and FSAR are given in the following for the criteria in this group

3.1.1.2.1 Criterion 6 - Reactor Core Design

The reactor core shall be designed to function throughout its design lifetime, without exceeding acceptable fuel damage limits which have been stipulated and justified. The core design, together with reliable process and decay heat removal systems, shall provide for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and for transient situations which can be anticipated, including the effects of the loss of power to recirculation pumps, tripping out of a turbine generator set, isolation of the reactor from its primary heat sink, and loss of all offsite power.

Response

The reactor core is designed so that there is no inherent tendency for sudden divergent oscillation of operation characteristics or divergent power transients in any mode of plant operation. The basis of the reactor core design, in combination with the plant equipment characteristics, nuclear instrumentation system, and the reactor protection system, is to provide margins to ensure that fuel damage will not occur in normal operation or operational transients caused by single operator error or equipment malfunction.

Applicable FSAR Sections: 1.2, 1.3, 1.4, 3.2, 3.3, 3.4, 3.5, 4.3, 4.4, 6.2, 7.2, 7.3, 7.4, 7.5, and 11.2.3
Applicable UFSAR Sections: 1.2.1.1, 3.9.5, 4.2, 4.3, 4.4, 4.6, 7.6, 15.2.3, and 15.2.4

3.1.1.2.2 Criterion 7 - Suppression of Power Oscillations

The core design, together with reliable controls, shall ensure that power oscillations which could cause damage in excess of acceptable fuel damage limits are not possible or can be readily suppressed.

Response

The reactor core is designed so that there is no inherent tendency for sudden divergent oscillation of operation characteristics or divergent power transients in any mode of plant operation. The basis of the reactor core design, in combination with the plant equipment characteristics, nuclear instrumentation system, and the reactor protection system, is to provide margins to ensure that fuel damage will not occur in normal operation or operational transients caused by single operator error or equipment malfunction.

Applicable FSAR Sections: 3.2, 3.3, 3.4, 3.5, 7.2, 7.3, 7.4, and 7.5
Applicable UFSAR Sections: 4.2, 4.3, 4.4, 4.6, and 7.6

3.1.1.2.3 Criterion 8 - Overall Power Coefficient

The reactor shall be designed so that the overall power coefficient in the power operating range shall not be positive.

Response

The reactor core is designed so that there is no inherent tendency for sudden divergent oscillation of operation characteristics or divergent power transients in any mode of plant operation.

Applicable FSAR Sections: 3.1 and 3.3
Applicable UFSAR Sections: 4.1 and 4.3

3.1.1.2.4 Criterion 9 - Reactor Coolant Pressure Boundary

The reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime.

Response

The reactor coolant system is designed to carry its dead weight and specified live loads separately or concurrently, such as pressure and temperature stresses, vibrations, and seismic loads as prescribed for the plant. Provisions are made to control or shut down the reactor coolant system in the event of malfunction of operating or leakage of coolant from the system. The reactor vessel and support structures are designed, within the limits of applicable criteria for low probability accident conditions, to withstand the forces that would be created by the full area flow of any vessel nozzle to the containment atmosphere with the reactor vessel at design pressure concurrent with the plant maximum hypothetical earthquake loads.

Applicable FSAR Sections: 4.3 and 4.2
Applicable UFSAR Sections: 3.9, 5.2, and 5.3

3.1.1.2.5 Criterion 10 - Containment

Containment shall be provided. The containment structure shall be designed to sustain the initial effects of gross equipment failures, such as a large coolant boundary break, without loss of required integrity and, together with other engineered safety features as may be necessary, to retain for as long as the situation requires the functional capability to protect the public.

Response

The plant containment barriers are the basic features which minimize release of radioactive materials and associated doses. A boiling water reactor provides seven means of containing and/or mitigating the release of fission products:

A. The high density ceramic UO_2 fuel;

B. The high integrity Zircaloy cladding;

C. The reactor vessel and its connected piping and isolation valves;

D. The drywell-suppression chamber primary containment;

E. The reactor building (secondary containment);

F. The reactor building standby gas treatment system utilizing high efficiency absolute and charcoal filters; and

G. The main chimney.

The primary containment system is designed, fabricated, and erected to accommodate without failure, the pressures and temperatures resulting from or subsequent to the double-ended rupture or equivalent failure of any coolant pipe within the primary containment. The reactor building, encompassing the primary containment system, provides secondary containment when the primary containment is closed and in service, and provides for primary containment when the primary containment is open. The two containment systems and such other associated engineered safety systems as may be necessary are designed and maintained so that offsite doses resulting from postulated design basis accidents are below the values stated in 10 CFR 50.67.

Applicable FSAR Sections: 5.2, 5.3, 6.2.4, and 6.3
Applicable UFSAR Sections: 6.2.1, 6.2.2, and 6.2.3

3.1.1.3 Group 3 - Nuclear and Radiation Controls

The intent of the proposed criteria for this group is to identify and define the plant instrumentation and control systems, necessary for maintaining the plant in a safe operational status. This also includes determining the adequacy of radiation shielding, effluent monitoring, and fission process controls, and providing for the effective sensing of abnormal conditions and initiation of engineered safety features.

It is concluded that the design of this plant is in conformance with the criteria of Group 3 based on our interpretation of the intent of these criteria.

The text of each criterion, CECo's response, and reference to applicable sections of the UFSAR and FSAR are given in the following for the individual criteria in this group.

3.1.1.3.1 Criterion 11 - Control Room

The facility shall be provided with a control room from which actions to maintain safe operational status of the plant can be controlled. Adequate radiation protection shall be provided to permit access, even under accident conditions, to equipment in the control room or other areas as necessary to shut down and maintain safe control of the facility without radiation exposures of personnel in excess of 10 CFR 20 limits. It shall be possible to shut the reactor down and maintain it in a safe condition if access to the control room is lost due to fire or other cause.

Response

The plant is provided with a centralized control room having adequate shielding, fire protection, air conditioning, and facilities to permit access and continuous occupancy under 10 CFR 20 limits during all design basis accident situations. The plant design does not contemplate the necessity for evacuation of the control room. However, if it is necessary to evacuate the control room, the design does not preclude the capability to bring the plant to a safe, cold shutdown from outside the control room. The necessary plant controls, instrumentation, and alarms for safe and orderly operation are located in the control room. These include such controls as the control rod position indication, the reactor core heat removal system, and the reactor coolant system leakage detection system.

Applicable FSAR Sections: 1.5, 12.1, and 12.2
Applicable UFSAR Sections: 1.2.4.4, 3.8.4.2, 6.4, 7.4.2, and 12.3.2

3.1.1.3.2 Criterion 12 - Instrumentation and Control Systems

Instrumentation and controls shall be provided as required to monitor and maintain variables within prescribed operating ranges.

Response

The reactor protection system, independent from the plant process control systems, overrides all other controls to initiate any required safety action. The reactor protection system automatically initiates appropriate action whenever the plant conditions approach pre-established operational limits. The system acts specifically to initiate the core standby and containment cooling systems, as required. These include such controls as the control rod position indication, the reactor core heat removal system, and the reactor coolant system leakage detection system.

Applicable FSAR Sections: 1.2.4, 1.3.6, 4.2, Chapter 7, Chapter 8, 11.2.3, and 11.3.3
Applicable UFSAR Sections: 1.2.1.4, 1.2.2.6, Chapter 7, and Chapter 8

3.1.1.3.3 Criterion 13 - Fission Process Monitors and Controls

Means shall be provided for monitoring and maintaining control over the fission process throughout core life and for all conditions that can be reasonably be anticipated to cause variations in reactivity of the core, such as indication of position of control rods and concentration of soluble reactivity control poisons.

Response

The necessary plant controls, instrumentation, and alarms for safe and orderly operation are located in the control room. These include such controls as the control rod position indication, the reactor core heat removal system, and the reactor coolant system leakage detection system. The performance of the reactor core and the indication of reactor power level are continuously monitored by the nuclear instrumentation system. The reactor protection system, independent from the plant process control systems, overrides all other controls to initiate any required safety action. The reactor protection system automatically initiates appropriate action whenever the plant conditions approach pre-established operational limits. The system acts specifically to initiate the core standby and containment cooling systems, as required.

Applicable FSAR Sections: 1.2.4, 1.3.6, 6.7, 7.3, 7.4, and 7.9
Applicable UFSAR Sections: 1.2.2.10, 7.6.1, 7.7.1.2, and 7.7.3.1

3.1.1.3.4 Criterion 14 - Core Protection Systems

Core protection systems, together with associated equipment, shall be designed to act automatically to prevent or to suppress conditions that could result in exceeding acceptable fuel damage limits.

Response

The reactor protection system, independent from the plant process control systems, overrides all other controls to initiate any required safety action. The reactor protection system automatically initiates appropriate action whenever the plant conditions approach pre-established operational limits. The system acts specifically to initiate the core standby and containment cooling systems, as required.

Applicable FSAR Sections: 1.2.4, 1.3.6, 4.5, 6.5, 6.6, 7.3, 7.4, 7.5, 7.7, 7.8, and 7.9
Applicable UFSAR Sections: 1.2.1.4, 1.2.2.6, and 7.2

3.1.1.3.5 Criterion 15 - Engineered Safety Features Protection Systems

Protection systems shall be provided for sensing accident situations and initiating the operation of necessary engineered safety features.

Response

The reactor protection system, independent from the plant process control systems, overrides all other controls to initiate any required safety action. The reactor protection system automatically initiates appropriate action whenever the plant conditions approach pre-established operational limits. The system acts specifically to initiate the core standby and containment cooling systems, as required.

Applicable FSAR Section: 6.7.7
Applicable UFSAR Section: 7.2

3.1.1.3.6 Criterion 16 - Monitoring Reactor Coolant Pressure Boundary

Means shall be provided for monitoring the reactor coolant pressure boundary to detect leakage.

Response

The necessary plant controls, instrumentation, and alarms for safe and orderly operation are located in the control room. These include such controls as the control rod position indication, the reactor core heat removal system, and the reactor coolant system leakage detection system.

Applicable FSAR Sections: 4.2.4, 7.5.7, and 7.7
Applicable UFSAR Section: 5.2.5

3.1.1.3.7 Criterion 17 - Monitoring Radioactive Release

Means shall be provided for monitoring the containment atmosphere, the facility effluent discharge paths, and the facility environs for radioactivity that could be released from normal operations, from anticipated transients, and from accident conditions.

Response

The plant radiation and process monitoring systems are provided for monitoring significant parameters from specific plant process systems and specific areas including the plant effluent to the site environment and to provide alarms and signals for appropriate corrective actions.

Applicable FSAR Sections: 1.2.6, 5.3.2, 7.6, 9.2, 9.3, and 9.5
Applicable UFSAR Sections: 1.2.1.6, 11.2, 11.3, and 11.5

3.1.1.3.8 Criterion 18 - Monitoring Fuel and Waste Storage

Monitoring and alarm instrumentation shall be provided for fuel and waste storage and handling areas for conditions that might contribute to loss of continuity in decay heat removal and to radiation exposures.

Response

The plant radiation and process monitoring systems are provided for monitoring significant parameters from specific plant process systems and specific areas including the plant effluent to the site environment and to provide alarms and signals for appropriate corrective actions.

Applicable FSAR Section: 7.6
Applicable UFSAR Section: 11.5

3.1.1.4 Group 4 - Reliability and Testability of Protection

The intent of the proposed criteria for this group is to identify and establish the functional reliability, inservice testability, redundancy, physical and electrical independence and separation, and failsafe design of the reactor protection instrumentation and control systems.

It is concluded that the design of this plant is in conformance with the criteria of Group 4 based on our interpretation of the intent of these criteria.

The text of each criterion, CECo's response, and references to applicable sections of the UFSAR and FSAR are given in the following for the criteria in this group.

3.1.1.4.1 Criterion 19 - Protection Systems Reliability

Protection systems shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed.

Response

Components of the redundant subsystems can be removed from service for testing and maintenance without negating the ability of the protection system to perform its protection functions (even when subjected to a single-event, multiple-failure incident) upon receipt of the appropriate signals.

Applicable FSAR Sections: 1.2.4, 1.3.6, 7.3, 7.4, 7.5, 7.6, 7.7, 7.8, 7.9, and 7.10
Applicable UFSAR Sections: 1.2.1.4, 1.2.2.6, and 7.2

3.1.1.4.2 Criterion 20 - Protection Systems Redundancy and Independence

Redundancy and independence designed into protection systems shall be sufficient to assure that no single failure or removal from service of any component or channel of a system will result in loss of the protection function. The redundancy provided shall include, as a minimum, two channels of protection for each protection function to be served. Different principles shall be used where necessary to achieve true independence of redundant instrumentation components.

Response

By means of a dual channel protection system with complete redundancy in each channel, no loss of the protection systems can occur by either component failure or removal from service. The reactor protection system acts to shut down the reactor, close primary containment isolation valves, or initiates the operation of the core standby cooling systems. The reactor protection system is designed so that any design basis plant transient or accident is sensed by different parametric measurements (e.g., loss-of-coolant accident is detected by high drywell pressure and low reactor water level monitors). Components of the redundant subsystems can be removed from service for testing and maintenance without negating the ability of the protection system to perform its protection functions (even when subjected to a single-event, multiple-failure incident) upon receipt of the appropriate signals.

Applicable FSAR Sections: 1.2.4, 1.3.6, 7.3, 7.4, 7.5, 7.6, and 7.7
Applicable UFSAR Sections: 1.2.1.4, 1.2.2.6, and 7.2

3.1.1.4.3 Criterion 21 - Single Failure Definition

Multiple failures resulting from a single event shall be treated as a single failure.

Response

Components of the redundant subsystems can be removed from service for testing and maintenance without negating the ability of the protection system to perform its protection functions (even when subjected to a single-event, multiple-failure incident) upon receipt of the appropriate signals. Multiple failures from a single event are not counted as more than one failure.

3.1.1.4.4 Criterion 22 - Separation of Protection and Control Instrumentation Systems

Protection systems shall be separated from control instrumentation systems to the extent that failure or removal from service of any control instrumentation system component or channel, or of those common to control instrumentation and protection circuitry, leaves intact a system satisfying all requirements for the protection channels.

Response

The reactor protection system automatically overrides the plant normal operational control systems (that is, functions independently) to initiate appropriate action whenever the plant conditions monitored (neutron flux, containment, and vessel pressure, etc.) by the system approach pre-established limits.

Applicable FSAR Sections: 7.5 and 7.7
Applicable UFSAR Section: 7.2.2

3.1.1.4.5 Criterion 23 - Protection Against Multiple Disability for Protection Systems

The effects of adverse conditions to which redundant channels or protection systems might be exposed in common, either under normal conditions or those of an accident, shall not result in loss of the protection function.

Response

The system circuits are isolated to preclude a circuit fault from inducing a fault in another circuit and to reduce the likelihood that adverse conditions, which might affect system reliability (one-out-of-two-twice logic), will encompass more than one circuit. The system sensors are electrically and physically dispersed with both sensors in any one trip channel not allowed to occupy the same local area or to be connected to the same power source or process measurement line. The system internal wiring or external cable routing arrangements are such as to negate any external influence (a fire or accident) on the system's performance

Applicable FSAR Sections: 1.2.4, 1.3.6, 1.3.8, 5.2, 7.4, 7.5, 7.6, and 7.7
Applicable UFSAR Sections: 1.2.1.4, 1.2.2.6, and 7.2

3.1.1.4.6 Criterion 24 - Emergency Power for Protection Systems

In the event of loss of all offsite power, sufficient alternate sources of power shall be provided to permit the required functioning of the protection systems.

Response

The system electrical power requirements are supplied from independent, redundant sources. The system circuits are isolated to preclude a circuit fault from inducing a fault in another circuit and to reduce the likelihood that adverse conditions, which might affect system reliability (one-out-of-two-twice logic), will encompass more than one circuit. The system sensors are electrically and physically dispersed with both sensors in any one trip channel not allowed to occupy the same local area or to be connected to the same power source or process measurement line. The system internal wiring or external cable routing arrangements are such as to negate any external influence (a fire or accident) on the system's performance.

Applicable FSAR Sections: 1.2.5, 1.3.9, and Chapter 8
Applicable UFSAR Sections: 1.2.1.5, 1.2.2.10, 7.2.2.2, and 8.3.1.4

3.1.1.4.7 Criterion 25 - Demonstration of Functional Operability of Protection Systems

Means shall be included for testing protection systems while the reactor is in operation to demonstrate that no failure or loss of redundancy has occurred.

Response

The design of the reactor protection system is such as to facilitate maintenance and trouble shooting while the reactor is at power operation without impeding the plant's operation or impairing its safety function. System faults are annunciated in the main control room.

Applicable FSAR Sections: 1.2.4, 1.3.6, 7.4, 7.5, 7.6, and 7.7
Applicable UFSAR Sections: 1.2.1.4, 1.2.2.6, and 7.2

3.1.1.4.8 Criterion 26 - Protection Systems Fail-Safe Design

The protection systems shall be designed to fail into a safe state or into a state established as tolerable on a defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or adverse environments (e.g., extreme heat or cold, fire, steam, or water) are experienced.

Response

A failure of any one reactor protection system input or subsystem component will produce a trip in one of the two channels. This is a situation insufficient to produce a reactor scram, but the logic is readily available to perform its protective function upon another trip (either by failure or by exceeding the present trip).

Applicable FSAR Sections: 1.2.4, 1.3.6, 7.4, 7.6, 7.5, and 7.7
Applicable UFSAR Sections: 1.2.1.4, 1.2.2.6, and 7.2

3.1.1.5 Group 5 - Reactivity Control

The intent of the proposed criteria for this group is to establish the reactor core reactivity insertion and withdrawal rate limitations and the means to control the plant operations within these limits.

It is concluded that the design of this plant is in conformance with the criteria of Group 5 based on our interpretation of the intent of these criteria.

The text of each criterion, CECo's response, and references to applicable sections of the UFSAR and FSAR are given in the following for the criteria in this group.

3.1.1.5.1 Criterion 27 - Redundancy of Reactivity Control

At least two independent reactivity control systems, preferably of different principles, shall be provided.

Response

The plant design contains two independent and different principal reactivity control systems. Control of reactivity is operationally provided by a combination of movable control rods, fixed control devices or curtains (control curtains are no longer used; see Section 1.2.2.3), and reactor coolant recirculation system flow. These systems accommodate fuel burnup, load changes, and long-term reactivity changes. Reactor shutdown by this control rod drive system is sufficiently rapid to prevent violation of fuel damage limits for all operating transients. A standby liquid control system is provided as a redundant, independent shutdown system to cover emergencies in the operational reactivity control system discussed above. This standby system is designed to shut down the reactor in about 2 hours.

The reactor core is designed to have (a) a reactivity response which regulates or damps changes in power level, and spatial distributions of power production to a level consistent with safe and efficient operation, (b) a negative reactivity feedback consistent with the requirements of overall plant nuclear-hydrodynamic stability, and (c) a strong negative reactivity feedback under severe power transient conditions.

Applicable FSAR Sections: 1.2.1, 1.3.3, 3.5, and 6.7
Applicable UFSAR Sections: 1.2.1.1, 4.6, and 9.3.5

3.1.1.5.2 Criterion 28 - Reactivity Hot Shutdown Capability

At least two of the reactivity control systems provided shall independently be capable of making and holding the core subcritical from any hot standby or hot operating condition, including those resulting from power changes, sufficiently fast to prevent exceeding acceptable fuel limits.

Response

The plant design contains two independent and different principal reactivity control systems. Control of reactivity is operationally provided by a combination of movable control rods, fixed control devices or curtains, and reactor coolant recirculation system flow. These systems accommodate fuel burnup, load changes, and long-term reactivity changes. Reactor shutdown by this control rod drive system is sufficiently rapid to prevent violation of fuel damage limits for all operating transients. A standby liquid control system is provided as a redundant, independent shutdown system to cover emergencies in the operational reactivity control system discussed above. This standby system is designed to shut down the reactor in about 2 hours.

Applicable FSAR Sections: 1.2.1, 3.5.2, 3.5.3, and 6.7
Applicable UFSAR Sections: 1.2.1.1, 4.6, and 9.3.5

3.1.1.5.3 Criterion 29 - Reactivity Shutdown Capability

At least one of the reactivity control systems provided shall be capable of making the core subcritical under any condition (including anticipated operational transients) sufficiently fast to prevent exceeding acceptable fuel damage limits. Shutdown margins greater than the maximum worth of the most effective control rod when fully withdrawn shall be provided.

Response

The reactivity control system is designed such that, under conditions of normal operation, sufficient reactivity compensation is always available to make the reactor adequately subcritical from its most reactive condition. Means are provided for continuous regulation of the reactor core excess reactivity and reactivity distribution. This system is also designed to be capable of compensating for positive and negative reactivity changes resulting from changes in nuclear coefficients, fuel depletion, and fission product transients and buildup. The system design is such that control rod worths, and the rate at which reactivity can be added, are limited to assure that reactivity accidents cannot cause a transient capable of damaging the reactor coolant system, disrupt the reactor core or its support structures, or other vessel internals sufficiently to impair the core standby cooling system effectiveness if needed. Acceptable fuel damage limits will not be exceeded for any reactivity transient resulting from a single equipment malfunction or operator error.

Applicable FSAR Sections: 1.2.1, 1.3.3, 3.5, and 7.3
Applicable UFSAR Sections: 1.2.1.1, 4.6, and 7.7.2

3.1.1.5.4 Criterion 30 - Reactivity Holddown Capability

At least one of the reactivity control systems provided shall be capable of making and holding the core subcritical under any conditions with appropriate margins for contingencies.

Response

The reactivity control system is designed such that, under conditions of normal operation, sufficient reactivity compensation is always available to make the reactor adequately subcritical from its most reactive condition. Means are provided for continuous regulation of the reactor core excess reactivity and reactivity distribution.

Applicable FSAR Sections: 1.2.1, 1.3.3, 3.5, and 6.7
Applicable UFSAR Sections: 1.2.1.1, 4.6, and 9.3.5

3.1.1.5.5 Criterion 31 - Reactivity Control Systems Malfunction

The reactivity control systems shall be capable of sustaining any single malfunction, such as unplanned continuous withdrawal (not ejection) of a control rod, without causing a reactivity transient which could result in exceeding acceptable fuel damage limits.

Response

The reactor core is designed to have (a) a reactivity response which regulates or damps changes in power level, and spatial distributions of power production to a level consistent with safe and efficient operation, (b) a negative reactivity feedback consistent with the requirements of overall plant nuclear-hydrodynamic stability, and (c) a strong negative reactivity feedback under severe power transient conditions. The reactivity control system design is such that control rod worths, and the rate at which reactivity can be added, are limited to assure that reactivity accidents cannot cause a transient capable of damaging the reactor coolant system, disrupt the reactor core or its support structures, or other vessel internals sufficiently to impair the core standby cooling system effectiveness if needed. Acceptable fuel damage limits will not be exceeded for any reactivity transient resulting from a single equipment malfunction or operator error.

Applicable FSAR Sections: 3.2.8, 3.5, 4.3.3, 6.7, and 7.3
Applicable UFSAR Section: 4.6

3.1.1.5.6 Criterion 32 - Maximum Reactivity Worth of Control Rods

Limits, which include considerable margin, shall be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary (b) disrupt the core, its support structures, or other vessel internals sufficiently to impair the effectiveness of emergency core cooling.

Response

The reactivity control system design is such that control rod worths, and the rate at which reactivity can be added, are limited to assure that reactivity accidents cannot cause a transient capable of damaging the reactor coolant system, disrupt the reactor core or its support structures, or other vessel internals sufficiently to impair the core standby cooling system effectiveness if needed. Acceptable fuel damage limits will not be exceeded for any reactivity transient resulting from a single equipment malfunction or operator error.

Applicable FSAR Sections: 1.2.1, 3.3.4.4, 6.5, and 6.6
Applicable UFSAR Sections: 4.3.2.1, 4.6, and 7.7.1.2

3.1.1.6 Group 6 - Reactor Coolant Pressure Boundary

The intent of the proposed criteria for this group is to establish the reactor coolant pressure boundary design requirements and to identify the means used to satisfy these design requirements.

It is concluded that the design of this plant is in conformance with the criteria of Group 6 based on our interpretation of the intent of these criteria.

The text of each criterion, CECo's response, and references to applicable sections of the UFSAR and FSAR are given in the following for the criteria in this group.

3.1.1.6.1 Criterion 33 - Reactor Coolant Pressure Boundary Capability

The reactor coolant pressure boundary shall be capable of accommodating without rupture, and with only limited allowance for energy absorption through plastic deformation, the static and dynamic loads imposed on any boundary component as a result of any inadvertent and sudden release of energy to the coolant. As a design reference, this sudden release shall be taken as that which would result from a sudden reactivity insertion such as a rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition.

Response

The inherent safety features of the reactor core design in combination with certain engineered safety features (control rod velocity limiter, control rod housing support and the plant reactivity control system) are such that the consequences of the most severe potential nuclear excursion accident, caused by a single component failure within the reactivity control system (rod drop accident) would not result in damage (either by motion or rupture) to the reactor coolant system.

Applicable FSAR Sections: 3.3.4.4, 6.5, 6.6, and 11.3.3
Applicable UFSAR Section: 4.6

3.1.1.6.2 Criterion 34 - Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention

The reactor coolant pressure boundary shall be designed to minimize the probability of rapidly propagating type failures. Consideration shall be given (a) to the notch-toughness properties of materials extending to the upper shelf of the Charpy transition curve, (b) to the state of stress of materials under static and transient loadings, (c) to the quality control specified for materials and component fabrication to limit flaw sizes, and (d) to the provisions for control over service temperature and irradiation effects which may require operational restrictions.

Response

The ASME and USAS Codes are used as the established and acceptable criteria for design, fabrication, and operation of components of the reactor primary pressure system. The reactor primary system is designed and fabricated to meet the following as a minimum:

1. Reactor Vessel - ASME Boiler and Pressure Vessel Code, Section III Nuclear Vessels, Subsection A

2. Pumps - ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels, Subsection C

3. Piping and Valves - USAS-B-31.1, Code for Pressure Power Piping

Applicable FSAR Sections: 4.2 and 4.3
Applicable UFSAR Sections: 5.2.3 and 5.3

3.1.1.6.3 Criterion 35 - Reactor Coolant Pressure Boundary Brittle Fracture Prevention

Under conditions where reactor coolant pressure boundary system components constructed of ferritic materials may be subjected to potential loadings such as a reactivity-induced loading, service temperatures shall be at least 120°F above the nil ductility transition (NDT) temperature of the component material if the resulting energy release is expected to be absorbed by plastic deformation of 60°F above the NDT temperature of the component material if the resulting energy release is expected to be absorbed within the elastic strain energy range.

Response

The brittle fracture failure mode of the reactor coolant pressure boundary system components is prevented by control of the notch toughness properties of the ferritic steel components. This control is exercised in the selection of materials and fabrication of equipment and the components. In the design, appropriate consideration is given to the different notch toughness requirements of each of the various ferritic steel product forms, including weld and heat-affected zones. In this way, assurance is provided that brittle fracture is prevented under all potential service loading temperatures. The applicant's selected approach to brittle fracture prevention is to use a temperature-based rule with modifications drawn from fracture mechanics technology. The approach, which is generally accepted by materials specialists, establishes the requirements for brittle fracture prevention. These requirements are less stringent, when measured in terms of NDT requirement, for thin section materials than thick sections compared to that assumed in the criterion.

Applicable FSAR Sections: 4.2 and 4.3
Applicable UFSAR Sections: 5.2.3 and 5.3

3.1.1.6.4 Criterion 36 - Reactor Coolant Pressure Boundary Surveillance

Reactor coolant pressure boundary components shall have provisions for inspection, testing, and surveillance by appropriate means to assess the structural and leaktight integrity of the boundary components during their service lifetime. For the reactor vessel, a material surveillance program conforming with ASTM-E-185-66 shall be provided.

Response

The reactor coolant system was given a final hydrostatic test at 1500 psig in accordance with Code requirements prior to initial reactor startup. A hydrostatic test, not exceeding system operating pressure, will be made on the reactor coolant system following each removal and replacement of the reactor vessel head. The system was checked for leaks, and abnormal conditions were corrected before reactor startup. The minimum vessel temperature during hydrostatic test shall at least be 60°F above the calculated NDT temperature prior to pressurizing the vessel. Extensive quality control assurance programs were also followed during the entire fabrication of the reactor coolant system. Vessel material surveillance samples are located within the reactor primary vessel to enable periodic monitoring of material properties with exposure. The program includes specimens of the base metal, heat affected zone metal and standards specimens. Leakage from the reactor coolant system is monitored during reactor operation.

Applicable FSAR Sections: 4.2.4 and 4.3.4
Applicable UFSAR Sections: 5.2.3, 5.2.4, and 5.3.3

3.1.1.7 Group 7 - Engineered Safety Features

The intent of the proposed criteria for this group is:

A. To identify the engineered safety features (ESF);

B. To examine each ESF for independence, redundancy, capability, testability, inspectability, and reliability;

C. To determine the suitability of each ESF for its intended duty; and

D. Justify that each ESF's capability-scope envelopes all the anticipated and credible phenomena associated with the plant operational transients or design basis accidents being considered.

It is concluded that the design of this plant is in conformance with the criteria of Group 7 based on our interpretation of the intent of these criteria.

The text of each criterion, CECo's response, and references to applicable sections of the UFSAR and FSAR are given in the following for the criteria in this group.

3.1.1.7.1 Criterion 37 - Engineered Safety Features Basis for Design

Engineered safety features shall be provided in the facility to back up the safety provided by the core design, the reactor coolant pressure boundary, and their protection systems. As a minimum, such engineered safety features shall be designed to cope with any size reactor coolant pressure boundary break up to and including the circumferential rupture of any pipe in that boundary assuming unobstructed discharge from both ends.

Response

The normal plant control systems are thoroughly engineered and backed up by a significant amount of experience in system design and operation. Even if an improbable maloperation or equipment failure occurs including a reactor coolant boundary break (up to an including the circumferential rupture of any pipe in that boundary assuming an unobstructed discharge) and variables exceed their operating limits, an extensive system of ESFs limits the transient and the effects to levels well below those which are of public safety concern. These ESFs include:

A. The normal protection systems (reactor core, reactor coolant system, plant containment systems, plant and reactor control systems, reactor protection system, other instrumentation and process systems, etc.);

B. Those systems which offer additional protection against a reactivity excursion (reactor standby liquid control system, control rod velocity limiters, and control rod housing supports);

C. Those systems which act to reduce the consequences of design basis accidents (main steam line flow restrictors, primary containment atmospheric control system); and

D. Those systems which provide core standby and containment cooling in the event of a loss of normal cooling (core spray cooling system, low pressure coolant injection system [LPCI], isolation condenser, automatic depressurization system, and the standby coolant supply system).

Applicable FSAR Sections: 1.2, 1.3, Chapter 5, Chapter 6, Chapter 8, 10.13, and 10.14
Applicable UFSAR Sections: 1.2.1, 1.2.2, and 6.0

3.1.1.7.2 Criterion 38 - Reliability and Testability of ESFs

All engineered safety features shall be designed to provide high functional reliability and ready testability. In determining the suitability of a facility for a proposed site, the degree of reliance upon and acceptance of the inherent and engineered safety afforded by the systems, including engineered safety features, will be influenced by the known and demonstrated performance capability and reliability of the systems, and by the extent to which the operability of such systems can be tested and inspected where appropriate during the life of the plant.

Response

The ESFs are designed to provide high reliability and ready testability. Specific provisions are made in each ESF to demonstrate operability and performance capabilities.

Applicable FSAR Sections: 1.2.2, 1.2.3, 1.2.5, 1.3.4, 1.3.5, 1.3.9, 1.4, 1.5.4.6, 1.6, Chapter 5, Chapter 6, Chapter 8, 10.13, and 10.14
Applicable UFSAR Sections: 6.2.3.4, 6.2.6, 6.3.4, 6.5.3.4, 9.2.8.4, and 9.3.5.4

3.1.1.7.3 Criterion 39 - Emergency Power for ESFs

Alternate power systems shall be provided and designed with adequate independence, redundancy, capacity, and testability to permit the functioning required of the engineered safety features. As a minimum, the onsite power system and the offsite power system shall each, independently, provide this capacity assuming a failure of a single active component in each power system.

Response

Sufficient offsite and standby (redundant, independent, and testable) auxiliary sources of electrical power are provided to attain prompt shutdown and continued maintenance of the plant in a safe condition under all credible circumstances. The capacity of the power sources are adequate to accomplish all required ESF functions under all postulated design basis accident conditions.

Applicable FSAR Sections: 1.2.5, 1.3.5, 1.3.9, 1.4, 1.5.3, 6.2.7.2, Chapter 8, 10.13, and 10.14
Applicable UFSAR Sections: 1.2.4.3, 6.3.2, and 8.3

3.1.1.7.4 Criterion 40 - Missile Protection

Protection for engineered safety features shall be provided against dynamic effects and missiles that might result from plant equipment failures.

Response

Components of the ESFs which are required to function after design basis accidents or incidents are designed to withstand the most severe forces and environmental effects, including missiles from plant equipment failures anticipated from the events, without impairment of performance capability and without accentuating adverse aftereffects of the accident.

Applicable FSAR Sections: 5.2.3.7, 6.2.3.2, 6.2.4.2, and 6.2.5.2
Applicable UFSAR Sections: 3.5.2 and 6.3.2.2

3.1.1.7.5 Criterion 41 - ESF Performance Capability

Engineered safety features such as emergency core cooling and containment heat removal systems shall provide sufficient performance capability to accommodate partial loss of installed capacity and still fulfill the required safety function. As a minimum, each engineered safety feature shall provide this required safety function assuming a failure of a single active component.

Response

The ECCS is designed so that at least two different ECCS of different phenomena are provided to prevent clad melt over the entire spectrum of postulated design basis reactor primary system breaks. Such capability is available concurrently with the loss of all offsite ac power. The ECCS individual systems themselves are designed to various levels of component redundancy such that no single active component failure in addition to the accident will negate the required emergency core cooling capability.

Applicable FSAR Sections: 5.1, 5.2.2, 5.3.2, Chapter 6, Chapter 8, 10.13, and 10.14
Applicable UFSAR Sections: 6.2.4 and 6.3.3
3.1.1.7.6 Criterion 42 - ESF Components Capability

Engineered safety features shall be designed so that the capability of each component and system to perform its required function is not impaired by the effects of a loss-of-coolant accident.

Response

Components of ESFs which are required to function after design basis accidents or incidents are designed to withstand the most severe forces and environmental effects, including missiles from plant equipment failures anticipated from the events, without impairment of performance capability and without accentuating adverse aftereffects of the accident.

Applicable FSAR Sections: 1.2.1, 5.2.3.7, and Chapter 6
Applicable UFSAR Sections: 6.2.1, 6.2.2, 6.2.3, 6.2.4, and 6.3.3

3.1.1.7.7 Criterion 43 - Accident Aggravation Prevention

Engineered safety features shall be designed so that any action of the engineered safety features which might accentuate the adverse aftereffects of the loss of normal cooling is avoided.

Response

Components of the ESF which are required to function after design basis accidents or incidents are designed to withstand the most severe forces and environmental effects, including missiles from plant equipment failures anticipated from the events, without impairment of performance capability and without accentuating adverse aftereffects of the accident.

Applicable FSAR Sections: 6.2.3.2, 6.2.3.5, 6.2.4.2, 6.2.5.2, 6.2.5.3, 6.2.5.4, and 6.3
Applicable UFSAR Sections: 6.3.2.1, 6.3.2.2, 6.3.2.3, and 6.3.3.1

3.1.1.7.8 Criterion 44 - Emergency Core Cooling System Capability

At least two emergency core cooling systems, preferably of different design principles, each with a capability for accomplishing abundant emergency core cooling, shall be provided. Each emergency core cooling system and the core shall be designed to prevent fuel and clad damage that would interfere with the emergency core cooling function and to limit the clad metal-water reaction to negligible amounts for all sizes of breaks in the reactor coolant pressure boundary, including the double-ended rupture of the largest pipe. The performance of each emergency core cooling system shall be evaluated conservatively in each area of uncertainty. The systems shall not share active components and shall not share other features or components unless it can be demonstrated that (a) the capability of the shared feature or component to perform its required function can be readily ascertained during reactor operation, (b) failure of the shared feature or component does not initiate a loss-of-coolant accident, and (c) capability of the shared feature or component to perform its required function is not impaired by the effects of a loss-of-coolant accident and is not lost during the entire period this function is required following the accident.

Response

The ECCS is designed so that at least two different ECCS of different phenomena are provided to prevent clad melt over the entire spectrum of postulated design basis reactor primary system breaks. Such capability is available concurrently with the loss of all offsite ac power. The ECCS individual systems themselves are designed to various levels of component redundancy such that no single active component failure in addition to the accident will negate the required emergency core cooling capability.

Applicable FSAR Sections: 5.2.3.4, 6.2.1, 6.2.4.2, 6.2.5.1, and 6.8
Applicable UFSAR Sections: 6.2.1.3 and 6.3.1

3.1.1.7.9 Criterion 45 - Inspection of Emergency Core Cooling Systems

Design provisions shall be made to facilitate physical inspection of all critical parts of the emergency core cooling systems, including reactor vessel internals and water injection nozzles.

Response

Design provisions have also been made to enable physical and visual inspection of the ECCS components.

Applicable FSAR Sections: 6.2.3.5, 6.2.4.4, and 6.2.5.4
Applicable UFSAR Sections: 6.3.4.1, 6.3.4.2, 6.3.4.3, and 6.3.4.4

3.1.1.7.10 Criterion 46 - Testing of ECCS Components

Design provisions shall be made so that active components of the emergency core cooling systems, such as pumps and valves, can be tested periodically for operability and required functional performance.

Response

To assure that the ECCS will function properly, if needed, specific provisions have been made for testing the sequential operability and functional performance of each individual system.

Applicable FSAR Sections: 6.2.3.5, 6.2.4.4, and 6.2.5.4
Applicable UFSAR Sections: 6.3.4.1, 6.3.4.2, 6.3.4.3, and 6.3.4.4

3.1.1.7.11 Criterion 47 - Testing of ECCS Criterion

A capability shall be provided to test periodically the delivery capability of the emergency core cooling systems at a location as close to the core as is practical.

Response

To assure that the ECCS will function properly, if needed, specific provisions have been made for testing the sequential operability and functional performance of each individual system.

Applicable FSAR Sections: 6.2.3.5, 6.2.4.4, and 6.2.5.4
Applicable UFSAR Sections: 6.3.4.1, 6.3.4.2, 6.3.4.3, and 6.3.4.4

3.1.1.7.12 Criterion 48 - Testing of Operational Sequence of ECCS

A capability shall be provided to test, under conditions as close to design as practical, the full operational sequence that would bring the emergency core cooling systems into action, including the transfer to alternate power sources.

Response

To assure that the ECCS will function properly, if needed, specific provisions have been made for testing the sequential operability and functional performance of each individual system.

Applicable FSAR Sections: 6.2.3.5, 6.2.4.4, 6.2.5.4, and 8.3.1
Applicable UFSAR Sections: 6.3.4.1, 6.3.4.2, 6.3.4.3, and 6.3.4.4

3.1.1.7.13 Criterion 49 - Containment Design Basis

The containment structure, including access openings and penetrations and any necessary containment heat removal systems, shall be designed so that the containment structure can accommodate without exceeding the design leakage rate the pressures and temperatures resulting from the largest credible energy release following a loss-of-coolant accident, including a considerable margin for effects from metal-water or other chemical reactions that could occur as a consequence of failure of emergency core cooling systems.

Response

The primary containment structure, including access openings and penetrations is designed to withstand the peak transient pressure and temperatures which could occur due to the postulated design basis loss of coolant accident. The containment design includes considerable allowance for energy addition, including from metal-water or other chemical reactions beyond conditions that would occur with normal operation of emergency core cooling systems (ECCS).

Applicable FSAR Sections: 1.2.3 and 5.2.3.4
Applicable UFSAR Sections: 1.2.1.3, 6.2.1.1, and 6.2.1.3

3.1.1.7.14 Criterion 50 - NDT Requirement for Containment Material

Principal load carrying components of ferritic materials exposed to the external environment shall be selected so that their temperatures under normal operating and testing conditions are not less than 30°F above NDT temperature.

Response

Plates, structural members, forgings, and pipe associated with the drywell have an initial NDT temperature of approximately 0°F when tested in accordance with the appropriate code for the materials. It is intended that the drywell will not be pressurized or subjected to substantial stress at temperatures below 30°F.

Provisions are made for the removal of heat from within the plant containment system and to isolate the various process system lines as may be necessary to maintain the integrity of the plant containment systems as long as necessary following the various postulated design basis accidents. The integrity of the complete plant containment is designed and maintained so that the offsite doses resulting from postulated design basis accidents will be below the values stated in 10 CFR 50.67.

Applicable FSAR Sections: 5.2.3 and 5.2.3.1
Applicable UFSAR Section: 6.2.1.3

3.1.1.7.15 Criterion 51 - Reactor Coolant Pressure Boundary Outside Containment

If part of the reactor coolant pressure boundary is outside the containment, appropriate features as necessary shall be provided to protect the health and safety of the public in case of an accidental rupture in that part. Determination of the appropriateness of features such as isolation valves and additional containment shall include consideration of the environmental and population conditions surrounding the site.

Response

Plates, structural members, forgings, and pipe associated with the drywell have an initial NDT temperature of approximately 0°F when tested in accordance with the appropriate code for the materials. It is intended that the drywell will not be pressurized or subjected to substantial stress at temperatures below 30°F. Provisions are made for the removal of heat from within the plant containment system and to isolate the various process system lines as may be necessary to maintain the integrity of the plant containment systems as long as necessary following the various postulated design basis accidents. The integrity of the complete plant containment is designed and maintained so that the offsite doses resulting from postulated design basis accidents will be below the values stated in 10 CFR 50.67.

Applicable FSAR Sections: 1.2.3, 5.2.2, 5.3, and 6.4
Applicable UFSAR Sections: 1.2.1.3, 5.2, 6.2.1.2, and 6.2.3

3.1.1.7.16 Criterion 52 - Containment Heat Removal Systems

Where active heat removal systems are needed under accident conditions to prevent exceeding containment design pressure, at least two systems, preferably of different principles, each with full capacity, shall be provided.

Response

The pressure suppression concept phenomena and the containment spray cooling system provide two different means to rapidly condense the steam portion of the flow from the postulated design basis loss-of-coolant accident so that the peak transient pressure would be substantially less than the primary containment design pressure.

Applicable FSAR Sections: 1.2.3, 5.2.1, 5.2.2, 5.3.2, and 6.2.4.2
Applicable UFSAR Sections: 1.2.1.3, 6.2.1.1, and 6.2.2

3.1.1.7.17 Criterion 53 - Containment Isolation Valves

Penetrations that require closure for the containment function shall be protected by redundant valving and associated apparatus.

Response

All pipes or ducts which penetrate the primary containment and which connect to the reactor coolant system or to the drywell are provided with at least two isolation valves in series.

Applicable FSAR Sections: 5.2.2, 5.2.4, 5.3.2, and 7.6.2
Applicable UFSAR Section: 6.2.4.1

3.1.1.7.18 Criterion 54 - Containment Leakage Rate Testing

Containment shall be designed so that an integrated leakage rate testing can be conducted at design pressure after completion and installation of all penetrations and the leakage rate measured over a sufficient period of time to verify its conformance with required performance.

Response

Plates, structural members, forgings, and pipe associated with the drywell have an initial NDT temperature of approximately 0°F when tested in accordance with the appropriate code for the materials. It is intended that the drywell will not be pressurized or subjected to substantial stress at temperatures below 30°F. Provisions are made for the removal of heat from within the plant containment system and to isolate the various process system lines as may be necessary to maintain the integrity of the plant containment systems as long as necessary following the various postulated design basis accidents. The integrity of the complete plant containment is designed and maintained so that the offsite doses resulting from postulated design basis accidents will be below the values stated in 10 CFR 50.67. The plant design includes preoperational pressure and leak rate testing of the primary containment system, and a capability for leak testing at design pressure after the plant has commenced operation.

Applicable FSAR Sections: 1.2.3, 5.2.4, and 5.3.4
Applicable UFSAR Sections: 1.2.1.3, 6.2.3.4, and 6.2.6

3.1.1.7.19 Criterion 55 - Containment Periodic Leakage Rate Testing

The containment shall be designed so that integrated leakage rate testing can be done periodically at design pressure during plant lifetime.

Response

The plant design includes preoperational pressure and leak rate testing of the primary containment system, and a capability for leak testing at design pressure after the plant has commenced operation. Periodic tests during the lifetime of the units are made at a pressure which permits extrapolation of results to the design accident pressure condition, using relationships established initially for comparative leakage at these two conditions.

Applicable FSAR Sections: 1.2.3, 5.2.4.1, and 5.3.4
Applicable UFSAR Sections: 1.2.1.3, 6.2.3.4, and 6.2.6

3.1.1.7.20 Criterion 56 - Provisions for Testing of Penetrations

Provisions shall be made for testing penetrations which have resilient seals or expansion bellows to permit leak-tightness to be demonstrated at design pressure at any time.

Response

Provisions are made for demonstrating the functional performance of the plant containment system isolation valves and leak testing of selected penetrations.

Applicable FSAR Section: 5.2.4.2
Applicable UFSAR Section: 6.2.6.2

3.1.1.7.21 Criterion 57 - Provisions for Testing of Isolation Valves

Capability shall be provided for testing functional operability of valves and associated apparatus essential to the containment function for establishing that no failure has occurred and for determining that valve leakage does not exceed acceptable limits.

Response

Provisions are made for demonstrating the functional performance of the plant containment system isolation valves and leak testing of selected penetrations.

Applicable FSAR Section: 5.2.4.3
Applicable UFSAR Section: 6.2.6.3

3.1.1.7.22 Criterion 58 - Inspection of Containment Pressure-Reducing Systems

Design provisions shall be made to facilitate the periodic physical inspection of all important components of the containment pressure-reducing systems, such as pumps, valves, spray nozzles, torus, and sumps.

Response

Demonstration of operability and ability to test the functional performance and inspect the containment spray/cooling system is provided.

Applicable FSAR Sections: 5.2.4, 6.2.4.4, and 6.2.4.1
Applicable UFSAR Section: 6.2.2.4

3.1.1.7.23 Criterion 59 - Testing of Containment Pressure-Reducing System Components

The containment pressure-reducing systems shall be designed so that active components, such as pumps and valves, can be tested periodically for operability and required functional performance.

Response

Demonstration of operability and ability to test the functional performance and inspect the containment spray/cooling system is provided.

Applicable FSAR Section: 6.2.4.4
Applicable UFSAR Section: 6.2.2.4

3.1.1.7.24 Criterion 60 - Testing of Containment Spray Systems

A capability shall be provided to test periodically the delivery capability of the containment spray system at a position as close to the spray nozzles as is practical.

Response

Demonstration of operability and ability to test the functional performance and inspect the containment spray/cooling system is provided.

Applicable FSAR Section: 6.2.4.4
Applicable UFSAR Section: 6.2.2.4

3.1.1.7.25 Criterion 61 - Testing of Operational Sequence of Containment Pressure-Reducing Systems

A capability shall be provided to test, under conditions as close to the design as practical, the full operational sequence that would bring the containment pressure-reducing systems into action, including the transfer to alternate power sources.

Response

Demonstration of operability and ability to test the functional performance and inspect the containment spray/cooling system is provided.

Applicable FSAR Sections: 5.2 and 6.2.8
Applicable UFSAR Section: 6.2.2.4

3.1.1.7.26 Criterion 62 - Inspection of Air Cleanup Systems

Design provisions shall be made to facilitate physical inspection of all critical parts of containment air cleanup systems, such as ducts, filters, fans, and dampers.

Response

The standby gas treatment system may be physically inspected and its operability demonstrated.

Applicable FSAR Sections: 5.3.2, 5.3.4, and 10.11
Applicable UFSAR Section: 6.5.3.4

3.1.1.7.27 Criterion 63 - Testing of Air Cleanup Systems Components

Design provisions shall be made so that active components of the air cleanup systems, such as fans and dampers, can be tested periodically for operability and required functional performance.

Response

The standby gas treatment system may be physically inspected and its operability demonstrated.

Applicable FSAR Sections: 5.3.2, 5.3.4, and 10.11
Applicable UFSAR Section: 6.5.3.4

3.1.1.7.28 Criterion 64 - Testing of Air Cleanup Systems

A capability shall be provided for in situ periodic testing and surveillance of the air cleanup systems to ensure (a) filter bypass paths have not developed and (b) filter and trapping materials have not deteriorated beyond acceptable limits.

Response

The secondary containment-standby gas treatment system is designed such that means are provided for periodic testing of the system performance including tracer injection and sampling.

Applicable FSAR Sections: 5.3.2, 5.3.4, and 10.11
Applicable UFSAR Section: 6.5.3.4

3.1.1.7.29 Criterion 65 - Testing of Operational Sequence of Air Cleanup Systems

A capability shall be provided to test, under conditions as close to design as practical, the full operational sequence that would bring the air cleanup systems into action, including the transfer to alternate power sources and the design air flow delivery capability.

Response

The standby gas treatment system may be physically inspected and its operability demonstrated.

Applicable FSAR Sections: 5.3.2, 5.3.4, 7.6.2.6, and 10.11
Applicable UFSAR Sections: 6.2.3.2 and 6.5.3.4

3.1.1.8 Group 8 - Fuel and Waste Storage Systems

The intent of the proposed criteria for this group is to establish the safety of fuel and waste storage systems design and to identify the means used to satisfy these design requirements.

It is concluded that the design of this plant is in conformance with criteria of Group 8 based on our interpretation of the intent of these criteria.

The text of each criterion, CECo's response, and references to applicable sections of the UFSAR and FSAR are given in the following for criteria in this group.

3.1.1.8.1 Criterion 66 - Prevention of Fuel Storage Criticality

Criticality in new and spent fuel storage shall be prevented by physical systems or processes. Such means as geometrically safe configurations shall be emphasized over procedural controls.

Response

Appropriate plant fuel handling and storage facilities are provided to preclude accidental criticality and to provide sufficient cooling for spent fuel. The new fuel storage vault racks (located inside the secondary containment reactor building) are top entry, and are designed to prevent an accidental critical array, even in the event the vault becomes flooded. Vault drainage is provided to prevent possible water collection. With the exception of spent fuel stored in Dry Cask Storage (DCS) systems as discussed in Section 9.1.2.2.4, the handling and storage of spent fuel, which takes place entirely within the reactor building (which provides containment), is done in the spent fuel storage pool. The pool has provisions to maintain water clarity, temperature control and instrumentation to monitor water level. Water depth in the pool will be such as to provide sufficient shielding for normal reactor building occupancy (10 CFR 20) by operating personnel. The racks in which spent fuel assemblies are placed are designed and arranged to insure subcriticality in the storage pool. The spent fuel pool cooling and demineralizer system is designed to maintain the pool water temperature (decay heat removal) to control water clarity (safe fuel movement), and to reduce water radioactivity (shielding and effluent release control).

Applicable FSAR Sections: 1.2.8, 1.3.7, and 10.1
Applicable UFSAR Sections: 1.2.1.8, 9.1.1, and 9.1.2

3.1.1.8.2 Criterion 67 - Fuel and Waste Storage Decay Heat

Reliable decay heat removal systems shall be designed to prevent damage to the fuel in storage facilities that could result in radioactivity release to plant operating areas or the public environs.

Response

Appropriate plant fuel handling and storage facilities are provided to preclude accidental criticality and to provide sufficient cooling for spent fuel. With the exception of spent fuel stored in Dry Cask Storage (DCS) systems as discussed in Section 9.1.2.2.4, the handling and storage of spent fuel, which takes place entirely within the reactor building (which provides containment), is done in the spent fuel storage pool. The pool has provisions to maintain water clarity, temperature control and instrumentation to monitor water level. Water depth in the pool will be such as to provide sufficient shielding for normal reactor building occupancy (10 CFR 20) by operating personnel. The racks in which spent fuel assemblies are placed are designed and arranged to insure subcriticality in the storage pool. The spent fuel pool cooling and demineralizer system is designed to maintain the pool water temperature (decay heat removal) to control water clarity (safe fuel movement), and to reduce water radioactivity (shielding and effluent release control).

Applicable FSAR Sections: 1.2.8 and 10.1
Applicable UFSAR Sections: 1.2.1.8 and 9.1.3

3.1.1.8.3 Criterion 68 - Fuel and Waste Storage Radiation Shielding

Shielding for radiation protection shall be provided in the design of spent fuel and waste storage facilities as required to meet the requirements of 10 CFR 20.

Response

With the exception of spent fuel stored in Dry Cask Storage (DCS) systems as discussed in Section 9.1.2.2.4, the handling and storage of spent fuel, which takes place entirely within the reactor building (which provides containment), is done in the spent fuel storage pool. Water depth in the pool will be such as to provide sufficient shielding for normal reactor building occupancy (10 CFR 20) by operating personnel. The racks in which spent fuel assemblies are placed are designed and arranged to insure subcriticality in the storage pool. The spent fuel pool cooling and demineralizer system is designed to maintain the pool water temperature (decay heat removal) to control water clarity (safe fuel movement), and to reduce water radioactivity (shielding and effluent release control). Accessible portions of the reactor and radwaste buildings have sufficient shielding to maintain dose rates within 10 CFR 20. The radwaste building is designed to preclude accidental release of radioactive materials to the environs.

Applicable FSAR Section: 10.1
Applicable UFSAR Sections: 9.1 and 12.3.2.1

3.1.1.8.4 Criterion 69 - Protection Against Radioactivity Release from Spent Fuel and Waste Storage

Containment of fuel and waste storage shall be provided if accidents could lead to release of undue amounts of radioactivity to the public environs.

Response

With the exception of spent fuel stored in Dry Cask Storage (DCS) systems as discussed in Section 9.1.2.2.4, the handling and storage of spent fuel, which takes place entirely within the reactor building (which provides containment), is done in the spent fuel storage pool. The pool has provisions to maintain water clarity, temperature control and instrumentation to monitor water level. Water depth in the pool will be such as to provide sufficient shielding for normal reactor building occupancy (10 CFR 20) by operating personnel. The racks in which spent fuel assemblies are placed are designed and arranged to insure subcriticality in the storage pool.

Applicable FSAR Sections: 1.2.3, 1.3.4.2, 5.3, Chapter 9, and 10.1
Applicable UFSAR Sections: 1.2.2.4, 6.2.3, and 9.1

3.1.1.9 Group 9 - Plant Effluents

The intent of the proposed criterion for this group is to establish the plant effluent release limits and to identify the means of controlling the release within these guide limits.

It is concluded that the design of this plant is in conformance with the criterion of Group 9 based on our interpretation of the intent of this criterion.

The text of the criterion, CECo's response, and references to applicable sections of the UFSAR and FSAR are given in the following for the criterion of this group.

3.1.1.9.1 Criterion 70 - Control of Release of Radioactivity to the Environment

The facility design shall include those means necessary to maintain control over the plant radioactive effluents, whether gaseous, liquid, or solid. Appropriate holdup capacity shall be provided for retention of gaseous, liquid, or solid effluents, particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment. In all cases, the design for radioactivity control shall be justified (a) on the basis of 10 CFR 20 requirements for normal operations and for any transient situation that might reasonably be anticipated to occur and (b) on the basis of 10 CFR 100 or 10CFR 50.67 as applicable dosage level guidelines for potential reactor accidents of exceedingly low probability of occurrence except that reduction of the recommended dosage levels may be required where high population densities or very large cities can be affected by the radioactive effluents.

Response

The plant radioactive waste control system (which includes the liquid gaseous and solid radwaste subsystems) is designed to limit the offsite radiation levels below those set forth in 10 CFR 20. The plant engineered safety systems (including the containment barriers) are designed to limit the offsite dose under various postulated design basis accidents to levels significantly below 10 CFR 100 or 10CFR 50.67 as applicable. The air ejector off-gas system is designed with sufficient holdup retention capacity so that during normal plant operation the controlled release of radioactive materials does not exceed the established release limits at the plant elevated stack.

Applicable FSAR Sections: 1.2.6, 1.3.11, 1.5.2, 5.2.1, 5.2.2, 5.3, 7.6, Chapter 9, and 10.11
Applicable UFSAR Sections: 1.2.1.3, 1.2.1.6, 1.2.2.12, 1.2.4.1, 1.2.4.2, and 11.5

3.1.2 Compliance with Final Design Criteria

This section contains an evaluation of the design basis of the Dresden Nuclear Power Station Unit 2 as measured against the NRC General Design Criteria (GDC) for Nuclear Power Plants, Appendix A of 10 CFR 50, effective May 21, 1971, and subsequently amended July 7, 1971. The General Design Criteria, which are divided into six groups, are intended to establish minimum requirement for the design of nuclear power plants.

It should be noted that the General Design Criteria were not written specifically for the boiling water reactor; rather, they were intended to guide the design of all water-cooled nuclear power plants. The criteria are generic in nature and have been subject to a variety of interpretations over the years. It should be noted that a significant number of guidance documents describe commonly implemented approaches to meeting these criteria. These guidance documents describe acceptable methods of conformance with the criteria, but do not describe the only method of conformance. Since much of this guidance was developed after Dresden was designed, it is inappropriate to infer that this conformance evaluation either creates commitments to or claims conformance to specific implementing guidance. The UFSAR sections applicable to individual systems, structures, or components generally note that specific commitments have been made or that conformance to specific guidance documents has been evaluated.

Based on the material contained in this application, CECo concluded that Dresden Station Unit 2 satisfies and is in compliance with the intent of the General Design Criteria. This evaluation was performed specifically for Unit 2 and may not fully apply to Unit 3; however, the high degree of similarity between the design of Units 2 and 3 indicates that Unit 3 also conforms to the intent of the General Design Criteria. Additional design information pertinent to each criterion is addressed in sections of the UFSAR describing various systems and structures.

3.1.2.1 Group 1 - Overall Requirements

3.1.2.1.1 Criterion 1 - Quality Standards and Records

Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

Response

Structures, systems, and components important to safety were designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed and in conformance with applicable codes and regulations identified in Section 3.8.

Quality assurance programs were implemented to assure that the structures, systems, and components satisfactorily comply with quality control specifications and procedures. In planning and executing the quality assurance programs, particular attention was given to the quality control specifications and their application to those structures, systems, and components important to safety.

Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety have been prepared and will be maintained throughout the life of the unit.

Suitable startup and operating tests of systems important to safety were performed to verify the adequacy of the design and compliance with applicable codes.

Programs have been established to ensure continued compliance with applicable governmental codes and regulations and to industry standards.

3.1.2.1.2 Criterion 2 - Design Bases for Protection Against Natural Phenomena

Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated; (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena; and (3) the importance of the safety functions to be performed.

Response

Structures, systems, and components important to safety have been designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions.

Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, the combination of the effects of normal and accident conditions, and the importance of the safety functions to be performed have been reflected in the facility design. Margins relative to the accuracy, quantity, and time base of the historical information have been applied.

Investigation into the effects of flooding of the site as a result of postulated conditions with historical precedent has been made with the conclusion that the facility is adequately protected. Class I structures were designed to withstand tornado forces, the most severe natural phenomena likely to affect the site area; structures have been designed to withstand tornado effects in excess of those having historical precedents. Structures are capable of withstanding penetration from tornado-generated missiles.

3.1.2.1.3 Criterion 3 - Fire Protection

Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat-resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Fire fighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

Response

Structures, systems, and components important to safety have been designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions.

Noncombustible and heat-resistant materials have been used wherever practical.

Fire detecting systems and fire fighting systems of appropriate design and capability have been provided to minimize the adverse effects of fires on structures, systems, and components important to safety. Fire fighting systems are designed to ensure that their rupture or inadvertent operation does not significantly impair the safety capability of the structures, systems, and components.

3.1.2.1.4 Criterion 4 - Environmental and Missile Design Bases

Structures, systems, and components important to safety 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, including loss-of-coolant accidents. These structures, systems and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.

Response

Structures, systems, and components have been designed to accommodate the effects of, and be compatible with, the environmental conditions associated with normal operation, maintenance, testing and postulated accidents, including LOCAs.

Dynamic effects, including the effects of missiles, pipe whipping and discharging fluids resulting from equipment failures, postulated accidents, and natural phenomena have been evaluated and appropriately protected against.

Pipe restraints and stops have been installed where such devices do not interfere with inspection and maintenance of a more critical nature. Equipment has been installed to minimize the possibility of a single event impairing the operability of redundant systems and to prohibit the attainment of forces sufficient to cause a loss of integrity to the drywell or other vital structures or systems.

Leak detection systems, pumping, and storage facilities sufficient to preclude flooding of safety-related systems and components have been incorporated in the facility design.

Class I structures, systems, and components are capable of withstanding the effects of missiles originating outside the nuclear power unit.

3.1.2.1.5 Criterion 5 - Sharing of Structures, Systems, and Components

Structures, systems, and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown an cooldown of the remaining units

Response

Structures, systems, and components that are important to safety and are shared by the other CECo plants have been discussed in documents submitted for those facilities. In all cases, the sharing enhances the ability of the facilities to perform their safety functions under adverse conditions.

3.1.2.2 Group 2 - Protection by Multiple Fission Product Barriers

3.1.2.2.1 Criterion 10 - Reactor Design

The reactor core and associated coolant, control and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

Response

The reactor core and associated coolant, control and protection systems are designed with appropriate margins to ensure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

The reactor core components consist of fuel assemblies, control rods, incore ion chambers and related items. The mechanical design is based on the conservative application of stress limits, operating experience and test results. The fuel is designed to provide high integrity over a complete range of power levels including transient conditions.

The general design bases employed for the core thermal and hydraulic design, taken in conjunction with the facility equipment characteristics, nuclear instrumentation and the reactor protection system, are utilized to ensure that no fuel damage will occur during normal operation or anticipated transients caused by single operator errors or equipment malfunctions.

3.1.2.2.2 Criterion 11 - Reactor Inherent Protection

The reactor core and associated systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

Response

The reactor and associated coolant systems are designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tend to compensate for a rapid increase in reactivity and, in conjunction with control and protection systems, will prevent, detect, suppress or regulate any power oscillations which could result in conditions exceeding specified acceptable fuel design limits.

The Doppler and moderator void coefficients (power coefficient) provide the inherent negative feedback and stability enhancement in the systems.

3.1.2.2.3 Criterion 12 - Suppression of Reactor Power Oscillations

The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

Response

The reactor and associated coolant systems are designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tend to compensate for a rapid increase in reactivity and, in conjunction with control and protection systems, will prevent, detect, suppress or regulate any power oscillations which could result in conditions exceeding specified acceptable fuel design limits.

The Doppler and moderator void coefficients (power coefficient) provide the inherent negative feedback and stability enhancement in the systems.

Coolant systems, control, and protective systems are designed to minimize the introduction of oscillation during anticipated operating occurrences.

In addition, Oscillating Power Range Monitors (OPRMs) have been installed to detect thermal-hydraulic instabilities which result in core power oscillations. If the instabilities grow to a point where the power oscillations could result in a condition exceeding specified fuel design limits, the OPRMs will, when fully installed and armed, automatically generate a trip which scrams the reactor.

3.1.2.2.4 Criterion 13 - Instrumentation and Control

Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

Response

The fission process is monitored and controlled for all conditions from the source range through the power range. The neutron monitoring system detects core conditions that could potentially threaten the overall integrity of the fuel barrier due to excess power generation and provides a corresponding signal to the reactor protection system. Fission chambers, located in the core, are used to sense neutron flux from the source range through the power range. The detectors are located to provide maximum sensitivity to control rod movement during startup and to provide optimum monitoring in the intermediate and power ranges.

Also, the reactor protection system is provided to initiate appropriate action automatically whenever specific facility conditions reach established limits. The protection system functions are tabulated in the Technical Specifications. The system is designed to mitigate the consequences of facility normal and accident transients, and operator errors, to ensure that core safety limits are not exceeded and to ensure the integrity of the reactor coolant boundary, containment, and associated systems.

Monitors are also provided to automatically detect potential core thermal oscillations in conditions of low reactor coolant flow and high power. These monitors, when fully installed and armed, will initiate an automatic suppression system trip to scram the reactor through the reactor protection system.

Instrumentation and control features of systems which can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems are described in detail in Chapter 7. Also described there are instrumentation and control features of other systems associated with the reactor.

3.1.2.2.5 Criterion 14 - Reactor Coolant Pressure Boundary

The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure and of gross rupture.

Response

Those systems comprising the reactor coolant pressure boundary were designed and analyzed conservatively when compared to ASME, ANSI, and other appropriate codes. The probability of abnormal leakage, rapidly propagating failure, and gross rupture is extremely low.

Tests, inspections, material certification, and a continuing surveillance program will ensure continuing satisfaction of this criterion.

3.1.2.2.6 Criterion 15 - Reactor Coolant System Design

The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

Response

The reactor coolant system consists of the reactor vessel and appurtenances, the reactor recirculation system, the nuclear pressure relief system, the main steam lines, the reactor core isolation cooling system, and the shutdown cooling system. These systems are designed, fabricated, erected, and tested to stringent quality requirements and appropriate codes and standards which assure high integrity of the reactor coolant pressure boundary throughout the plant lifetime. The reactor coolant system is designed and fabricated to meet the following as a minimum:

1. Reactor Vessel - ASME Boiler and Pressure Vessel Code, Section III, Subsection A

2. Pumps - ASME Boiler and Pressure Vessel Code, Section III, Subsection C

3. Piping and Valves - USAS B31.1, Code for Pressure Power Piping.

The auxiliary, control, and protection systems associated with the reactor coolant system act to provide sufficient margin to ensure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences. As described in other criteria, instrumentation is provided to monitor essential variables to ensure that they are within prescribed operational limits. If the monitored variables exceed their predetermined settings, the auxiliary, control, and protection systems automatically respond to maintain the variables and systems within allowable design limits.

3.1.2.2.7 Criterion 16 - Containment Design

Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

Response

A pressure-suppression containment system consisting of a drywell, suppression chamber (torus), and interconnecting vent piping is the primary containment for the main coolant system. During normal operation, the reactor building, containing the pressure-suppression system, provides a secondary containment barrier.

To ensure the integrity of the primary containment, integrated leak tests were performed prior to facility operation, and will be periodically performed thereafter as provided in the Technical Specifications. The results of the initial tests demonstrated that the containment met the design leak rate and therefore, provides an essentially leak-tight barrier. The design basis LOCA was evaluated as the primary containment maximum allowable accident leak rate. The analysis demonstrates that the offsite doses from this accident would be well within the limits of 10 CFR 100 or 10CFR 50.67 as applicable.

Double isolation valves are provided on lines penetrating the primary containment and open to the free space of the containment. Closure of one of these valves is sufficient to maintain integrity of the containment.

3.1.2.2.8 Criterion 17 - Electric Power Systems

An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety functions for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences, and (2) the core is cooled and containment integrity and other vital functions are maintained in the vent of postulated accidents.

The onsite electric power supplies, including the batteries, and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.

Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights-of-way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a LOCA to assure that core cooling, containment integrity and other vital safety functions are maintained.

Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.

<u>Response</u>

Electric power from Units 2 and 3 is transmitted through a 345 kV system. Auxiliary power can be supplied from five separate and independent sources: Units 2 and 3, the 138 and 345 kV transmission system, and diesel generators.

The auxiliary power supply from the 138 and 345 kV transmission system is protected against the effect of unplanned outages by the diversity of seven separate 345 kV circuits and six 138 kV circuits and two major generating units feeding into the two switchyards at the Dresden site.

Each unit has adequate auxiliary power supplied from either the 138 or 345 kV switchyard.

One auxiliary power supply for Unit 2 is from a unit auxiliary transformer connected to its generator leads. The reserve auxiliary power supply is from a transformer connected to the 345 kV switchyard via transformer 86, which is connected to the 138 kV output of Transformer 86 via a short transmission line and a 138 kV circuit breaker. Transformer 86 is a 345 kV to 138 kV auto transformer located adjacent to the 345 kV switchyard and connected to the 345 kV switchyard through the circuit breakers.

Each connection is equipped with two disconnecting switches in series so that failure of any one disconnecting switch can involve not more than one 138 kV bus section. The switching sequence will be protected by electrical interlocks so that one connection must be opened before the other can be closed.

The auxiliary power supply for Unit 3 is split between the unit auxiliary power transformer which is connected to the generator leads and the reserve auxiliary power transformer which is connected to the 345 kV bus at Dresden.

It is impossible for the failure of any one component of either the 138 or 345 kV transmission systems to cause a simultaneous outage of both buses at Dresden.

Two of the 345 kV circuits leave the Dresden bus on double circuited towers in a southerly direction for a distance of approximately one mile, then turn east for a distance of 2¾ miles and then turn north easterly to Gooding's Grove Transmission Substation.

One of the 345 kV circuits leaves the Dresden bus on single circuited towers in a southwesterly direction to Pontiac Midpoint Transmission Substation.

The auxiliary power system provides adequate power to operate all the station auxiliary loads necessary for station operation. The station auxiliary bus is also connected by appropriate switching sequences to diesel generators which will provide standby power in the event of total loss of auxiliary power from offsite sources. This power source is physically independent from any normal power system and each power source, up to the point of its connection to the auxiliary power bus, is capable of complete and rapid electrical isolation from any other sources.

Provisions are made for supplying a source of electrical power which is self contained within the plant and not dependent on normal sources of supply. The diesel generator system produces ac power at a voltage and frequency compatible with normal bus requirements. The diesel generators are sized so that one can carry the ECCS power requirements on one unit within the rating of the diesel or that power necessary for safe shutdown of the unit. The second diesel generator is shared by both Units 2 and 3. In addition, the system is of sufficient capacity to start all initial loads it is expected to drive.

A total of three (250-volt, 125-volt, and 24/48-volt) station battery systems are provided for each unit. One 250-volt "power" battery is provided to serve the larger loads such as dc motor driven pumps, valves, etc. One 125-volt "control" battery is provided to supply the power required for exit lighting and all dc control functions such as that required for control of the 4 kV breakers, 480-volt breakers, various control relays, annunciators, etc. Two 24/48-volt batteries are provided to supply the neutron monitoring system.

The battery rooms and their locations are discussed in Section 8.3.2.

The 250-volt station battery is sized with a capacity suitable to supply emergency power for a time deemed adequate to safeguard the plant until normal sources of power are restored. The battery chargers are sized with a capacity suitable for restoring the battery to full charge under normal (not emergency) load in a time commensurate with the recommendations of the battery manufacturer.

The 125 volt station battery is sized with a capacity suitable to supply emergency power for a time deemed adequate to safeguard the plant until normal sources of power are restored. The battery chargers are sized with a capacity suitable for restoring the battery to full charge under normal (not emergency) load in a time commensurate with the recommendations of the battery manufacturer.

3.1.2.2.9 Criterion 18 - Inspection and Testing of Electric Power Systems

Electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses, and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system.

Response

The electrical power systems are designed with the capability of periodic testing for operability. Components of the systems, e.g., onsite power sources, relays, and switches are similarly capable of being periodically tested. Passive components such has wiring, connections, switchboards, and buses are capable of periodic inspection. In accordance with the Technical Specifications, individual power sources are checked for operability during plant operation. In accordance with the surveillance frequency control program, each system is functionally checked as a whole, under conditions as close to design as practical. The operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, an associated power transfers, is effected insofar as is practical and within the limits of safety.

3.1.2.2.10 Criterion 19 - Control Room

A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including LOCAs. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. For those accidents analyzed using Alternative Source Term (AST), radiation exposure criteria are listed in 10 CFR 50.67.

Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

Response

The facility is equipped with a central control room having adequate instrumentation for safe operation of the plant under normal conditions and to maintain the plant in a safe condition under accident conditions.

The control room has adequate shielding, fire protection, breathing air supplies, and access facilities to permit continuous occupancy under 10 CFR 20 dose limits during all design basis accident conditions. Although it should never be necessary to evacuate the control room, the facility design includes the capability for shutting down the reactor and establishing safe conditions from outside the control room.

3.1.2.3 Group 3 - Protection and Reactivity Control Systems

3.1.2.3.1 Criterion 20 - Protection System Functions

The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences, and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

Response

A dual-redundant fail-safe reactor protection system incorporating multiple logic inputs is provided to automatically initiate appropriate action whenever facility conditions reach pre-established limits to ensure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences.

The protection system is capable of sensing accident conditions and will initiate the operation of systems and components important to safety.

The protection system functions are tabulated in the Technical Specifications.

3.1.2.3.2 Criterion 21 - Protection System Reliability and Testability

The protective system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function, and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

Response

The protection system is designed for high functional reliability and inservice testability which is considered to be commensurate with the safety functions to be performed. Redundancy is designed into the protection system and is sufficient to ensure that no design basis single failure results in loss of the protective function. Removal from service for brief intervals during anticipated testing, calibration or maintenance does not disable the protective function. The high reliability of the function during these intervals can be demonstrated.

Active components in the protection system and redundant subsystems are capable of being tested or removed from service during reactor operation without compromising the protection function, even in the event of a subsequent single failure.

A failure of any one reactor protection system input or subsystem will not prevent a subsequent trip signal or a tripped condition on both channels from initiating the protective function. All control equipment and motive power sources are designed to operate under conditions of design basis environmental extremes. The protection system is designed to assure that the effects of natural phenomena and of normal operating maintenance testing and postulated accident conditions on redundant channels do not result in loss of the protective function.

The design employs spatial and/or functional diversity sufficient to meet a single failure criterion which allows unlimited damage to any single active device or module and unlimited damage to any single panel outside the main control room.

3.1.2.3.3 Criterion 22 - Protection System Independence

The protection system shall be designed to assure that the effects of natural phenomena and of normal operating, maintenance, testing and postulated accident conditions of redundant channels do not result in loss of the protection function or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

Response

The protection system is designed for high functional reliability and inservice testability which is considered to be commensurate with the safety functions to be performed. Redundancy is designed into the protection system and is sufficient to ensure that no design basis single failure results in loss of the protective function. Removal from service for brief intervals during anticipated testing, calibration or maintenance does not disable the protective function. The high reliability of the function during these intervals can be demonstrated.

Active components in the protection system and redundant subsystems are capable of being tested or removed from service during reactor operation without compromising the protection function, even in the event of a subsequent single failure.

A failure of any one reactor protection system input or subsystem will not prevent a subsequent trip signal or a tripped condition on both channels from initiating the protective function. All control equipment and motive power sources are designed to operate under conditions of design basis environmental extremes.

The protection system is designed to assure that the effects of natural phenomena and of normal operating maintenance testing and postulated accident conditions on redundant channels do not result in loss of the protective function.

The design employs spatial and/or functional diversity sufficient to meet a single failure criterion which allows unlimited damage to any single active device or module and unlimited damage to any single panel outside the main control room.

3.1.2.3.4 Criterion 23 - Protection System Failure Modes

The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced

Response

The protection system is designed for high functional reliability and inservice testability which is considered to be commensurate with the safety functions to be performed. Redundancy is designed into the protection system and is sufficient to ensure that no design basis single failure results in loss of the protective function. Removal from service for brief intervals during anticipated testing, calibration or maintenance does not disable the protective function. The high reliability of the function during these intervals can be demonstrated.

Active components in the protection system and redundant subsystems are capable of being tested or removed from service during reactor operation without compromising the protection function, even in the event of a subsequent single failure.

A failure of any one reactor protection system input or subsystem will not prevent a subsequent trip signal or a tripped condition on both channels from initiating the protective function. All control equipment and motive power sources are designed to operate under conditions of design basis environmental extremes.

The protection system is designed to assure that the effects of natural phenomena and of normal operating maintenance testing and postulated accident conditions on redundant channels do not result in loss of the protective function.

The design employs spatial and/or functional diversity sufficient to meet a single failure criterion which allows unlimited damage to any single active device or module and unlimited damage to any single panel outside the main control room.

3.1.2.3.5 Criterion 24 - Separation of Protection and Control Systems

The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems, leaves intact a system satisfying all reliability, redundancy and independence requirements of the protection system. Interconnection of the protection and control system shall be limited so as to assure that safety is not significantly impaired.

Response

Sensors and electrical circuits (except in certain portions of main control room panels) necessary to the functioning of the protection systems are physically and electrically separated to prevent any single event, including single failures in the control systems, from compromising the protection function.

There are not active elements of the reactor or nuclear plant control systems whose failure can impair the operation of the reactor protection systems, containment isolation system or emergency core cooling system. All active sensory equipment for the reactor protection, isolation and safeguards functions are independent of those used for automatic controls. Passive components (sensing lines) are arranged in redundant sets such that failure of one set will not disable the protective function.

3.1.2.3.6 Criterion 25 - Protection System Requirements for Reactivity Control Malfunctions

The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.

Response

The reactor protection system is designed to ensure that the specified fuel design limits are not exceeded for any single malfunction of the reactivity control systems. Acceptable fuel design limits will not be exceeded for any reactivity transient resulting from a single equipment malfunction or operator error.

3.1.2.3.7 Criterion 26 - Reactivity Control System Redundancy and Capability

Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

Response

The facility contains a control rod system and a liquid poison system for the control of reactivity. These systems are based on different design principles and are independent. The control rod system, in conjunction with the use of burnable poison in the fuel and the reactor coolant recirculation system flow control, has the capability of controlling reactivity changes resulting from load changes, long-term reactivity changes, xenon burnout and fuel burnup. Reactor shutdown by the control rod system, in conjunction with the reactor protection system, is sufficiently rapid to prevent fuel design limits from being exceeded during any anticipated operational transients. The control rod system is designed with a positive means of insertion and is capable of maintaining the reactor subcritical under hot or cold conditions with the highest worth control rod in the fully withdrawn position.

The liquid poison system is capable of bringing and maintaining the reactor core subcritical either in its hot or in its most reactive condition independent of the control rod system.

3.1.2.3.8 Criterion 27 - Combined Reactivity Control Systems Capability

The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that, under postulated accident conditions and with appropriate margin for stuck rods, the capability to cool the core is maintained.

Response

The liquid poison system is provided to bring the reactor to a cold shutdown condition at any time in core life independent of the control rod system capabilities. The most severe requirement imposed on the liquid poison system is to shut down the reactor from a full-power operating condition, assuming complete failure of the withdrawn control rods to respond to an insertion signal. The rate of negative reactivity insertion provided by the liquid poison system is designed to exceed the rate of reactivity gain associated with reactor cooldown from the full-power condition. The liquid poison and the emergency core cooling systems are separate and independent systems.

3.1.2.3.9 Criterion 28 - Reactivity Limits

The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding, nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

Response

The facility design incorporates a control rod velocity limiter to limit the rate of reactivity addition in the event of a control rod drop accident and a control rod housing support to prevent the control rod ejection accident. These engineered safeguards ensure that reactivity additions to the core will not result in damage to the reactor coolant system. The effects of these reactivity additions will not disturb the reactor core, its support structures, or other vessel internals sufficiently to impair core cooling.

Chapter 15 of the UFSAR evaluates the postulated reactivity accidents as well a abnormal operational transients in detail. The results of these analyses indicate that none of the postulated transients result in damage to the reactor coolant pressure boundary.

3.1.2.3.10 Criterion 29 - Protection Against Anticipated Operational Occurrences

The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

Response

A high functional reliability of the protection and reactivity control systems has been achieved through the combination of logic arrangement, redundancy, physical and electrical independence, functional separation, fail-safe design and inservice testability.

The instrumentation and control for the protection system, containment isolation system and emergency core cooling system have been designed to assure an extremely high probability for accomplishing safety functions in the event of any anticipated operational occurrences, including single operator error or single equipment malfunction.

The high probability of correct protection and reactivity control system response to anticipated operational occurrences is maintained by a thorough program of inservice testing and surveillance.

Active components can be tested or removed from service for maintenance during reactor operation without compromising the protection or reactivity control functions even in the event of a subsequent single failure.

Chapter 15 of the UFSAR evaluates the postulated reactivity accidents as well as abnormal operational transients in detail. The results of these analyses indicate that none of the postulated transients result in damage to the reactor coolant pressure boundary.

3.1.2.4 Group 4 - Fluid Systems

3.1.2.4.1 Criterion 30 - Quality of Reactor Coolant Pressure Boundary

Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

Response

As discussed in Section 3.1.2.2.5, the reactor coolant pressure boundary is designed, fabricated, erected, and tested conservatively when compared to appropriate codes and standards.

Detection devices capable of determining the magnitude of a broad spectrum of primary system leaks have been installed.

3.1.2.4.2 Criterion 31 - Fracture Prevention of Reactor Coolant Pressure Boundary

The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating maintenance, testing, and postulated accident condition, (1) the boundary behaves in a non-brittle manner, and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady-state and transient stresses, and (4) size of flaws.

Response

The reactor coolant pressure boundary was fabricated, inspected, and tested in accordance with applicable codes (e.g., ASME, ASA) as described in Section 5.2. These codes are intended to ensure that the boundary behaves in a non-brittle manner and the probability of a rapid propagating fracture is minimized. A maximum NDT temperature for the vessel shell material was established.

Conformance to the codes is described in Section 3.2. Suitable representative metallurgical specimens from the vessel and other areas of interest are located within the core region to permit periodic monitoring of exposure effects on the specimen properties relative to unirradiated control samples.

Transient analyses based on design values have shown that the design of the primary coolant boundary adequately satisfies the applicable design codes.

All piping, pumps, and valves defined by ASME Boiler and Pressure Vessel Code, Section XI, January 1, 1970 issue, to be part of the primary coolant pressure boundary and all components of the reactor vessel above the biological shield, such as the closure head, head to flange, vessel to flange, and nozzles are accessible for inspection during refueling. Also, the reactor pressure vessel within the biological shield is accessible for inspection at the nozzles through removal of shield plugs and thermal insulation around the nozzles. Surveillance specimens have been provided for periodic testing.

3.1.2.4.3 Criterion 32 - Inspection of Reactor Coolant Pressure Boundary

Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leak-tight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

Response

The reactor coolant pressure boundary was fabricated, inspected, and tested in accordance with applicable codes (e.g., ASME, ASA) as described in Section 5.2. These codes are intended to ensure that the boundary behaves in a non-brittle manner and the probability of a rapid propagating fracture is minimized. A maximum NDT temperature for the vessel shell material was established.

Conformance to the codes is described in Section 3.2. Suitable representative metallurgical specimens from the vessel and other areas of interest are located within the core region to permit periodic monitoring of exposure effects on the specimen properties relative to unirradiated control samples.

Transient analyses based on design values have shown that the design of the primary coolant boundary adequately satisfies the applicable design codes.

All piping, pumps, and valves defined by ASME Boiler and Pressure Vessel Code, Section XI, January 1, 1970 issue, to be part of the primary coolant pressure boundary and all components of the reactor vessel above the biological shield, such as the closure head, head to flange, vessel to flange, and nozzles are accessible for inspection during refueling. Also, the reactor pressure vessel within the biological shield is accessible for inspection at the nozzles through removal of shield plugs and thermal insulation around the nozzles. Surveillance specimens have been provided for periodic testing.

3.1.2.4.4 Criterion 33 - Reactor Coolant Makeup

A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.

Response

During normal operation, water level in the reactor vessel is maintained by the feedwater system. For small breaks in the reactor coolant pressure boundary, the makeup capability is provided by the feedwater system and the control rod drive system. The emergency core cooling system provides inventory makeup for breaks beyond the capability of the feedwater system.

Various sensing devices capable of determining the severity of a rupture in the reactor coolant boundary initiate proper sequences in the reactor protection system to start the required equipment necessary to assure acceptable fuel design limits are not exceeded. These systems are designed to operate under various accident situations and can be powered by redundant offsite sources, as appropriate.

3.1.2.4.5 Criterion 34 - Residual Heat Removal

A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

Suitable redundancy in components and features and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Response

The Dresden RHR capability is accomplished through the reactor shutdown cooling system (SDCS) and the LPCI system.

Since the system also functions under emergency cooling conditions, sufficient redundancy, independent power sources and high reliability are inherent. The system retains all isolation, leak detection, independence and interconnection features regardless of the operating mode.

3.1.2.4.6 Criterion 35 - Emergency Core Cooling

A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented, and (2) clad metal-water reaction is limited to negligible amounts.

Suitable redundancy in components and features and suitable interconnections, leak detection, isolation and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Response

An emergency core cooling system has been provided that is designed to transfer heat from the reactor core following any LOCA at a rate such that fuel and clad damage that could interfere with continued effective core cooling is prevented and clad metal-water reaction is limited to negligible amounts.

The ECCS is comprised of several independent, redundant systems, designed to provide adequate cooling for the core throughout the pressure/temperature range from full power to cold shutdown. The systems are further designed to operate from several redundant power sources and in a sequence conducive to the attainment of stable conditions.

The systems include high and low pressure flooding, spray, and residual heat rejection capacity in excess of that required for any given occurrence. The Technical Specifications and previous criteria discussions describe the component and system tests and inspections that verify the integrity and capability of the systems.

3.1.2.4.7 Criterion 36 - Inspection of Emergency Core Cooling System

The emergency core cooling system shall be designed to permit appropriate periodic inspection of important components, such as spray rings in the reactor pressure vessel, eater injection nozzles and piping, to assure the integrity and capability of the system.

Response

An emergency core cooling system has been provided that is designed to transfer heat from the reactor core following any LOCA at a rate such that fuel and clad damage that could interfere with continued effective core cooling is prevented and clad metal-water reaction is limited to negligible amounts.

The ECCS is comprised of several independent, redundant systems, designed to provide adequate cooling for the core throughout the pressure/temperature range from full power to cold shutdown. The systems are further designed to operate from several redundant power sources and in a sequence conducive to the attainment of stable conditions.

The systems include high and low pressure flooding, spray, and residual heat rejection capacity in excess of that required for any given occurrence. The Technical Specifications and previous criteria discussions describe the component and system tests and inspections that verify the integrity and capability of the systems.

3.1.2.4.8 Criterion 37 - Testing of Emergency Core Cooling System

The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leak-tight integrity of its components, (2) the operability and performance of the active components of the system, and 93) the operability of the system as a whole and, under conditions as close to the design as practical, the performance of the full operational sequence that bring the system into operation, including operation of applicable operations of the protection system, the transfer between normal and emergency power sources and the operation of the associated cooling water system.

Response

An emergency core cooling system has been provided that is designed to transfer heat from the reactor core following any LOCA at a rate such that fuel and clad damage that could interfere with continued effective core cooling is prevented and clad metal-water reaction is limited to negligible amounts.

The ECCS is comprised of several independent, redundant systems, designed to provide adequate cooling for the core throughout the pressure/temperature range from full power to cold shutdown. The systems are further designed to operate from several redundant power sources and in a sequence conductive to the attainment of stable conditions.

The systems include high and low pressure flooding, spray, and residual heat rejection capacity in excess of that required for any given occurrence. The Technical Specification and previous criteria discussions describe the component and system tests and inspections that verify the integrity and capability of the systems.

3.1.2.4.9 Criterion 38 - Containment Heat Removal

A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any LOCA and maintain them at acceptably low levels.

Suitable redundancy in components and features and suitable interconnections, leak detection, isolation and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Response

Two separate and independent containment spray systems are provided to remove heat, reduce pressure, and restore the pressure suppression system temperature following a LOCA. Each system is capable of removing all the decay heat and, in addition, the energy from any credible metal-water reaction at a rate which will prevent containment pressures and temperatures from exceeding their design values.

The power for the pumps is provided from diverse offsite supply systems or from one of two emergency generators. The spray system pumps start automatically on the combined condition of high drywell pressure and low-low reactor water level. The spray systems are manually controlled from the main control room.

3.1.2.4.10 Criterion 39 - Inspection of Containment Heat Removal System

The containment heat removal system shall be designed to permit appropriate periodic inspection of important components, such as the torus, sumps, spray nozzles, and piping to assure the integrity and capability of the system.

Response

The containment heat removal system is designed to permit periodic inspection of all important components to ensure the integrity and capability of the system.

The system is designed to permit appropriate periodic pressure and functional testing. Pumps are periodically tested for flow, developed pressure, and automatic initiation. The test program demonstrates that pump sets function, under simulated condition, in the same manner in which the systems are required to operate under accident conditions.

Testing of emergency power sources for containment cooling is periodically performed. The power systems are tested for automatic pickup of load required for the LOCA. The testing simulates accident conditions.

3.1.2.4.11 Criterion 40 - Testing of Containment Heat Removal System

The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leak-tight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole, and, under conditions as close to the design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources and the operation of the associated cooling water system.

Response

The containment heat removal system is designed to permit periodic inspection of all important components to ensure the integrity and capability of the system.

The system is designed to permit appropriate periodic pressure and functional testing. Pumps are periodically tested for flow, developed pressure, and automatic initiation. The test program demonstrates that pump sets function, under simulated condition, in the same manner in which the systems are required to operate under accident conditions.

Testing of emergency power sources for containment cooling is periodically performed. The power systems are tested for automatic pickup of load required for the LOCA. The testing simulates accident conditions.

3.1.2.4.12 Criterion 41 - Containment Atmosphere Cleanup

Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quantity of fission products released to the environment following postulated accidents and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

Response

The station design provides two containments which act as barriers to fission products, hydrogen, oxygen, and other substances that may be released into the containments.

Primary containment is provided by a nitrogen-inerted, pressure suppression system. The nitrogen precludes combustion of hydrogen from metal-water reactions. Means to accommodate hydrogen and oxygen produced by radiolysis are discussed in Section 6.2.5.

Secondary containment is provided by the reactor building. This building is used for control and cleanup of fission products released during postulated accidents. This containment is maintained at subatmospheric pressure so that releases can be controlled and directed to the stack through the standby gas treatment system.

3.1.2.4.13 Criterion 42 Inspection of Containment Atmosphere Cleanup Systems

The containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components, such as filter frames, ducts, and piping to assure the integrity and capability of the systems.

Response

The two redundant containment standby gas treatment systems have been designed to permit appropriate periodic inspection and testing to assure the integrity and capability of the systems. Procedures have been developed for measuring the efficiency of the system using test gases. Each system is tested for automatic operation once each operating cycle.

3.1.2.4.14 Criterion 43 - Testing of Containment Atmosphere Cleanup Systems

The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leak-tight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves, and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources and the operation of associated systems.

Response

The two redundant containment standby gas treatment systems have been designed to permit appropriate periodic inspection and testing to assure the integrity and capability of the systems. Procedures have been developed for measuring the efficiency of the system using test gases. Each system is tested for automatic operation once each operating cycle.

3.1.2.4.15 Criterion 44 - Cooling Water

A system to transfer heat from structures, systems, and components important to safety to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.

Suitable redundancy in components and features and suitable inter-connections, leak detection and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Response

Systems have been designed to transfer heat from structures, systems, and components important to safety. Ultimately, heat is rejected to the atmosphere or the Illinois River. The systems are designed for individual, complementary or supplemental operation depending on the conditions in the facility. Appropriate redundancy, interconnections, isolation capability and alternate power sources have been provided to ensure system safety functions can be accomplished in the event of a single failure.

3.1.2.4.16 Criterion 45 - Inspection of Cooling Water System

The cooling water system shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the system.

Response

The cooling water systems are designed to permit appropriate periodic inspection of all important components to ensure integrity and capability of the systems. The inspection program for cooling water systems is included in Section 9.2.

3.1.2.4.17 Criterion 46 - Testing of Cooling Water System

The cooling water system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leak-tight integrity of its components, (2) the operability and the performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for reactor shutdown and for LOCAs, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

Response

The cooling water systems are designed to permit appropriate periodic pressure and functional testing to ensure structural and leak-tight integrity of the components, operability and performance of components and the system as a whole. Procedures for testing the active components of the system have been developed that ensure operability and integrity. Overall system operation is verified under conditions as close to design as practical and through operation of the appropriate protection system, including power transfer between normal and emergency sources. The testing procedures can be related to all plant functions including the postulated LOCAs.

3.1.2.5 Group 5 - Reactor Containment

3.1.2.5.1 Criterion 50 - Containment Design Basis

The reactor containment structure, including access openings, penetrations and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and energy from metal-water and other chemical reactions that may result from degraded emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

Response

The reactor containment structure, including penetrations, valves, access openings and the containment spray system are designed with sufficient margin to accommodate the temperatures and pressures associated with the LOCA. The design bases and response of the containment system are described in Section 6.2.

The containment system is designed to accommodate design temperature and pressure while maintaining the low leakage rate required by the Technical

Specifications. Special precautions, such as double-sealed access ways and penetrations, are taken to minimize containment leakage. The integrity of the complete containment is designed and will be maintained to limit offsite doses from postulated design basis accidents to a value below the guideline values of 10 CFR 100 or 10CFR 50.67 as applicable.

A large body of experimental data has been obtained on BWR suppression containment performance. Furthermore, very conservative assumptions have been used in the analytical model.

3.1.2.5.2 Criterion 51 - Fracture Prevention of Containment Pressure Boundary

The reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing and postulated accident conditions (1) its ferritic materials behave in a non-brittle manner, and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady-state and transient stresses, and (3) size of flaws.

Response

The reactor containment boundary is designed with sufficient margin to ensure that under operating, maintenance, testing, and postulated accident conditions, ferritic materials behave in a non-brittle manner, and the probability of a rapidly propagating fracture is minimized.

The primary containment is designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, as described in Section 3.8.2.1.2. Transients for normal and accident conditions were evaluated. The analysis includes transients for the steel shell and for penetrations. The low stress levels achieved allow adequate safety margins for uncertainty in determining exact material properties and stresses.

3.1.2.5.3 Criterion 52 - Capability for Containment Leakage Rate Testing

The reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.

Response

The leakage rate testing program for the primary containment is described in the Technical Specifications. Preoperational tests were conducted at the design pressures of the containment and also at the calculated accident pressures. These tests established the leakage relationship to containment pressure. Subsequent periodic leakage tests will be conducted at either 25 psig or 43.9 psig.

3.1.2.5.4 Criterion 53 - Provisions for Containment Testing and Inspection

The reactor containment shall be designed to permit (1) appropriate periodic inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leak-tightness of penetrations which have resilient seals and expansion bellows.

Response

The reactor containment is designed to permit appropriate periodic inspection of all important areas, penetrations, and seals. Provisions for leak testing on a scheduled basis have been provided. A surveillance program has been implemented.

3.1.2.5.5 Criterion 54 - Piping Systems Penetrating Containment

Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

Response

Compliance with Appendix A to 10 CFR 50, "General Design Criteria for Nuclear Power Plants," effective July 7, 1971, was not included as a design basis for structures, systems or components for Dresden Unit 2 since the General Design Criteria were not in existence at the time of issuance of the plant construction permit (April 15, 1966) or the plant provisional operating license (December 22, 1969).

The primary containment isolation system was designed in accordance with the licensing criteria in effect at the time of issuance of the provisional operating license.

These criteria are stated as follows:

A. The primary containment isolation system complies with IEEE-279 (August 1968), "Proposed Criteria for Nuclear Power Plant Protection Systems."

B. Piping other than instrument sensing lines which penetrates the primary containment and is in direct communication with the reactor coolant system has two automatic isolation valves, one inside the primary containment and one outside the primary containment.

C. Piping other than instrument sensing lines which penetrates the primary containment and communicates with the primary containment free space has two automatic isolation valves, both located outside the primary containment.

D. Piping which penetrates the primary containment but communicates directly with neither the reactor vessel, the primary containment free space, nor the environs, has at least one isolation valve located outside the primary containment which may close either by process action (reverse flow) or by remote manual operation.

E. Instrument sensing lines which communicate directly with the reactor coolant system have one excess flow check valve and one manual globe valve located outside the primary containment.

F. Instrument sensing lines which communicate directly with the primary containment free space have one manual globe valve located outside the primary containment.

G. The operability of each automatic isolation valve may be tested periodically.

The criteria used meet the intent of GDC 54 which gives general requirements for isolation system redundancy, reliability, and testability.

GDC 55, 56, and 57 give specific requirements for particular classes of penetrations; the isolation valving arrangements used on all fluid line penetrations comply with the applicable GDC.

3.1.2.5.6 Criterion 55 - Reactor Coolant Pressure Boundary Penetrating Containment

Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines such as instrument lines, are acceptable on some other defined basis.

Isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them, shall be provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication and testing, additional provisions for inservice inspection, protection against more severe natural phenomena and additional isolation valves and containment, shall include consideration of the population density, use characteristics and physical characteristics of the site environs.

Response

Compliance with Appendix A to 10 CFR 50, "General Design Criteria for Nuclear Power Plants," effective July 7, 1971, was not included as a design basis for structures, systems or components for Dresden Unit 2 since the General Design Criteria were not in existence at the time of issuance of the plant construction permit (April 15, 1966) or the plant provisional operating license (December 22, 1969).

The primary containment isolation system was designed in accordance with the licensing criteria in effect at the time of issuance of the provisional operating license.

These criteria are stated as follows:

A. The primary containment isolation system complies with IEEE-279 (August 1968), "Proposed Criteria for Nuclear Power Plant Protection Systems."

B. Piping other than instrument sensing lines which penetrates the primary containment and is in direct communication with the reactor coolant system has two automatic isolation valves, one inside the primary containment and one outside the primary containment.

C. Piping other than instrument sensing lines which penetrates the primary containment and communicates with the primary containment free space has two automatic isolation valves, both located outside the primary containment.

D. Piping which penetrates the primary containment but communicates directly with neither the reactor vessel, the primary containment free space, nor the environs, has at least one isolation valve located outside the primary containment which may close either by process action (reverse flow) or by remote manual operation.

E. Instrument sensing lines which communicate directly with the reactor coolant system have one excess flow check valve and one manual globe valve located outside the primary containment.

F. Instrument sensing lines which communicate directly with the primary containment free space have one manual globe valve located outside the primary containment.

G. The operability of each automatic isolation valve may be tested periodically.

The criteria used meet the intent of GDC 54 which gives general requirements for isolation system redundancy, reliability, and testability.

GDC 55, 56, and 57 give specific requirements for particular classes of penetrations; the isolation valving arrangements used on all fluid line penetrations comply with the applicable GDC.

3.1.2.5.7 Criterion 56 - Primary Containment Isolation

Each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis.

Response

Compliance with Appendix A to 10 CFR 50, "General Design Criteria for Nuclear Power Plants," effective July 7, 1971, was not included as a design basis for structures, systems or components for Dresden Unit 2 since the General Design Criteria were not in existence at the time of issuance of the plant construction permit (April 15, 1966) or the plant provisional operating license (December 22, 1969).

The primary containment isolation system was designed in accordance with the licensing criteria in effect at the time of issuance of the provisional operating license.

These criteria are stated as follows:

A. The primary containment isolation system complies with IEEE-279 (August 1968), "Proposed Criteria for Nuclear Power Plant Protection Systems."

B. Piping other than instrument sensing lines which penetrates the primary containment and is in direct communication with the reactor coolant system has two automatic isolation valves, one inside the primary containment and one outside the primary containment.

C. Piping other than instrument sensing lines which penetrates the primary containment and communicates with the primary containment free space has two automatic isolation valves, both located outside the primary containment.

D. Piping which penetrates the primary containment but communicates directly with neither the reactor vessel, the primary containment free space, nor the environs, has at least one isolation valve located outside the primary containment which may close either by process action (reverse flow) or by remote manual operation.

E. Instrument sensing lines which communicate directly with the reactor coolant system have one excess flow check valve and one manual globe valve located outside the primary containment.

F. Instrument sensing lines which communicate directly with the primary containment free space have one manual globe valve located outside the primary containment.

G. The operability of each automatic isolation valve may be tested periodically.

The criteria used meet the intent of GDC 54 which gives general requirements for isolation system redundancy, reliability, and testability.

GDC 55, 56, and 57 give specific requirements for particular classes of penetrations; the isolation valving arrangements used on all fluid line penetrations comply with the applicable GDC.

3.1.2.5.8 Criterion 57 - Closed System Isolation Valves

Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic or locked closed or capable of remote manual operation. This valve shall be outside containment and located as closed to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

Response

Compliance with Appendix A to 10 CFR 50, "General Design Criteria for Nuclear Power Plants," effective July 7, 1971, was not included as a design basis for structures, systems or components for Dresden Unit 2 since the GDC were not in existence at the time of issuance of the plant construction permit (April 15, 1966) or the plant provisional operating license (December 22, 1969).

The primary containment isolation system was designed in accordance with the licensing criteria in effect at the time of issuance of the provisional operating license.

These criteria are stated as follows:

A. The primary containment isolation system complies with IEEE-279 (August 1968), "Proposed Criteria for Nuclear Power Plant Protection Systems."

B. Piping other than instrument sensing lines which penetrates the primary containment and is in direct communication with the reactor coolant system has two automatic isolation valves, one inside the primary containment and one outside the primary containment.

C. Piping other than instrument sensing lines which penetrates the primary containment and communicates with the primary containment free space has two automatic isolation valves, both located outside the primary containment.

D. Piping which penetrates the primary containment but communicates directly with neither the reactor vessel, the primary containment free space, nor the environs, has at least one isolation valve located outside the primary containment which may close either by process action (reverse flow) or by remote manual operation.

E. Instrument sensing lines which communicate directly with the reactor coolant system have one excess flow check valve and one manual globe valve located outside the primary containment.

F. Instrument sensing lines which communicate directly with the primary containment free space have one manual globe valve located outside the primary containment.

G. The operability of each automatic isolation valve may be tested periodically.

The criteria used meet the intent of GDC 54 which gives general requirements for isolation system redundancy, reliability, and testability.

GDC 55, 56, and 57 give specific requirements for particular classes of penetrations; the isolation valving arrangements used on all fluid line penetrations comply with the applicable GDC.

3.1.2.6 Group 6 - Fuel and Radioactivity Control

3.1.2.6.1 Criterion 60 - Control of Releases of Radioactive Materials to the Environment

The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

Response

The station radioactive waste control systems (including the liquid, gaseous, and solid radwaste subsystems) are designed to limit the offsite radiation exposure to levels below those of 10 CFR 20. The plant engineered safety systems (including the containment barriers) are designed to limit the offsite dose under various postulated design basis accidents to levels significantly below 10 CFR 100 or 10CFR 50.67 as applicable. The off-gas system is designed with sufficient holdup retention capacity so that during normal plant operation the controlled release of radioactive materials does not exceed the established release limits at the elevated plant stack.

Dresden Station Unit 2 is provided with the means to control all radioactive effluents. No unfavorable environmental conditions limit the release of radioactivity under normal operating conditions. Hence, the released radioactivity can and will be controlled and limited to the requirements of 10 CFR 20.

Evaluation of accidents for the facility provides assurance that the combined doses in offsite areas do not exceed the guidelines set forth in 10 CFR 100 or 10CFR 50.67 as applicable.

An augmented radwaste system was installed in 1974 to meet the guidance of 10 CFR 50, Appendix I.

Refer to Section 9.1.2.2.4 for a description of spent fuel storage and handling of Dry Cask Storage (DCS) systems and the Independent Spent Fuel Storage Installation (ISFSI).

3.1.2.6.2 Criterion 61 - Fuel Storage and Handling, and Radioactivity Control

The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

Response

With the exception of spent fuel stored in Dry Cask Storage (DCS) systems as discussed in Section 9.1.2.2.4, the fuel storage and handling systems and the radioactive waste systems are designed to ensure safety under normal and accident conditions. The design of these systems and the inspection and testing program for the fuel storage and handling systems are described in Section 9.1. Procedures and interlocks used in fuel handling are described in the Technical Specifications.

Shielding and residual heat removal systems are provided for fuel handling and storage and for the radioactive waste systems.

The fuel handling and storage systems are in the rector building, i.e., the secondary containment. The waste handling systems are in the waste disposal building. Both of these buildings have filter systems and serve as confinement for radioactivity.

3.1.2.6.3 Criterion 62 - Prevention of Criticality in Fuel Storage and Handling

Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

Response

The new fuel storage vault racks located inside the secondary containment reactor building have been designed for top entry and to prevent an accidental critical array even in the event the vault becomes flooded. Vault drainage is provided to prevent possible water collection.

With the exception of spent fuel stored in Dry Cask Storage (DCS) systems as discussed in Section 9.1.2.2.4, the handling and storage of spent fuel which is entirely within the reactor building (the secondary containment system), will be done in the spent fuel storage pool. The pool has provisions to maintain water clarity, temperature control and instrumentation to monitor water level.

The racks in which fuel assemblies are placed are designed, and arranged to ensure subcriticality in the storage pool.

3.1.2.6.4 Criterion 63 - Monitoring Fuel and Waste Storage

Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels, and (2) to initiate appropriate safety actions.

Response

Station radiation and process monitoring systems have been provided to monitor the significant process parameters and radiation levels throughout the facility, including fuel storage and waste handling. Control room readouts and alarms permit immediate action in the event abnormal conditions occur for both eventualities.

3.1.2.6.5 Criterion 64 - Monitoring Radioactive Releases

Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences and from postulated accidents.

Response

Station radiation and process monitoring systems have been provided for monitoring significant parameters from specific station process systems and specific areas including the station effluents to site environs and to provide alarms and signals for appropriate corrective actions. A complete description of the station radiation monitoring system is given in Section 11.5.

Semiannual effluent reports are submitted to the NRC. These reports include specific information on the quantities of the principal radionuclides released to the environs.

3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

The plant structures and equipment were divided into two categories. The categories were defined as:

A. Class I - Structures and equipment whose failure could cause a significant release of radioactivity or which are vital for a safe shutdown of the plant and the removal of decay heat.

B. Class II - Structures and equipment, either essential or nonessential to the operation of the station, but which are not essential for a safe shutdown.

The application of these categories in the original design is described in Sections 3.2.1 through 3.2.6 for systems, structures and components. These categories however still only apply to structures. Systems and components are now classified as described in Section 3.2.7.

All Class I structures, systems, and components are designed to accommodate the load conditions and stress criteria presented in Sections 3.8.2 and 3.8.4. Unless otherwise specified in the FSAR, structures and equipment not listed as Class I are Class II. The Class I systems and structures are listed in the following subsections.

3.2.1 Class I - Structures

The following structures are classified as Class I:

A. Drywell, and torus including vents and penetrations;

B. Reactor building, including Unit 2/3 HPCI building;

C. Control room;

D. 310-foot chimney; and

E. Floor drain surge tank.

3.2.2 Class I - Systems (Mechanical)

The following mechanical systems are classified as Class I:

A. Emergency core cooling systems including:

1. Core spray system;
2. High pressure coolant injection (HPCI) system
3. Low pressure coolant injection (LPCI)/containment cooling system
4. Automatic depressurization system

B. Containment cooling water system;

C. Standby gas treatment system;

D. Standby liquid control (SBLC) system; and

E. Isolation condenser system;

3.2.3 Class I - Nuclear Steam Supply Equipment

The following equipment is classified as Class I:

A. Reactor vessel;

B. Reactor vessel supports;

C. Control rods, control rod drive housings, control rod drive housing supports;

D. Fuel assemblies;

E. Core shroud;

F. Core supports;

G. Steam separator;

H. Steam dryer;

I. Reactor recirculating water subsystem including pumps and valves; and

J. All piping connections from the reactor vessel up to and including the first isolation valve external to the drywell

3.2.4 Class I - Systems (Electrical)

The following electrical systems are classified as Class I:

A. Station batteries;

B. Emergency diesel generators; and

C. Essential buses and other electrical gear for power to critical equipment.

3.2.5 Class I - Instrumentation and Controls

The following instrumentation and controls are classified as Class I:

A. Reactor level instrumentation;

B. Feedwater control instrumentation;

C. SBLC system instrumentation;

D. Control rod instrumentation;

E. Reactor protection system;

F. Neutron monitoring system;

G. Incore neutron monitoring system;

H. Area radiation monitors;

I. Process radiation monitors; and

J. Alternate rod insertion (this system is safety-related but is not seismically qualified).

CECo's Engineering Department has reviewed the rod control system and determined that this system is not safety-related. The rod control system is comprised of the rod position information system (RPIS), rod worth minimizer (RWM), and reactor manual control system (RMCS). These three subsystems interact to ensure proper insert/withdraw rod sequences. Maintaining proper sequence limits the consequence of a control rod drop accident, minimizing fuel damage. The concern associated with this accident is excessive radiation release to the environment. Yet, safety-related systems ensuring isolation on high radiation presently exist to satisfy 10 CFR 100 or 10CFR 50.67 as applicable requirements.

3.2.6 Class I - Miscellaneous Category

The following equipment, piping, and valves are classified as Class I:

A. Fuel storage facilities, including spent fuel handling equipment and new fuel storage equipment;

B. All piping connections from the reactor coolant pressure boundary up to and including the first isolation valve external to the drywell;

C. Main steam safety valves;

D. Electromatic relief valves;

E. Safety relief valve (Target Rock);

F. Containment penetrations, and

G. Primary containment isolation valves and piping (other than valves identified under Item B).

Class II structures supporting Class I structures, systems, or components were designed to Class II requirements but have been investigated to assure that the integrity of the Class I items is not compromised.

Class I items located within Class II structures include the following:

A. Control room complex (see Drawings M-3 and M-4);

B. Station batteries (see Drawing M-3);

C. Diesel generators and cooling pumps (see Drawings M-4 and M-10);

D. Containment cooling service water pumps (see Drawing M-5);

E. Standby gas treatment system (see Drawing M-3);

F. Essential service buses; and

G. Other electrical gear for power to critical equipment.

The integrity of these Class I items is not compromised.

3.2.7 Identification of Safety-Related Components of Systems or Structures

Generic Letter 83-28, "Required Actions Based on Generic Implications of Salem ATWS Events," defines safety-related systems and components as those necessary to ensure the following:

A. The integrity of the reactor coolant pressure boundary,

B. The capability to shut down the reactor and maintain it in a safe shutdown condition, or

C. The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to those referred to in 10 CFR 100.11 or 10CFR 50.67 as applicable.

Subsequent to Generic Letter 83-28 a reclassification of mechanical and electrical systems and components was under taken utilizing the “Guideline for safety classification of systems, components, and parts used in Nuclear Power Plant Applications (NCIG-17) NP-6895 Research Project Q101-20 Final Report, February 1991”. The definitions of safety related systems and components in NCIG-17 are the same as in Generic Letter 83-28. For purposes of the reclassification, the term safety related as defined above and safety Class I as defined in Section 3.2 are considered to be synonymous. However, should a difference arise, the licensing commitment (i.e., historical definition of Safety Class I) shall govern.

Classification of a system or structure as safety-related does not imply that every associated component is safety-related; individual components of safety-related systems or structures may be classified as nonsafety-related. Detailed application of safety-related classification is identified in the Master Equipment List (MEL). The MEL complies with Generic Letter 83-28 and NCIG-17 for safety-related equipment classification identification. The MEL also includes the Requirements Summary Matrices (RSM’s) which delineate the approach to system and components where a graded QA program was applied. The mechanical/electrical component section, snubber section, and the EQ section, have been transferred from MEL to an eletronic work control program such as EWCS

3.2.8 Industry Code Applicability to the Reactor Coolant System Pressure Boundary Components

Codes and standards applied to individual systems and components are contained in their specific UFSAR sections.

3.2.8.1 Valves (Except Main Steam Isolation, Safety, Relief, and Safety Relief Valves)

The following codes and standards are applicable to valves (except main steam isolation, safety, relief, and safety relief valves):

A. USAS B-31.1, and

B. ASME Section I.

3.2.8.2 Reactor Recirculation Pumps

ASME Section III, Class C, is applicable to the reactor recirculation pumps.

3.2.8.3 Main Steam Isolation, Safety, Relief, and Safety Relief Valves and Flow Restrictors

The following codes and standards are applicable to main steam isolation, safety, relief, flow restrictors, and safety relief valves:

A. USAS B-31.1,

B. ASME Section I, and

C. ASME Section III (Safety Relief Valve)

3.2.8.4 Reactor Pressure Vessel

The following codes and standards are applicable to the reactor pressure vessel:

A. ASME Section III, Subsection A, and

B. ASME Section I Code Case 1339.

3.2.8.5 Piping System

The following codes and standards are applicable to the piping system:

A. USAS B-31.1, and

B. ASME Section I.

3.2.9 Industry Code Applicability to Non-Reactor Coolant Pressure Boundary Components

The following list contains other components or systems for which codes and standards are applicable in-whole or in-part:

System/Component	Code
Batteries (Station batteries)	IEEE 308-1974; IEEE 450
Cable (new cable installations)	IEEE 384
Condenser pit level alarms	IEEE 279
Containment	ASME Section III, 1965 Edition, Class B
Containment air monitoring (CAM)	ASME III, Class 2; IEEE 323-1974; IEEE 344-1975
Containment penetrations	ASME Section III, Class B
Containment penetration fitting design	ASME Section VIII
Control rod drive	ASME Section III
Core spray piping	USAS B31.1
Core spray pump casing	ASME Section III, Class C
Core spray spargers and nozzles	ASME Section III
Core spray vessel nozzle	ASME SA 336, Code Case 1332
Fuel pool cooling heat exchanger	ASME Section III
Fuel pool cooling pump	ASME VIII
HPCI piping	USAS B-31.1 and ASME Section I
HPCI pumps	ASME Section III
Hydrogen injection system	USAS B-31.1 and ASME Section VIII
Isolation condenser heat exchanger shell	ASME Section VIII
Isolation condenser heat exchanger tube side	ASME Section III
Instruments (replacement instruments and new Regulatory Guide 1.97 instruments	IEEE 344-1975

LPCI pump casings	ASME Section III, Class C
Main steam piping	USAS B-31.1; ASME Section I and III
Off-gas piping	USAS ASA B-31.1
Off-gas recombiner/adsorber	ASME Section III, Subsection ND, Class 3
Oxygen injection tank (inner vessel)	ASME Section VIII, Division I
RBCCW heat exchangers	ASME Section VIII
Reactor protection system	IEEE 279-1968
Reactor water cleanup (RWCU) vessels (except as noted below)	ASME Section III
RWCU Regenerative Heat Exchanger	ASME Section VIII, 1992, 92 Add. (Reconciled to 1965 ASME Section III, Class C)
RWCU Non-Regenerative Heat Exchanger (tube Side)	ASME Section III, Class C 1965
RWCU Non-Regenerative Heat Exchanger (shell side)	ASME Section VIII
Shutdown cooling system	ASME III, Class C)
Suppression pool temperature monitoring system	IEEE 279-1971, 323-1974, 344-1971, 344-1975
Traversing incore probe guide tubes	ASME Section VIII

3.2.10 Control of Purchased Material, Equipment, and Services

The purpose of this section is to discuss the requirements for procurement, fabrication, and material documentation of component replacements and piping installation at the Dresden Station.

For replacements of components that were originally constructed to ASME Code Section III, Classes 1, 2 or 3 or to other standards within the scope of the current edition of Section XI, Division 1 of the code, Article IWA-4000 of Section XI specifies general requirements. Article IWA-4000 requires replacements of components to meet the requirements of the original construction code used or those of a later code edition and/or addenda approved in the codes and standards rule 10 CFR 50.55a. The material documentation and the non-destructive examination (NDE) requirements for piping installation and valve/pump fabrication are also provided in detail in ASME Code Section III and Section XI, ANSI B31.1, and ANSI B16.34. These requirements can be applied to the procurement of the replacement parts for existing valve disks and removable seats. These requirements, however, do not apply to other replacement valve internals.

If the replacement components available are found not to be in full compliance of the original code, it should be ensured that the level of quality of a replacement component is at least equivalent to the original code, as recommended in Generic Letter 89-09.

Currently at Dresden, repair/replacement materials for components that were originally constructed to ASME Section III, Classes 1, 2, or 3 or to other standards within the scope of Section XI of the code, are ordered in accordance with code requirements, i.e., the materials shall meet the requirements of the original construction code used or those of a later code edition and/or addenda approved in 10 CFR 50.55a. In the event of conflict between the existing procurement specifications for replacement components/parts and the codes, the codes shall take precedence.

3.2.11 References

3.3 WIND AND TORNADO LOADINGS

This section summarizes the pertinent features used in the design of the station for consideration of wind and tornado loads.

Two major considerations result from wind and tornado loadings: first, the capability of the structure to withstand pressure forces generated (Sections 3.3.1 and 3.3.2) and, second, the resulting effects of missiles (Section 3.5) propelled by excessively high winds that could occur in some tornado conditions.

3.3.1 Wind Loadings

3.3.1.1 Design Wind Velocity

As a minimum, all structures are designed to withstand a 110-mph wind load, which is in excess of the Uniform Building Code requirements. The applicability of 110 mph as the design wind velocity is discussed in Section 2.3.2. Certain structures, the reactor building for example, are capable of withstanding pressures generated by winds well in excess of 110 mph. These safety-related structures are discussed in greater detail in the following sections.

3.3.1.1.1 Reactor Building

The reactor building structure encloses the reactor, the primary containment, and most of the equipment associated with the safe operation of the reactor. However, operation of the plant requires that certain parts of the building be removable, and as a result, two types of construction have been selected. The lower portion of the building is a reinforced concrete structure, whereas the upper portion is a structural steel design with metal siding.

The entire structure encompasses what is known as the secondary containment, which is discussed in Section 6.2. The reinforced concrete portion of the building extends from the foundation at elevation 472'-6" to the reactor refueling floor at elevation 613'-0". The structural steel superstructure extends from elevation 613'-0" to the roof level. These two portions of the building have different wind resistance capacities.

The entire superstructure is capable of withstanding a load generated by a 170-mph wind. The resistance for wind loading of the reinforced concrete portion and of the steel frame and siding portions of the superstructure is discussed in more detail in Section 3.3.2.2.1.

3.3.1.1.2 Chimney

A single chimney is utilized for gaseous waste discharge for the combined operation of Units 2 and 3.

The Unit 2/3 chimney is a tapered, reinforced, concrete structure 310 feet high. The chimney is founded on rock at elevation 500'-0" and is designed to be fixed at the base.

The top of the chimney is at elevation 827'-6" or 168 feet above the top of the reactor building. The chimney is located on the centerline between the units and 303.5 feet north of the reactor building north wall. Thus the top of the chimney could not strike the operating floor of the reactor building, even if it toppled intact about its base, since this point is 318 feet from the centerline of the base. The ventilation exhaust is brought through breaching which gathers the various exhaust ducts above the radwaste building and bridges the space between the radwaste building and the chimney.

Access to the interior of the chimney is through an airtight door from the outside at elevation 561'-0". Exterior access to the top of the chimney and to the external platforms at elevations 561'-0" and 718'-0" is from the ground by means of a guarded ladder. At the gas sampling level, an all-around platform provides access and a working area. Chimney lighting is provided by means of fixed flood lights located near grade elevation.

Computer design calculations for each of the following three cases were made:

A. American Concrete Institute (ACI) Wind Design,

B. Proposed ACI-307 Earthquake Design, and

C. Uniform Building Code.

The chimney was designed for a minimum ambient temperature of -20°F and a maximum ambient temperature of 150°F. The normal temperature of gaseous waste is 70°F. Design and construction of the chimney were in accordance with applicable requirements of ACI "Standard Specification for the Design and Construction of Reinforced Concrete Stacks" (ACI-505), with the following exceptions and with those described in Section 3.8.4.3.

A. Wind Pressures

1. For normal design conditions, the chimney was designed to specified ACI allowable unit stresses. Wind pressures on projected areas for various height zones above ground conformed to the following requirements for a basic wind pressure of 30 lb/ft^2:

    a. Up to 100 feet above grade: 23 lb/ft^2

    b. Above 100 feet: 31 lb/ft^2

2. For tornado purposes, the tabular value for pressure was increased for a gust velocity of 110 mph in accordance with Paragraph 400 of ACI-505. The chimney was originally investigated to assure that failure would not occur for tornado winds of 300 mph, as described in Section 3.3.2.2.3.

    The chimney was designed for wind loads in accordance with ACI-505 as follows:

Elevation Above Ground	Required	Design Pressure
0 to 100 feet	23 lb/ft^2 + gust factor	26 lb/ft^2
100 to 310 feet	31 lb/ft^2 + gust factor	36 lb/ft^2

The American Society of Civil Engineers (ASCE) Paper No. 3269, Requirements For Wind Load Pressures (not used in the design, but discussed here for comparison), provides for a more analytical approach to the design of cylindrical shaped concrete chimneys:

The siting location, maximum wind conditions, and chimney configuration establish the following design parameters from Paper No. 3269:

A. Location: Northwestern Illinois (Area of maximum wind velocities for State of Illinois)

Maximum wind velocity (V)
50-year storm = 100 mph
100-year storm = 110 mph
For a 150-foot to 400-foot high chimney (inland area) = 140 mph

B. Drag Factor: The drag factor (C_D) allowance for a 310-foot high cylindrical shaped concrete chimney is determined by the relation of chimney height versus average diameter which

is $\frac{h}{d_A} = 16$

For conservatism, use $\frac{h}{d} = 25$ for a maximum tested drag factor of $C_D = 0.55$.

C. Gust Factor: The gust factor for the portion of the chimney from 0 feet to 100 feet high, assuming a 100-year storm, is the gust velocity divided by the wind speed (mph), or 130/110 = 1.2.

The gust factor for the portion of the chimney from the 100-foot to the 310-foot level is 1.4. This is a maximum condition based on the most severe inland wind for this location.

D. Computations: Design pressures are in accordance with ASCE Paper No. 3269.

1. q = Dynamic pressure for air at standard conditions

= $0.002558\ V^2$ where V = 110 mph (max.)

= 31 lb/ft^2.

2. p1 = Average pressure for the portion of the chimney from 0 feet to 100 feet

   = (q) (C_D) (gust factor)

   = (31) (0.55) (1.2)

   = Approximately 21 lb/ft^2

3. p2 = Average pressure for the portion of the chimney from 100 feet to 310 feet

   = (q) (C_D) (gust factor)

   = (31) (0.55) (1.4)

   = Approximately 23 lb/ft^2.

The wind load pressures used for the design of the chimney, therefore, are conservative compared to the criteria in ASCE Paper No. 3269.

An overturning effect was included in the design analysis of the chimney. Calculations show that the overturning factor of safety is greater than that required by the design codes as indicated below (also see Figure 3.3-1). The calculations show that the overturning factor of safety is actually 2.1, whereas the factor of safety allowed by the design code is 1.5. The safety factor for overturning due to the safe shutdown earthquake is 1.56.

Calculations for overturning:

V = Total vertical load including weight of concrete shaft, weight of concrete ring, and weight of foundation = 4710 k

M^W = Moment due to wind ≅28,120 k-ft

H^W = Horizontal wind shear = 192 k

M^{tot}= Total overturning moment = 28,120 + 192 x 17 ≅ 31,380 k-ft

M^s = Stability moment = 4710 x 14.25 ≅ 67,000 k-ft

n = Overturning safety factor = M^s/M^{tot} = 67,000/31,380 = 2.1, greater than the 1.5 allowed.

The factor of safety can be considered to be even higher because the chimney foundation is founded in and surrounded by rock.

The chimney working stress design was controlled by the wind from grade to an elevation of 160 feet above grade. Critical stresses occur at the flue opening as shown in Figure 3.3-2, where the reinforcing steel stress is 11,646 psi and the concrete stress is 729 psi.

3.3.2 Tornado Loadings

This subsection presents design basis tornado loadings for Dresden. Section 3.5 addresses potential missiles generated from tornadoes.

3.3.2.1 Applicable Design Parameters

The tornado parameters used in the design of structures at Dresden Station are as follows:

A. A maximum tangential velocity of 300 mph, and

B. A pressure drop of 6.3 lb/in.^{2}.

The design intent for the protection of systems from tornadoes is that the capability to shutdown the reactor is not impaired. As discussed throughout the UFSAR, there are a number of ways to safely shutdown the plant besides the standard use of the isolation condenser and shutdown cooling systems. Hence, the critical systems required for protection against tornadoes are the systems associated with a reactor scram and emergency core cooling. On this premise then, the only critical areas of concern are the reactor building and the control room.

3.3.2.2 Determination of Forces on Structures

3.3.2.2.1 Surface Pressure Effects on the Reactor Building

The reactor building can sustain 300-mph winds and still adequately permit shutdown of the reactor.

The structural steel frame of the reactor building can withstand the force of a 300-mph wind. The design criteria permitted stresses up to the yield point; however, the actual stresses are below the yield point. The structural system of the reactor building superstructure consists of horizontal roof trusses which transmit local loads from each bay to vertically braced bays at either end of the building. In the design of this system, loads caused by a 300-mph tornado were reduced to design loads by a factor of 22/36, which is the ratio of working stress to yield stress for Type A36 steel. As a result of the truss system, members were sized primarily as compression members. The factor of safety for a typical compression member varies between 1.67 and 1.92, or an average of 1.85 (which compares with 36/22, or 1.63, from which the loads were determined). This factor, coupled with the fact that reference loads for compressive failure are always below yield, demonstrates that the principle load carrying members will always be stressed below yield and therefore no yield deformation can take place. In the event that a tornado passes directly over the reactor building and removes the steel-framed part of the building above the operating floor, the capability to shut down the reactor will not be impaired.

Below the operating floor, the reactor building walls have been designed to accommodate tornado winds of 300 mph. The walls which provide the secondary containment are comprised of from 1.5-foot to 3-foot thick reinforced concrete. Additional protection for the primary containment is achieved by the 6.5-foot thick concrete shielding wall surrounding the drywell. The wind design, however, does not exceed the earthquake design requirements; therefore, the earthquake design governs. Since the earthquake design governs, the wall can accommodate tornado winds of up to 500 mph without exceeding normal stresses for 4000 psi concrete. The ultimate capability of the walls to withstand a tornado wind is based on allowing the reinforcing steel to approach a yield stress. At this point, the walls can accommodate a wind velocity of 860 mph.

The lower compartments of the reactor building are also constructed of reinforced concrete. Floor slabs are designed for live loads of 200 lbs/ft^2 to 1000 lbs/ft^2, and the walls above grade can accommodate exterior tornado wind velocities of 500 mph (equivalent to 908 lbs/ft^2) without exceeding normal stresses. The slabs were also designed for seismic loadings which would produce stresses equivalent to an 860-mph wind velocity or a pressure of 2980 lbs/ft^2 at the yield stress.

3.3.2.2.2 Surface Pressure Effects on the Control Room

The Unit 2/3 control room is a totally enclosed, heavy-walled, reinforced concrete and reinforced concrete masonry structure and has similar tornado resistance capabilities to the reactor building. In addition, the location of the control room is such that the west wall is protected by the entire turbine building for Units 2 and 3, and the east wall is protected by the Unit 1 turbine building. The north wall of the control room is protected by the north wall of the Unit 1 and Unit 2 turbine buildings, and the south wall of the control room is protected by the south wall of the Unit 1 and Unit 2 turbine buildings. The reinforced concrete sections of the Unit 2 turbine building are similar to the reactor building and can withstand tornado winds of 500 mph.

During the Unit 1 Control Room retirement modification work, additional reenforced concrete masonry walls along column line H, E, and approximately 14 feet east of column line 30 were added to separate the Control Room area from the new office area located in the east portion of the old Unit 1 Control Room. These new walls are designed for a tornado loading higher than capacity of the structural components in the Unit 1 Turbine Building.

Licensing documents for Dresden Units 2 and 3 show the opening between the Unit 2/3 and old Unit 1 control rooms and indicate that the Unit 1 turbine building provides tornado protection for the Unit 2/3 control room. The plant configuration does, in fact, satisfy this commitment. The extent of protection afforded by the Unit 1 turbine building is not a commitment defined in the licensing documents. Therefore, licensing documents do not address the issue of a postulated collapse of the Unit 1 turbine building adjacent to the opening into the Unit 2/3 control room.

The effect of design basis tornado missiles on this opening has been studied probabilistically and has been shown to be a noncredible event at a probability level of 1×10^{-7}. However, the effect of wind and debris resulting from structural collapse has not been studied. All control panels, etc., are located away from the control room walls so that any minor wall damage would have no effect on the control panels themselves.

The Unit 1 license does not include the provision for tornado design. However, a tornado magnitude at an equivalent probability to the safe shutdown earthquake can be postulated, and the Unit 1 turbine building structures have been shown to have a capacity in excess of this force.

In summary, the control room is an enclosed, reinforced, concrete and reinforced concrete masonry structure, having similar tornado resistance capabilities to the reactor building. The control room concrete structure is additionally protected by the Unit 2 turbine building on the west and the Unit 1 turbine building on the east. Therefore, the likelihood of adverse tornado effects on the safe operation of the Unit 2/3 control room is low.

3.3.2.2.3 Surface Pressure Effects on the Chimney

The ventilation chimney was originally investigated to assure that failure would not occur for tornado winds of 300 mph. For this analysis, consideration was given to the embedment of the chimney foundation within the rock strata. The maximum calculated bearing pressure in the rock generated from this analysis did not exceed 175 ksf. The unconfined ultimate compressive strength of the rock at elevation 500'-0" was found to be 3011 psi or 435 ksf. The stresses in the critical section of the chimney shell were determined to be 1700 psi compression in the concrete and 46,000 psi tension in the reinforcing steel. The calculated stresses for this condition were below the allowable stresses of 3200 psi and 54,000 psi in the concrete and steel respectively. Therefore, at a wind velocity of 300 mph, it was assumed that the chimney shell would remain intact.

Additional analyses were performed as a result of the Systematic Evaluation Program (SEP) for Unit 2. The SEP analyses were performed using the ACI ultimate strength method based on the ultimate strength of the reinforced steel used in the construction of the chimney. The chimney was calculated to fail at a tornado wind speed of 255 mph at elevation of 577'-0" (60 feet above grade) or higher. Despite the differences between the results of the original analysis and the SEP analyses, it was determined that modifications to the plant were not warranted. The justification for this determination was based upon the low probability of occurrence (1.35×10^{-6} per year) for a tornado wind speed exceeding 255 mph and of the assumed probability (1.35×10^{-7} per year) of the chimney collapsing and causing damage to safety-related equipment and structures.

Refer to Section 3.3.1.1.2 for additional information pertaining to the analysis of the chimney.

3.3.2.2.4 Pressure Drop Effects

Tornado data on pressure gradients is practically nonexistent, but the lowest recorded air pressure during a tornado occurred at St. Louis, Missouri, on May 27, 1896. This reading was 26.94 inches and was 2.42 inches lower than the pressure recorded at the same time at the weather bureau office seven blocks away. The 2.42-inch pressure differential is equivalent to a loading of 170 lbs/ft^2, which is less than the designed load. A loading of 170 lbs/ft^2 would be induced only by an instantaneous pressure drop. Since the travel time of a tornado at 40 mph would result in a pressure drop taking place in 0.5 seconds, the vacuum load is not expected to exceed 170 lbs/ft^2. The design of the reactor building concrete walls is based on a differential pressure of 900 lbs/ft^2 (6.3 psi) without exceeding the normal working stresses of the reinforcing steel or concrete. The design of the exterior concrete walls is such that the reinforcing steel quantities are the same in both the inside and outside faces. Therefore, the walls are capable of withstanding equal internal and external differential pressures. The building as a whole can withstand a significantly higher external pressure.

The reactor building superstructure is designed to accommodate a wind velocity of up to 170 mph, which is equivalent to a pressure of 75 lb/ft^2. When a wind pressure of 70 lb/ft^2 is reached, certain sections of the building superstructure siding will blow off. The panels are about 20 feet wide, extend the full height of the insulated siding, and are secured with notched "blow-off" bolts designed to break at the predetermined pressure. See Figure 6.2-43 for details on blow-off panels. The removal of these blow-off panels controls tornado damage and rapidly reduces the internal pressure by venting.

Of the approximately 38,200 square feet of insulated superstructure siding on the reactor building, approximately 5400 square feet is attached with the blow-off bolts on 6-inch centers. The remainder of the siding, of identical construction, is attached with self-tapping screws. The total quantity of insulated superstructure siding for the reactor and turbine buildings combined is approximately 138,500 square feet with approximately 16,300 square feet attached with blow-off bolts.

In a paper prepared for the American Meteorological Society, G.W. Reynolds[1] reports that a vent area of 1 square foot for every 1000 cubic feet of contained space should reduce the pressure differential to a safe level. The reactor building relief panel area provides approximately 4.8 square feet for every 1000 cubic feet of contained space for a safety ratio of almost 5:1. Tests have been made to verify the performance of these panels.

3.3.2.3 Effect of Failure of Structures or of Components Not Designed for Tornado Loads

Although the shutdown cooling and isolation condenser systems are not required for a safe shutdown (since they are fully backed up by the other core cooling systems), they are in protected areas. The isolation condenser system is within the reactor building, with the exception of various subsystems that can supply makeup water when needed. These subsystems are located in separate areas: the condensate transfer pumps are on the ground floor elevation near the center of the turbine building, the clean demineralized water transfer pumps are located on the ground floor of the turbine building east of the Unit 2 reactor feed pumps, the isolation condenser makeup water pumps are located in the isolation condenser pumphouse, and the fire pumps are located in the crib house approximately 8 feet below ground level. Likewise, the shutdown cooling system is located primarily in the protected reactor building, a Class I structure. The only parts of the system not in the reactor building are the service water pumps located in the crib house. Hence, it is extremely improbable that tornado damage could eliminate use of these systems which, to reiterate, are not required for a safe shutdown.

3.3.2.3.1 Turbine Building

The turbine building concrete deck units and insulated metal side panel upper story can be damaged by a tornado and shutdown capability can still be maintained.

3.3.2.3.2 Crib House and Radwaste Building

The effects of a tornado on the crib house and radwaste building are discussed briefly. The radwaste building is a labyrinth of concrete walls, all relatively thick for shielding purposes. The roof is at elevation 529'-6", only 12 feet above ground. Most of the equipment is below ground, including the high radiation level storage tanks. The solid waste, low radiation level drum storage is at ground level. The building has no effect on reactor shutdown, is highly resistant to wind loadings, and presents no problems due to missiles.

The radwaste building was evaluated under the SEP Topic III-4.A with respect to tornado generated missiles. Two cases were evaluated and it was determined that the radwaste building provides adequate tornado missile protection since the radwaste building walls are at least 12-inch thick reinforced concrete and only the upper 12 feet of the building are above ground.

The crib house is not necessary for safe shutdown of the reactor and, like the radwaste building, is a low, reinforced concrete structure. The diesel-driven fire pump is located in the center of the building below ground level and is protected by a concrete wall on the south, the service water pumps on the east and west, and the traveling screens on the north in addition to the concrete outer walls. The contaminated condensate storage tanks have no tornado protection, so they could be ruptured by either a differential pressure or a generated missile.

The crib house was evaluated under SEP Topic III-2. The diesel generator cooling water pumps are the only safety-related components in the crib house. Calculations performed by the NRC Staff indicated that these pumps are adequately protected from tornadoes.

3.3.2.3.3 Dresden Island Lock and Dam

It is not possible to assess the resistance of the Dresden Island Lock and Dam to tornado force winds, since the detailed design information is not available. In discussions with the Army Corps of Engineers, it appears that the brick control building for lock operation could be damaged in a tornado. However, any damage to the Dresden Island Lock and Dam due to a tornado will not affect the capability of safe reactor shutdown.

3.3.3 References

1. Reynolds, G.W., "Venting and Other Building Practices as Practical Means of Reducing Damage from Tornado Low Pressures," Bulletin of the American Meteorological Society, No. 1, Volume 39, January 1958.

3.4 WATER LEVEL (FLOOD) DESIGN

The Dresden site is not "flood dry," i.e., the site will be inundated by a probable maximum flood (PMF) event, and consequently is protected by structural or other measures (e.g., emergency procedures).

This section describes the Dresden flood protection features and major flood design evaluations. Section 3.4.1 summarizes the flood protection measures implemented for external and internal floods. Section 3.4.2 describes the evaluation methods used to determine the static and dynamic effects of flood loads for the drywell.

3.4.1 Flood Protection

The flood protection measures for Class I structures and components are described in terms of external or internal flood sources, as follows.

3.4.1.1 External Flood Protection Measures

Dresden's external flood control efforts are directed toward the prevention of damage resulting from the PMF of the Illinois and Kankakee rivers. The PMF, described in Section 2.4.3, produces a peak flood to elevation 528'-0" at the Dresden site. This is significantly above the grade elevation (517'-0") and the lowest opening leading to safety-related equipment which is at elevation 510'-4".

The structures for Units 2 and 3 have been analyzed to withstand hydrostatic loads, combined with other applicable loads, with no flooding of the interior of those structures, up to the PMF levels.

The current flood protection procedures mandate the installation of deployable flood barriers to protect the Reactor Building from flooding in order to support the hardened Containment Vent System (HCVS) venting capability. This strategy includes protecting the HPCI/U2/3 EDG Building. Additional flood protection measures have been implemented in order to protect the U2 and U3 EDG Rooms in the Turbine Building for asset protection.

These structures, as well as the associated flood barriers, have been qualified for applicable hydrostatic and hydrodynamic loads in combination with other applicable design loads up to the PMF level. Additionally, these structures have been analyzed to resist any floatation forces caused by the installation of temporary flood barriers.

The emergency protection measures available to minimize the impact of an external flood event are described below.

The existing flood protection procedure for a PMF makes optimum use of the onsite equipment to achieve a safe shutdown of the reactor. The procedure utilizes the reactor cooling systems to discharge decay heat throughout the term of the PMF.

Essential to the implementation of the flood procedure is the availability of reliable monitoring and the forecast of data pertaining to regional precipitation and water level rise. This information is provided by rainfall monitoring stations located throughout the Kankakee and Illinois river basins as well as by the National Weather Service and the firm of Murray and Trettle, EGC's meteorological consultants. The information provided from these sources, in conjunction with the computed PMF hydrograph, provides advance warning in the event of a PMF. Based on the critical-time hydrograph, the Illinois River can reach elevation 517.5 ft (plant grade) approximately 33 hours after the onset of rain. In the highly unlikely event that a PMF is predicted, the plant will shutdown in advance of the time predicted for flood stage occurrence, i.e., grade level.

The PMF flood procedure is implemented upon a forecast of river levels exceeding 506.5 feet. Operators then maintain a close watch on the Unit 2/3 intake canal water level, which, if predicted to exceed 510'-4", will trigger further operator actions. Two service water pump motors from Unit 2 will be moved to an elevation above the 530-foot level; the remaining such motors will be secured as time allows. The rationale behind relocating pump motors above the predicted PMF level is that they will be available for service, upon reinstallation, once the flood recedes.

Prior to water level reaching 510'-4" all reactors are shutdown, the drywells are deinerted, and the vessels are flooded. The reactors are cooled to the lowest legal temperature as quickly as possible.

If the water level reaches 513 feet at the plant site, cooling of the reactors will be transferred to the isolation condensers, which will thereafter maintain the primary system in a safe shutdown condition until the flood waters have receded and startup procedures can be initiated. A procedure specifies the instruments at a local instrument rack that can be used to monitor vessel pressure and level.

If forecasted flood levels exceed elevation 517'-0", concurrent with the operations described for core cooling, the following procedures are implemented. Motors and other electrical equipment, located at elevation 517'-0", below the PMF crest level, are deenergized. In addition, all above-ground water tanks are filled to a level of 10 feet above grade, and all below-ground water tanks are filled with demineralized water. Also, vents to below-ground diesel oil storage tanks will either be sealed or extended 25 feet above ground and a diesel-driven emergency flood pump is connected by hoses to a fire system header in each unit. Through these fire system headers, the emergency flood pump is capable of providing at least 175 gpm of flow to each unit. This flow is used for make-up to the shell of the isolation condensers, spent fuel pools, and the RPVs. A crosstie between the Fire Protection System also allows the 2/3 EDG cooling water discharge flow to serve as an alternate makeup water source for the isolation condensers, spent fuel pools and reactor pressure vessels. Other auxiliary equipment obtained upon prediction of a PMF includes gasoline-powered generators and motorized boats to facilitate operations in and around the site subsequent to deenergizing transformers at the time the water level reaches plant grade.

Once flood waters have receded, power from offsite sources is restored and a plant cleanup plan is established.

3.4.1.2 Internal Flood Protection Measures

Internal flood control efforts at Dresden are directed toward the protection of the Class I plant structures from internal sources that can conceivably introduce large amounts of water into below-grade areas. The consequences of flooding due to high energy line breaks, including the effects of a steam or feedwater line break in the steam tunnel, are discussed in Section 3.6.

The specifics of the internal flood protection measures are described below, and the locations of the protected systems and components are found in the general arrangement drawings of Section 1.2.

3.4.1.2.1 Protection of the Condensate Pump Room and Containment Cooling Service Water Pump Room

On June 7, 1972, a water hammer event occurred at Quad Cities Unit 1 that led to a failure of a rubber expansion joint in the 120-inch circulating water line and the subsequent flooding of the condenser pit and condensate pump room. As a result of this flooding, modifications were implemented at Dresden. This subsection describes these design changes.

Three design modifications were installed at Dresden:

A. The flood water paths from the condenser pit to the condensate pump room have been permanently sealed. This isolates the condenser pit and prevents any source of flood water in the condenser pit from becoming a source of flood water to the condensate pump room and consequently to the containment cooling service water (CCSW) pumps located on the floor above the condensate pump room.

B. Two of the four CCSW pumps per unit have been enclosed in a watertight vault. These vaults are designed to ensure that a postulated rupture in the CCSW system will not result in loss of all four pumps due to flooding.

C. The diesel generator cooling water pumps, which are located in the crib house pit, have been replaced with submersible-type pumps and motor drives. In the event that the crib house pit is flooded, the pumps can operate while submerged.

The design details of these modifications are discussed in the following subsections.

3.4.1.2.1.1 Design Modification of the Condensate Pump Room

The following modifications have been made to isolate the condenser pit from the condensate pump room in the event of a flood to elevation 508'-0":

A. The two ventilation openings between the condenser pit and the condensate pump pit have been permanently sealed by means of 1-inch thick steel plates.

   The ventilation opening seals have been analyzed and found acceptable to withstand a head of 30 feet of water in addition to a 0.12 g vertical acceleration as applied to the mass of water. The plate and structural members providing these seals undergo stresses which do not exceed material allowable working stresses increased by 33%.

B. The doorway between the condenser pit and the condensate pump room has been sealed by installing a watertight door. The watertight door has been designed to withstand a head of 30 feet of water plus the effects of a 0.18 g horizontal seismic load. Also included in the design parameters is the sloshing effect of the 30 feet head of water. The design analyses used the methods of TID-7024, Nuclear Reactors and Earthquakes. The door materials undergo stresses which do not exceed material allowable working stresses increased by 33%.

C. The piping and electrical penetrations between the condenser pit and the condensate pump room have been permanently sealed with concrete, silicone sealant, or rubber-boot-type seals to prevent leakage between the condenser pit and the condensate pump room in the event of a flood. These seals were designed for the maximum flood condition in relation to the elevation where each is located.

D. The wall between the condenser pit and the condensate pump room provides a watertight seal, in addition to the seals specified above. This wall has been analyzed and found capable of withstanding a head of 30 feet of water acting on the condenser side of the wall plus the effects of a 0.18 g horizontal and a 0.12 g vertical seismic occurrence. This includes sloshing effects as in Item B. The construction of the existing wall is more than adequate to sustain the water pressure and the effects of the seismic occurrence with stresses in the concrete and reinforcing steel not exceeding allowable working stresses increased by 33%.

   It is recognized that the flooding of the condenser pit is an additional load onto the foundation of the structure; however, the added weight is more than adequately supported directly on the founding rock through the concrete mat on which the structure rests.

E. The floor and equipment drains which run from the condenser pit to the sumps in the condensate pump room have been permanently sealed or sealed with removable plugs where periodic drainage is necessary. The seals and plugs have been designed to withstand the maximum flood level.

F. A system of level switches has been installed in the condenser pit to indicate and control the flooding of the condenser area. The following switches are installed:

Level Above Floor	Function
1. 1 foot (1 switch)	Alarm, condenser pit high water level
2. 3 feet (1 switch)	Alarm, condenser pit high circulating water level
3. 5 feet (2 redundant switch pairs)	Alarm, condenser pit high-high water level and circulating water pump trip

Level 1 indicates water in the condenser pit from either the hotwell or the circulating water system. Level 2 is above the hotwell capacity and indicates a probable circulating water failure. At the level 2 alarm, if the rate of rising level or observations indicate a circulating water system failure, the operator in the control room shall manually trip the circulating water pumps for the affected unit, thereby preventing continued flooding due to operation of the pumps.

Should the switches at levels 1 and 2 fail or the operator fail to trip the circulating water pumps on alarm at level 2, the actuation of either level switch at level 3 shall automatically trip the circulating water pumps of the affected unit and activate an alarm in the control room. These redundant level switch pairs at level 3 are designed and installed to IEEE 279.

3.4.1.2.1.2 Isolation of the Containment Cooling Service Water Pumps from Flood Water

Each unit at the Dresden Station has four CCSW pumps, any two of which can provide the required containment cooling. Each unit has an isolated vault installed around two of the pumps mentioned above. The vaults were constructed by building a new flood protection wall around the B and C CCSW pumps in each unit. These new walls are keyed into existing walls and slabs and are secured using drilled-in anchors.

The walls were designed and constructed to withstand a head of 13 feet of water on either side of the wall in addition to the combined horizontal effect of a 0.18 g seismic occurrence, vertical effect of a 0.12 g seismic occurrence, and a sloshing effect of the water per TID-7024, Nuclear Reactors and Earthquakes.

The walls are capable of withstanding the above mentioned combined loads with stresses not exceeding the material allowable working stresses increased by 33%. Stresses in these reinforced walls are such that no cracking, which would cause flooding of the CCSW pump vaults, is expected to occur.

Access into the CCSW pump vaults is by means of watertight, steel bulkhead or submarine-type doors. These doors are designed and constructed to the same flood and seismic criteria as the walls.

Pipe and small electrical conduit penetrations through the vault walls are sealed.

Electrical cables supplying power to the CCSW pumps are sealed.

Junction boxes connected to the conduit which penetrates the vault walls are sealed with a silicone rubber sealant to prevent leakage into the vaults.

Electrical instrumentation, controls and other components for the CCSW pumps, which are susceptible to water damage, have been located either inside the vaults or at an elevation above the flood level.

The flood protection vaults have floor drains and CCSW pump bed-plate drains.

The floor drain and CCSW pump bed-plate drains are used to drain the CCSW pumps and their associated vault coolers. Each bed-plate drain is routed to the floor slab drain.

As a means of preventing backflow from outside of the vaults in the event of a flood, a check valve and an air-operated control valve have been installed the vault floor drain. The control valve is open when energized and closed upon any of the following:

A. Loss of air,

B. High level in the condensate pump room,

Closure of the control valve on high water level in the condensate pump room is accomplished by use of a level switch located in the condensate pump room. Upon actuation of the swtich, the control valve will close and the alarm in the control room will activate.

3.4.1.2.1.3 Protection of the Diesel Generator Cooling Water Pumps

The three diesel generator cooling water pumps have been replaced with watertight, submersible canned-type pumps and motor drives, with capacity and head equal to that of the original pumps. The electrical conduit to theses pumps was also replaced with watertight conduit. The electrical motor control centers for these pumps are located in the turbine building and are not susceptible to damage from this type of internal flood as they are above the flood level.

3.4.1.2.2 Protection of the Emergency Core Cooling System, Drywell, and Torus

A number of emergency core cooling system (ECCS) motor-pump units are located on the lower levels of the reactor building. Since these units are vital to safe shutdown and containment in the event of an accident, steps have been taken to provide protection from flooding for these motor-pump units.

The torus, torus header, and associated piping systems are considered logical extensions of the primary containment, and as such, must meet the same design, surveillance, and testing criteria as the primary containment. The ring header and piping is constructed of heavy-walled pipe firmly supported from the lower portion of the suppression chamber. Maximum protection of the ring header and piping is provided by its physical location, i.e., adjacent to the suppression chamber in a reinforced concrete room containing no or mechanical equipment. The HPCI steam supply line is located in the Torus Room and the effect of high energy line breaks (HELBS) on the Torus and other safety-related systems has been considered. Pipe whip restraints are provided at the critical break locations to reduce the impingement loads as described in UFSAR Section 3.6.1.1. Because of the above design considerations, rupture of this low pressure system is not considered credible.

In order to remove postulated water leakage from valve stems, flanges, etc., the reactor building has been equipped with two floor drain sump pumps each with a capacity of 50 gal/min for a total removal capacity of 100 gal/min. Leakage of 100 gal/min corresponds to a system rupture equivalent to a 1-inch diameter hole. Two additional submersible sump pumps were installed by a modification to provide redundancy in the system. Excess operation of the sump pumps would also be noted in the radwaste facility. In addition, torus water level is constantly monitored in the control room and alarmed for departure from normal water level.

Additionally, submarine-type doors have been installed between the low pressure coolant injection (LPCI) rooms and the torus basement to provide a leaktight barrier in the event of flooding in the torus basement. Should the torus basement become flooded, it will be necessary to pump the water out of it. This is accomplished via a 2-inch pipe that has been installed in one of the submarine doors. These submarine-type doors are verified to be properly closed daily.

Due to the design of the drywell and torus, it is not possible to accumulate any large quantity of water under the reactor vessel. During a loss-of-coolant accident (LOCA) or similar event, the design permits the area under the vessel to drain into the torus via the downcomers. Without pumping, the sumps would overflow to the drywell floor, and once the water level reached the downcomers, water would flow into the torus. Since the torus level is monitored, this increase would be noted and investigated. Even under this scenario, the water would not reach a level where it could adversely affect any critical system.

3.4.2 Analytical Procedures

This section describes the analytical procedures by which the static and dynamic effects of the flood conditions are applied to safety-related structures, systems, and components.

3.4.2.1 Drywell

The drywell was analyzed to determine the additional effects due to seismic loads when it is in a flooded condition. From the analysis it was concluded that the natural period of vibration of the drywell is about 0.07 seconds for the empty condition and 0.12 seconds for the flooded condition. Thus, the resulting effect of hydrodynamics is to reduce the total seismic forces. Because of this, the effects of the dynamic response of the fluid in the drywell were conservatively neglected in the dynamic analysis employed for the drywell design. Refer to Sections 3.7 and 3.8 for more detailed information concerning the seismic analyses performed for the drywell.

3.5 MISSILE PROTECTION

This section describes the missile protection of applicable station structures, equipment, and systems. These components are protected from the effects of postulated missiles either by barriers or, in the case of redundant systems or components, by physical separation. The missile protection description provided below is presented in terms of the missile sources such as internally generated missiles, turbine missiles, missiles generated by natural phenomena, events near the site, or aircraft hazards.

3.5.1 Physical Separation Criteria

As appropriate, safety-related equipment is protected from missiles through basic station component arrangement such that, if equipment failure were to occur, redundant equipment would remain available to perform the safety function.

Electrical equipment and wiring for the primary containment isolation system (PCIS), the high pressure coolant injection (HPCI) system, the low pressure coolant injection (LPCI) system, the core spray (CS) system, and the automatic depressurization system (ADS) are segregated into at least two separate divisions such that in the event of a design basis accident, removal of decay heat from the core and isolation of the primary containment is assured. Separation requirements apply to control power and motive power for all systems concerned. Arrangement and/or protective barriers are designed such that no locally generated force or missile can destroy both of the redundant functions. In the absence of confirming analysis to support less stringent requirements for the systems named above, the following criteria were generally followed for initial plant design considerations.

A. In rooms or compartments having rotating heavy machinery (such as the main turbine generator, the reactor recirculating pump motor-generator sets, or the reactor feed pumps) or in rooms containing high-pressure feedwater piping or high-pressure steam lines (such as those that exist in the drywell and between the reactor and the turbine), a minimum separation of 20 feet or a 6-inch thick reinforced concrete wall (or equivalent) is required between trays containing cables of different divisions.

B. Any switchgear, panels, or instrument racks associated with two redundant safety systems and located in a missile-prone zone have a minimum horizontal separation of 20 feet or are separated by a protective wall (equivalent to a 6-inch thick reinforced concrete wall). The switchgear or equipment of redundant safety systems may be less than 20 feet apart if the two pieces of equipment are not in a straight line along a likely missile path.

C. In any compartment containing an operating crane, such as the turbine building main floor and the region above the reactor pressure vessel, there must be enough separation between trays containing cables of the two divisions such that a moving crane load cannot damage the cables of both divisions in a single accident.

Pipe whip restraints have been installed on high-energy lines outside the containment to prevent the lines from becoming a potential source of missiles. Pipe rupture outside of the containment is described in Section 3.6.1.

3.5.2 Internally Generated Missiles

Missile protection is given special consideration under assumed accident conditions. The following text summarizes the pertinent design considerations for protection against internally generated missiles.

The driving force for potential missiles within the primary containment boundary is assumed to come from the energy within the working fluid. In the case of a break in a pipe carrying liquid the maximum liquid velocity attainable at the break is 200 ft/s because of choking. Similarly, the velocity of fluid from a steam line break is limited to the critical velocity of 1500 ft/s at the break. The drag force of the fluid which propels any potential missile is proportional to the product of the density and the velocity squared. Even though the velocity of the steam exceeds that of the water, the even larger ratio of water density to steam density (at containment ambient conditions) means that projectiles originating from a water line will have a greater drag force applied and will therefore achieve a larger kinetic energy.

Consideration is given to the possibility of having missiles in the following forms:

A. Valve bonnets (large and small),

B. Valve stems,

C. Thermowells,

D. Vessel head bolts,

E. Instrument thimbles,

F. Nuts and bolts, and

G. Pieces of pipe.

Missiles originating from steam lines were neglected as being insignificant relative to missiles originating from liquid lines. All small missiles propelled by liquid were assumed to achieve and maintain, until impact, the maximum liquid velocity of 200 ft/s. This assumption is conservative because a missile, after being dislodged, requires a finite time for acceleration before it can approach a velocity of 200 ft/s. In addition, for missiles directed in a horizontal direction, there is a tendency for the missile, which is traveling slower than the driving jet, to fall out of the jet as it is acted upon by gravity. Therefore, the driving force acts for a shorter time and the missile achieves a lower maximum velocity.

Using the above conservative design criteria it was found that no small missiles (e.g., thermowells, small valve components, etc.) originating from the liquid lines would achieve sufficient energy to penetrate the drywell, nor was there sufficient strain energy in the pressure vessel head bolts to cause penetration.

The method of calculation used to determine the energy required to penetrate the containment shell was based on extensive tests conducted by the Stanford Research Institute. During these tests rod-shaped missiles (traveling at velocities that could possibly be produced within the drywell) were impacted against square steel plates having clamped edges.

The results of the tests have been described by the following expression for minimum energy per unit diameter of missile required for perforation of a steel plate:

$$E/D = U\,(0.344T^2 + 0.032T)$$

where:

E = critical kinetic energy required for penetration, ft-lb

D = diameter of missile, inches

U = ultimate tensile strength, psi

T = plate thickness, inches

The results of this equation are plotted for the various thicknesses of the drywell shell as shown in Figure 3.5-1.

The most serious potential missile appeared to be a dislodged valve bonnet originating from a recirculation loop valve. It was assumed that the face of the bonnet (35-inch diameter) was acted upon by the water jet and that the massive (3000-pound) bonnet-stem assembly impacted with the containment wall with the stem (4-inch diameter) making initial contact. This is a conservatively chosen event because it requires that all bolts holding the bonnet sever simultaneously, that the bonnet and stem move as a massive unit, and that the stem end (smallest impact area) strike the containment first.

Since the valve bonnet is so heavy, it would achieve a kinetic energy of 1,860,000 ft-lb if it were traveling at 200 ft/s. Therefore, a more refined calculation was necessary to show that a velocity of 200 ft/s is not actually attained. This calculation showed that the bonnet would have to accelerate for a distance of about 20 feet and reach a velocity slightly in excess of 35 ft/s in order to acquire the energy (60,000 ft-lb) necessary for the 4-inch diameter bonnet to penetrate the 0.75-inch thick containment.

It was determined from the arrangement of components within the drywell that, even though the recirculation valves are oriented such that a dislodged valve bonnet could strike the containment directly, there is either insufficient distance available between the stem and drywell to achieve the energy necessary to penetrate or the bonnet would be deflected by obstructions, hangers, or uneven failure of the bolting.

It has been shown in experiments conducted by Chicago Bridge & Iron Company (CB&I)[1] that large, slowly applied loads acting over an area of 1.08 square feet on a plate 0.75 inches thick would not cause cracking to develop until a deflection greater than 3 inches had occurred. Since the drywell shell is reinforced by concrete, located nominally 2.75 inches away, it appears improbable that objects having a large impact area will be able to penetrate the steel without also penetrating the concrete. Small missiles do not achieve a high enough velocity, based on the assumed design criteria, to attain an energy level sufficient to penetrate sound containment shell material. Therefore, it is concluded that missile penetration of the containment is a highly improbable event.

Consideration has been given to achieving missile protection where possible through basic plant component arrangement such that, if failure were to occur, the direction of flight of the missile would be away from the containment vessel. In addition to the care with which equipment is oriented with regard to missiles, special care has been taken in component arrangements to see that equipment associated with engineered safety features (ESFs), such as the core spray and the containment spray are segregated. The segregation is designed in such a manner that the failure of one system does not cause the failure of the other or that the failure of any component which necessitates use of these ESFs does not render the system inoperable.

Additionally, the control rod drive mechanisms are located in a concrete vault that provides protection from potential missiles. The suppression chamber has no source of internal or external missile generation, and the vent pipes connecting it to the drywell are protected by the jet deflectors. The vent discharge headers and piping are designed to withstand the jet reaction force caused by flow discharge into the suppression pool. See Section 6.2.1.3.5 for a discussion of the Mark I hydrodynamic loads analysis.

The NRC reviewed the systems and components needed to perform safety functions as a part of the Systematic Evaluation Program (SEP). This review included a comparison of the existing internal missile protection features to the design criteria for such protection. The NRC concluded that the design of protection from internally generated missiles meets the intent of General Design Criteria 4, Regulatory Guide 1.13 and Regulatory Guide 1.27.

3.5.3 Turbine Missiles

The information contained in the following sections provides a summary of the NRC's review and conclusions for SEP Topic III-4.B. The safety objective of SEP Topic III-4.B was to assure that Unit 2 structures, systems, and components important to safety are adequately protected from potential turbine missiles. The NRC did not review Unit 3 for protection from potential turbine missiles during the SEP process.

Each unit has a turbine that is comprised of one high-pressure element and three low-pressure elements. The low-pressure elements have eight wheels at each end, and each wheel carries a single row of buckets. The last stage buckets are the 38-inch type. For further information on the design of the turbines refer to Section 10.2.2.

The turbines are located on the main floor of the turbine building and are oriented in an east-west direction. The high-pressure end of the Unit 2 turbine faces the high-pressure end of the Unit 3 turbine. For more detail of the layout for the turbines refer to Drawings M-2 and M-7.

Fragments from low-pressure turbine wheels are considered to be the primary missiles produced during turbine failure. There are basically two modes of turbine failure that result in the ejection of missiles:

A. Design failure, caused by a flaw-induced failure of turbine wheel material at approximately the normal operating speed, and

B. Destructive overspeed failure, due to a failure of the overspeed control system.

With few exceptions, for other turbine failures, broken parts are relatively small and contained within the turbine casing.

Protection against overspeed failure is accomplished by three independent systems: a normal speed control, a primary and emergency overspeed trip. Each of these systems, how they interface, and their surveillance tests and inspections are discussed in Section 10.2.

The probability of unacceptable damage due to turbine missiles (P_4) is generally expressed as the product of (a) the probability of turbine failure resulting in the ejection of the turbine disk (or internal structure) fragments through the turbine casing (P_1), (b) the probability of ejected missiles perforating intervening barriers and striking safety-related structures, systems, or components (P_2), and (c) the probability of struck structures, systems, or components failing to perform their safety function (P_3). The probability of unacceptable damage from turbine missiles should be less than one chance in ten million per year for an individual plant, i.e., $P_4 < 10^{-7}$ per year.

3.5.3.1 Turbine Missile Targets

The original evaluations of turbine missiles focused on the probabilities that turbine missiles would strike safety-related equipment (P_2). However, the NRC staff made a complete shift of emphasis in the review of turbine missile evaluations, away from the review of strike probabilities to the review of missile generation probabilities (P_1).

Because of the uncertainties involved in calculating P_2, the NRC staff concluded that P_2 analyses are "ball park" or "order of magnitude" only. On the basis of simple estimates for a variety of plant layouts, the staff further concluded that the strike and damage probability product can be reasonably taken to fall in a characteristic narrow range which is dependent on the gross features of turbine generator orientation.

(1) For favorably oriented turbine generators, P_2 * P_3 tends to lie in the range of 10^{-4} to 10^{-3}
(2) For unfavorably oriented turbine generators P_2 * P_3 tends to lie in the range of 10^{-3} to 10^{-2}

For these reasons (and because of inadequate data, controversial assumptions, and modeling difficulties), in the evaluation of P_4, the staff gives credit for the product of the strike and damage probabilities of 10^{-3} for a favorably oriented turbine and 10^{-2} for an unfavorably oriented turbine, and does not encourage calculations of them. Both Dresden Unit 2 and Unit 3 have unfavorably oriented turbines.

3.5.3.1.1 Reactor Coolant Pressure Boundary

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3.5.3.1.2 Safe Shutdown Systems

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3.5.3.1.3 Sources of Radioactive Release

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3.5.3.2 Turbine Missile Evaluation

The original evaluations of turbine missiles focused on the probabilities that turbine missiles would strike safety-related equipment (P_2). However, the NRC staff made a complete shift of emphasis in the review of turbine missile evaluations, away from the review of strike probabilities to the review of missile generation probabilities (P_1). The new approach, which emphasizes turbine reliability, improves regulation of turbine generator systems reliability, reduces considerably the analytical burden placed on licensees and at the same time maintains the high level of protection of public health and safety.

The new approach places the burden of demonstrating turbine reliability on the turbine vendor. This shift of emphasis requires nuclear steam turbine manufacturers to develop and implement volumetric (ultrasonic) examination techniques suitable for in-service inspection of turbine disks and shaft, and to prepare reports for NRC review which describe their methods for determining turbine missile generation probabilities. These methods are to relate disk design, material properties, and in-service volumetric inspection interval to the destructive overspeed missile generation probability. Following vendor licensees tables of missile generation probabilities versus time (in-service volumetric disk inspection interval for rated speed or design overspeed failure, and in-service valve testing interval for destructive overspeed failure) for their particular turbine, which could then be used to establish inspection schedules which meet the NRC safety objectives.

Between 1986 and 1989, the original GE LP rotors with shrunk-on disks were replaced with welded LP rotors manufactured by Brown Boveri Corporation (BBC). The evaluation of missile generation probabilities for the welded LP-rotor design is contained in a BBC report entitled "Missile Analysis of the LP-Shaft for the Units Quad Cities/Dresden of CECO" 17.4.86/TGD. Because Dresden is considered an unfavorably oriented plant with a P_1 of 10^{-5}, the maximum volumetric inspection interval of the rotors, with respect to stress corrosion cracking, is 17 years. In the general inspection and overhaul plans, BBC recommends major rotor inspection intervals of 100,000 equivalent operating hours. The risk of stress corrosion cracking is therefore completely covered by the usual inspection and overhaul programs and no additional measures are required.

Based on a missile generation rate of 10^{-4} per year, the NRC estimated that total probability of unacceptable damage from low and high trajectory missiles is less than 10^{-6} per year.

The Unit 2 and Unit 3 LP rotors manufactured by BBC have been replaced with welded LP rotors manufactured by Alstom during 2011 and 2012. The evaluation of missile generation probabilities for these rotors is contained in the Alstom report STD0010156 "Quad Cities and Dresden LP Retrofit – Missile Analysis", dated August 28, 2009. The results of the evaluation demonstrated that for the intended LP rotor service period of 100,000 hours between major inspections, the calculated risk is lower than that calculated previously for the BBC LP rotors.

3.5.4 Missiles Generated by Natural Phenomena

Section 3.1.2 of the UFSAR contains the General Design Criteria as proposed by the NRC in July 1967. Dresden is a pre-General Design Criteria (pre-GDC) facility. The original design criteria for Dresden Units 2 and 3 addresses tornados generally, but not tornado missiles. Tornado parameters used in the design of structures at Dresden Station are addressed in UFSAR Section 3.3.2.1.

FSAR records dated July of 1969 provide the tornado missile characteristics to which the Reactor Building is designed and licensed. The original design is based on concrete ultimate stress. The following characteristics of missile types generated by natural phenomena such as high winds and tornados were identified in FSAR Section 12.1.2.1 for the Reactor Building.[15][16]

A. A telephone pole 35'-0" long having a velocity of 150 mph with a butt diameter of 13" and unit weight of 50 lbs. per cubic ft.

B. A one-ton mass having a velocity of 100 mph with a contact area of 25 sq. ft.

At the time of the original design and licensing, design criteria for non-horizontal missiles (i.e., missiles having a vertical velocity component due to free fall, updraft or other means) did not exist.[17][18] As such, the site is not committed to any specific criteria for non-horizontal missiles. The combined units DRE-2 and DRE-3 are designed for a single impact by either one of the two horizontal missile types listed above without a vertical component.[25][26] No other accidents or events are presumed to occur concurrent with a tornado event, and no caused single failure is postulated aside from the equipment damaged by a single postulated horizontal missile impact.[27]

An analysis of Dresden Unit 2 was performed by the NRC, as a part of the Systematic Evaluation Program (SEP) to determine the ability of certain structures to adequately protect the systems contained within them from postulated tornado missiles.[19][20] SEP evaluated the plant using beyond license input (360 mph wind instead of 300 mph) and acceptance criteria. The analysis involved new missile types capable of striking in all directions. These beyond design basis missile types had vertical speeds equal to 80% of horizontal speeds. The beyond design basis evaluation performed by Dresden engaged TORMIS calculations [21], which cannot be used in licensing applications without a License Amendment Request (LAR).[22] Since Dresden did not process a LAR, the beyond design basis evaluation and review performed by the NRC were conducted solely within the confines of SEP and do not affect Dresden's licenses.

The structures that were evaluated under SEP included:

A. The reactor building;

B. The turbine building;

C. The control room structure;

D. The radwaste buildings;

E. The diesel-generator buildings; and

F. The crib house (intake structure).

Under SEP, the NRC concluded that Dresden Unit 2 is adequately protected from the effects of tornado missiles. Further, the NRC concluded that Unit 2 meets the requirements of General Design Criteria 2 and 4 with respect to protection against tornado missiles.

As a result of the NRC's SEP review, the control room ventilation system was upgraded in response to NUREG-0737, "Clarification of TMI Action Plan Requirements," which states that all licensees should provide assurance that the habitability systems will operate under all postulated conditions. For the other systems of concern to the NRC, a tornado missile damage probabilistic analysis was performed for both Dresden units. The results of this analysis are discussed below.

The objectives of the SEP tornado missile hazard analysis were to determine the probabilities of the target loads, load characteristics, structural response effects, and to provide specific design provisions, if necessary, for reducing to acceptable levels the probability of damage occurring to essential structures. Regulatory criteria in place at the time of the analysis specified that, if the probability of damage to the plant structures and systems is greater than the order of 10^{-7} per reactor year, specific design provisions must be provided to reduce the estimate of unacceptable damage to about 10^{-7} per reactor year. The target loads that were analyzed included the following:

A. The exterior air intake and exhaust pipes and silencers of diesel generator (DG) 2/3 (Target 1). These pipes are located on the roof of the DG 2/3 and HPCI building.

B. The exterior air intake and exhaust pipes and silencers of DG 2 (Target 2). These pipes are located on the exterior south wall and roof of the Unit 2 turbine building.

C. The masonry wall No. 105 on column line 31 of the Unit 2 turbine building (Target 3).

D. The exterior wall area on column row C between the radwaste building and the turbine building (Target 4).

E. The crib house, containing water pumps and piping systems for cooling purposes, (Target 5).

The analysis used variance reduction techniques to achieve more realistic probabilities. By employing these techniques and the most severe tornado occurrence and strike data obtained through analysis of 29 years of tornado information[3] and the NRC tornado missile spectrum, the probabilities of tornado generated missiles striking these specific targets were conservatively estimated.

The probabilities of tornado missiles striking Target 1 and Target 2 were estimated to be 3.90×10^{-7} and 2.42×10^{-7} per year respectively (NRC estimates were 1.9×10^{-6} and 1.2×10^{-6} per year respectively).[13] The probabilities of hitting Targets 3 and 4 were much lower than 10^{-7} per year. The probability of a tornado missile striking and damaging the critical components within Target 5 was estimated to be 4.52×10^{-7} per year.

For Target 1 and Target 2, the postulated missile damage is to the air intake, exhaust and exhaust silencers, and not to the diesel generators themselves. CECo (EGC) has staged equipment to repair or remove damaged components to restore diesel generator operability, should damage to these components render the diesel generator inoperable.[13][14][23] Since the probability values, as estimated by CECo (EGC) for the five targets were of the order of 10^{-7} per year or lower, and provisions are in place should damage occur to the diesel generator air intake, exhaust and exhaust silencers, CECo (EGC) concluded, with NRC concurrence, that no design modifications were needed.[13][14][19][20]

3.5.4.1 Reactor Building Hatch Covers

The tornado pressure gradient criteria used in the design of the reactor building are such that no hatch cover will lift loose and act as a missile during venting due to differential pressure caused by tornadoes. That is, those hatch covers which could be lifted loose do not act as potential missiles. The lighter weight hatch covers are located over areas with a small volume of air, thus precluding large upward airflow from the covered area which could cause the hatch covers to become potential missiles. For larger hatch covers, sufficient venting is provided throughout the building in the form of stairwells, an elevator shaft, and open floor grating to preclude the possibility of lifting the covers and creating missile action. Furthermore, it would be physically impossible to lift some of the larger concrete hatch covers with the small differential pressures present.

3.5.4.2 Utility Poles and One-Ton Mass

The walls of the reactor building were analyzed (as discussed in Section 3.5.4) for the effect of these missiles and the analysis was based on ultimate stresses. The telephone poles were considered to have a perpendicular incidence at midpoint of a wall panel. Upon impact of the pole on the wall, compression waves are transmitted to the opposite face of the struck wall with a velocity equal to that of sound in reinforced concrete. Based on the analysis method defined in the NRC Standard Review Plan (SRP), Section 3.5.3, the FSAR tornado missiles will not penetrate or cause spalling of the exterior building walls.[3] Additional analysis indicate FSAR tornado missile types will not penetrate or cause spalling of concrete walls 12-inches thick.[24]

3.5.5 Missiles Generated by Events Near the Site

The potential for hazardous accidents in the vicinity of Dresden Unit 2 due to industrial, transportation, and military facilities is addressed in a report[4] under SEP Topic II-1.C. In that report, 12 sources of hazard are identified. Table 3.5-4 lists these sources of hazard. The possibility of missiles resulting from these sources is addressed in SEP Topic III-4.D and the NRC's review and conclusions for this SEP topic are summarized as follows:

A. It is concluded in SEP Topic II-1.C that explosions as a result of the first 5 sources listed in Table 3.5-4 do not constitute design basis events, since the safe distance criterion of Regulatory Guide 1.91 with (regard to blast overpressure) is met. Therefore, per Regulatory Guide 1.91, hazards from missile generation from sources 1 through 5 were not considered.

B. The explosions resulting from sources 6 and 7 may give rise to overpressure levels at the site in excess of 1 psi; however, missile generation from these explosions is not possible since the hazard is due to traveling clouds and no direct missile is expected from an explosion cloud.

C. Sources 8 through 10 in Table 3.5-4 do not clearly relate to missile generating events. They are included in the table to provide ease in cross-referencing information between the UFSAR and the SEP report.

D. Sources 11 and 12 in Table 3.5-4 relate to aircraft crashes into safety-related structures. The frequency of such occurrences, based on conservative assumptions, is calculated to be 4.16×10^{-7} per year. Therefore, these sources can be ruled out as potential missile hazards. Section 3.5.6 provides more detail on the aircraft hazards analysis.

Based on the results listed above, the NRC concluded that Dresden Unit 2 is considered acceptably safe against site-proximity missiles, including aircraft. For additional information on the industrial, transportation, and military facilities near the Dresden site refer to Section 2.2.

3.5.6 Aircraft Hazards

The probability per year of an aircraft crashing into the safety-related structures of Unit 2, P_A, was calculated using the methodology described in Reference 5 and summarized below:

$$P_A = \sum_{i=1}^{L} \sum_{j=1}^{M} \sum_{k=1}^{K} N_{ijk} \times A_{jk} \times R_{jk} \times D_k$$

where:

L	=	number of flight paths affecting the plant
M	=	number of different types of aircraft using the airport
K	=	number of aircraft maneuvers - i.e., landings or takeoffs
N_{ijk}	=	number of landings or takeoffs per year by the j^{th} aircraft along the i^{th} flight path
A_{jk}	=	effective plant area, in square miles, for the j^{th} aircraft in the k^{th} maneuver
R_{jk}	=	accident rate for the j^{th} aircraft for the k^{th} maneuver

D_k = distribution function characterizing the location of a crash relative to the runway being used, per square mile

The pertinent data for the probability calculation are given in Tables 2.2-4 and 3.5-5.

Since the nearby airports are used mostly by small aircraft, the number of different types of aircraft M was assumed to be equal to 1 for each airport. This is a conservative assumption because the accident rate per operation for small aircraft is higher than that for large aircraft.

The number of landings and takeoffs at Fromm and Adelmann airports are taken to be equally distributed along the two possible flight paths at each airport. The Morris[6] and Joliet[7] airport managers reported that about 25% or fewer of the operations are on the turf runway and 75% are on the asphalt runway at the airports. Furthermore, the Joliet turf runway handles only 5 to 10% of the total landing operations in the northeasterly direction. Therefore to obtain a more realistic distribution of the traffic along the possible flight path, the number of landings and takeoffs shown in Table 3.5-5 for Morris and Joliet airports were proportioned according to the ratios given above.

The effective area of the plant includes the actual plan area of the safety-related and nonsafety-related buildings housing safety-related equipment, a 45° shadow area to account for aircraft descent trajectories, and a 100-foot long skid area to account for aircraft crashing in front of the plant and skidding into these buildings. The effective area varies with the type of aircraft and flight path.

For conservatism, the maximum effective area was used in the calculations; this area corresponds to the north-to-south flight path, and its value when all three units are considered is calculated as 0.01684 square miles. The buildings considered in calculating the effective area are the reactor, turbine, and HPCI buildings of Units 1 (retired), 2, and 3; the crib house of Units 2 and 3, and the fuel handling buildings of Unit 1 (retired). The effective area of Unit 2, which was used in the calculations, is one-third of the plant effective area; i.e., 0.0056167 square miles.

The values of R_{jk} were evaluated by Vallance[8] using the general aviation accident data for the 5-year period 1964-1968. Because of improvements in technology and air traffic control, there has since been a decrease in the accident rate per aircraft movements. Assuming that the same rate of decrease holds for accident rates for landing and takeoff operations, and using 1966 as the base year for Vallance's[8] accident statistics, the accident rates for the year 1980 were derived [9] to be 0.9×10^{-6} and 2.4×10^{-6} per takeoff and landing, respectively. These accident rates were used in the calculations. The distribution function D_k used in the calculations was derived by Vallance[5] and is expressed as follows:

$$D_k = \frac{0.22}{r} e^{-r/2} e^{-\theta/80} \quad \text{for takeoff}$$

$$D_k = \frac{0.31}{r} e^{-r/2.5} e^{-\theta/43} \quad \text{for landing}$$

where:

r = distance from the end of the runway to the crash location, in miles

θ = angle between the extended runway centerline and a line from the end of the runway to the crash location, in degrees

As described in Section 2.2, there are four federal airways that pass within 10 miles of the station. As such, in accordance with SRP Section 3.5.1.6, the probability per year of a commercial aircraft traveling along one of these airways and crashing into the station (P_{FA}) was calculated. The following formula was utilized to determine P_{FA}:

P_{FA} = (C x N x A)/w
= $(4 \times 10^{-10} \times 2 \times 326 \times 365 \times 0.0056167)/9.21$
= 0.581×10^{-7} per year

where:

C = the in-flight crash rate per mile for commercial aircraft (constant 4×10^{-10})[10]

N = the number of flights per year along the airway[11]

A = the effective area of the plant in square miles (Table 3.5-5)

w = the width of the airway (plus twice the distance from the airway edge to the site when the site is outside the airway), in miles[11]

The present and projected probabilities of aircraft from the four airports and nearby airways crashing into Dresden Unit 2 are presented in Table 3.5-6. As shown in the table, the total projected probability of an aircraft crashing into Dresden 2 is 4.16×10^{-7} per year.

It cannot be determined if the "restricted landing area - seaplane base" was included in the local aircraft activity study used for the projected probability. On November 27, 1980, the Division of Aeronautics for the State of Illinois approved Michael S. Greenwald's application (see attached Figure 3.5-2).

Conservatisms included in these calculations are as follows:

A. The use of the maximum effective area of Unit 2.

B. The distribution function D_k and accident rate R_{uk} used for the Morris and Joliet airports are those derived for crashes within 5 miles of an airport. Use of these rates for Morris and Joliet, which are 8 miles and 10 miles from the station, respectively, is conservative.

C. The use of peak daily or monthly traffic from FAA-supplied documents to calculate the yearly number of operations.

D. VFR traffic was estimated to be equal to peak IFR traffic before the PATCO strike.

Based on these conservatisms and since the total probability (Table 3.5-6) of aircraft impact is within the acceptance criteria of SRP 3.5.1.6, it is concluded that aircraft operations in the vicinity of Dresden do not present an undue risk to the plant.

3.5.7 References

1. "Loads on Spherical Shells," by P. Thullen, Chicago Bridge & Iron Co., Oak Brook, Illinois, August 1964.

2. "Tornado Missile Simulation and Design Methodology, Vol. 2, Model Verification and Data Base Updates," NP-2005, by L.A. Twisdale and W.L. Dunn, Electric Power Research Institute, Palo Alto, California, August 1987.

3. Letter from P. A. Gazda, S&L Project Engineer to J.E. Hausman, SNED, dated September 12, 1986, Project No. 7592-00.

4. "SEP Topic II-1.C, Potential Hazards Due to Nearby Industrial, Transportation, and Military Facilities," CECo, December 1981.

5. "A Study of the Probability of an Aircraft Using Waukegan Memorial Airport Hitting the Zion Station," by J.M. Vallance, Pickard, Lowe, & Associates, Inc., April 7, 1972.

6. Telephone conversation between Morris Municipal Airport and J.A. Wilson, S&L, March 16, 1982.

7. Telephone conversations between E. White, Joliet Airport and J. A. Wilson, S&L, March 16 and 24, 1982.

8. "Supplement to a Study of the Probability of an Aircraft Using Waukegan Memorial Airport Hitting the Zion Station," by J. M. Vallance, Pickard, Lowe, and Associates, Inc., August 2, 1972.

9. "Report on ASCE Committee on Impactive and Impulsive Loads," Civil Engineering and Nuclear Power, Vol. V, September 15 to 17, 1980.

10. "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800, Section 3.5.1.6, Revision 2, July 1981.

11. Letter from G. C. Gunter, Chicago ARTCC, February 25, 1982.

12. Federal Aviation Administration Aviation Forecasts, 1981 to 1992.

13. Integrated Plant Safety Assessment Final Report, Systematic Evaluation Program, Dresden Nuclear Power Station, Unit 2, NUREG-0823, Supplement No. 1, October 1989, Section 2.2.2(2), Section 2.3.2 and Table 4.1 Item (14).

14. NRC Routine Inspection Report 91022, Submitted with Letter Dated September 12, 1991, Detail Item 8.

15. FSAR Volume II, Structures and Shielding, Chapter 12, Record Set Identifier Tab No. 1067-019, dated July 1, 1969, Section 12.1.2.1.

16. “Dresden Station Units 2 & 3 Updated Final Safety Analysis Report, Volume 1 Commonwealth Edison,” page 12.1.2-2. (The initial issue of the Updated FSAR is not dated but it is assumed to be one year prior to Revision 1, which is dated June 1983).

17. “Integrated Plant Safety Assessment Final Report, Systematic Evaluation Program, Dresden Nuclear Power Station, Unit 2,” NUREG-0823, February 1983, Sections 1.1 and 1.2.

18. NEI, Report APC 17-10, “Industry Position on Protection from Non-Horizontal Tornado-Generated Missiles,” March 8, 2017.

19. “Safety Evaluation Report related to the full-term operating license for Dresden Nuclear Power Station, Unit 2,” NUREG-1403, October 1990, Section 2, pages 2-1 & 2-2.

20. “Integrated Plant Safety Assessment Final Report, Systematic Evaluation Program, Dresden Nuclear Power Station, Unit 2,” NUREG-0823, February 1983, Section 4.3.1.

21. Dresden Station TORMIS Calculations used in support of SEP:
8.9.1-1, Revision 000.001, “SEP item III-4A Calc Pole and Rod,” October 22, 1981.
8.9.1-1, Revision 000.002, “Evaluation of RB & TB Outer Walls for Tornado-Generated Missiles,” May 22, 1986.
8.9.1-2, Revision 006, “Tornado Missile Hazard Analysis,” April 25, 1994.

22. “NRC Regulatory Issue Summary 2015-06 (RIS 2015-06) Tornado Missile Protection,” June 10, 2015, page 2 of ML15020A419.

23. Station Procedure DOA 0010-02, Revision 23; “Tornado Warning/Sever Winds.”

24. Calculation DRE17-0012, Revision 0; “Tornado Missile Protection (TMP) From Horizontal Impacts.”

25. EC 621047, Revision 1; “Vulnerability to Impact by a Single Tornado Missile at Dresden Station.”

26. NEI. APC 18-04 cover letter and Attachment, “NEI Industry Position on the Design Basis for Tornado-Generated Missile Penetrations Missile Protection ‘Single Missile’ and ‘Single Failure’,” dated 2/26/2018.

27. EC 619558, Revision 0; “Tornado/Wind Generated missile Vulnerability Evaluation Tornado Missile Protection (TMP) Project – Phase 2;” Appendix 1, page 13.

Table 3.5-4

SOURCES OF HAZARD IDENTIFIED FOR SEP TOPIC II-1.C
AND THEIR IMPLICATIONS FOR SITE PROXIMITY MISSILES

Case	Source of Potential Missile Hazard	Implication for Site Proximity Missiles
	Explosion from:	
1	Industrial facilities	Ruled out on the basis of the Regulatory Guide 1.91 standoff distance.
2	Highway transportation	
3	Railway transportation	
4	Waterway transportation	
5	Military facilities	
6	Pipeline	Ruled out because explosion would occur in the air.
7	Vapor cloud explosion from waterway transportation	
8	Toxic chemicals	Not related to missile generation.
9	Collision with intake structures	
10	Liquid spills	
	Aircraft impact from:	
11	Airports	Ruled out on the basis of low probability.
12	Airways	

Table 3.5-5

DATA FOR AIRCRAFT CRASH AND PROBABILITY ANALYSIS

Airport	Operating Mode	r (mi)	Theta (°)	D(r,Theta) (/mi²)	R x 10^{-6} (/operation)	N (operations/ yr)	A (mi²)	NARD x 10^{-7} (/yr)
Fromm	Landing	4.5	90	0.0014	2.4	150	0.0056167	0.02833
		4.5	90	0.0014	2.4	150	0.0056167	0.02833
	Takeoff	4.5	90	0.00167	0.9	150	0.0056167	0.01267
		4.5	90	0.00167	0.9	150	0.0056167	0.01267
Morris	Landing	8.0	25	0.000883	2.4	1,456	0.0056167	0.17333
		8.0	155	0.000043	2.4	1,456	0.0056167	0.00833
		8.0	65	0.00035	2.4	4,370	0.0056167	0.206
		8.0	115	0.00011	2.4	4,370	0.0056167	0.06467
	Takeoff	8.0	25	0.000369	0.9	1,456	0.0056167	0.027
		8.0	155	0.000073	0.9	1,456	0.0056167	0.00533
		8.0	65	0.00022	0.9	4,370	0.0056167	0.04867
		8.0	115	0.00012	0.9	4,370	0.0056167	0.02633
Joliet	Landing	10.0	10	0.00045	2.4	6,000	0.0056167	0.364
		10.0	170	0.000011	2.4	9,000	0.0056167	0.01333
		10.0	80	0.000088	2.4	22,500	0.0056167	0.26667
		10.0	100	0.000056	2.4	22,500	0.0056167	0.17
	Takeoff	10.0	10	0.00013	0.9	7,500	0.0056167	0.04933
		10.0	170	0.00018	0.9	7,500	0.0056167	0.00667
		10.0	80	0.000055	0.9	22,500	0.0056167	0.06267
		10.0	100	0.00056	0.9	22,500	0.0056167	0.049
Adelmann	Landing	1.0	115	0.01433	2.4	60	0.0056167	0.116
		0.9	80	0.0374	2.4	60	0.0056167	0.30233
	Takeoff	1.0	115	0.0317	0.9	60	0.0056167	0.0906
		0.9	80	0.05734	0.9	60	0.0056167	0.174

Table 3.5-6

PROBABILITY OF AIRCRAFT IMPACT FROM AIRPORTS AND AIRWAYS WITHIN 10 MILES OF DRESDEN STATION

Source	Present Probability of Impact per Year	Projected[1] Probability of Impact per Year
Fromm Airport	0.082×10^{-7}	0.082×10^{-7}
Morris Airport	0.560×10^{-7}	0.896×10^{-7}
Joliet Airport	0.982×10^{-7}	1.571×10^{-7}
Adelmann Airstrip	0.683×10^{-7}	0.683×10^{-7}
Airways	0.58×10^{-7}	0.929×10^{-7}
TOTAL	2.887×10^{-7}	4.161×10^{-7}

Note:

1. Air traffic projections are based on the forecasts given in Reference 12.

3.6 PROTECTION AGAINST DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING

This section describes design bases and design measures that ensure that the primary containment, secondary containment, and all essential equipment inside or outside primary containment (including components of the reactor coolant pressure boundary) are adequately protected against the effects of blowdown jet forces, reactive forces, and pipe whip resulting from the postulated rupture of piping located either inside or outside primary containment. Sections 3.6.1 and 3.6.2 address postulated pipe failures outside and inside primary containment, respectively.

3.6.1 Postulated Piping Failures in Fluid Systems Outside of Containment

3.6.1.1 High Energy Piping

In December 1972, the AEC issued letters to the licensees of all operating nuclear power plants, requiring reviews of the effects of piping failures outside of the primary containment structure. The letter applicable to Dresden[1] referenced General Design Criterion 4 of 10 CFR 50 Appendix A, as well as the previous version of this criterion, as the basis for the review detailed in an attachment to the letter. This attachment contained 21 items for which the licensee was required to provide detailed information. The final version of the CECo report responding to this request, Special Report No. 37, (Reference 8).

The following paragraphs summarize the CECo report and the NRC's evaluation[2] of the report. The evaluation of high energy line breaks (HELBs) outside containment was reviewed and accepted by the NRC for Dresden Units 2 and 3. For Unit 2, this topic was also reviewed under systematic evaluation program (SEP) Topic III-5.B.[3]

3.6.1.1.1 Criteria

A summary of the criteria and requirements included in the AEC letter of December 14, 1972,[1] is set forth in the following subsections.

3.6.1.1.1.1 Postulated Line Breaks

Protection of equipment and structures necessary to shut down the reactor and maintain it in a safe shutdown condition (assuming a concurrent and unrelated single active failure of protected equipment) should be provided from all effects resulting from ruptures in pipes carrying high energy fluid, where the temperature or pressure conditions of the fluid exceed 200°F and 275 psig, respectively, including the double-ended rupture of the largest pipe in the main steam and feedwater systems. The rupture effects to be considered include pipe whip, structural (including the effects of jet impingement), and environmental.

The original basis for the selection of break locations was to postulate breaks at terminal ends and at fittings selected on a reasonable basis to provide protection. The criteria used to determine design basis piping break locations for subsequent evaluations which are based on detailed piping stress analysis, utilized the following criteria:

A. ASME Section III, Class I piping breaks are postulated to occur at the following locations in each piping run or branch run:

1. The terminal ends;

2. Any intermediate locations between terminal ends where the primary plus secondary stress intensities S_n (circumferential or longitudinal) derived on an elastically calculated basis under the loadings associated with one-half safe shutdown earthquake and operational plant conditions exceed 2.0 S_M for ferritic steel or 2.4 S_M for austenitic steel (where S_M is the design stress intensity as specified in ASME Section III);

3. Any intermediate locations between terminal ends where the cumulative usage factor (U) derived from the piping fatigue analysis (based on all normal, upset, and testing plant conditions) exceeds 0.1; and

4. At intermediate locations in addition to those determined by Items 2 and 3 above, selected on a reasonable basis as necessary to provide protection. As a minimum, there should be two intermediate locations for each piping run or branch run.

B. ASME Section III, Code Class 2 and 3 piping breaks should be postulated to occur at the following locations in each piping run or branch run;

1. The terminal ends;

2. Any intermediate locations between terminal ends where either the circumferential or longitudinal stresses derived on an elastically calculated basis under the loadings associated with seismic events and operational plant conditions exceed 0.8 (S_h + S_A) or the expansion stresses exceed 0.8 S_A (where S_h is the stress calculated by the rules of NC-3600 and ND-3600 for Class 2 and 3 components, respectively, of the ASME Section III Winter 1972 Addenda, and S_A is the allowable stress range for expansion stress calculated by the rules of NC-3600 of the ASME Section III or the USAS Code for Pressure Piping, ANSI B31.1-1967); and

3. Intermediate locations in addition to those determined by Item 2 above, selected on a reasonable basis as necessary to provide protection. As a minimum, there should be two intermediate locations for each piping run or branch run.

3.6.1.1.1.2 Postulated Pipe Cracks

Protection of equipment and structures necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming a concurrent and unrelated single active failure of protected equipment, should be provided against the environmental and structural effects (including the effects of jet impingement) resulting from a single open crack at the most adverse location in pipes carrying fluid routed in the vicinity of this equipment. The size of the cracks should be assumed to be one-half the pipe diameter in length and one-half the wall thickness in width (defined as "critical crack size").

3.6.1.1.2 High Energy Systems

For pipe breaks outside primary containment, the following systems were analyzed:

A. Main steam;

B. Feedwater;

C. High pressure coolant injection (HPCI);

D. Reactor water cleanup (RWCU); and

E. Isolation condenser.

Break locations were selected at all elbows, terminal ends, and at least two intermediate locations between terminal ends in each high energy piping run, in accordance with the above NRC criteria. Only those break locations where significant effects on structures, components, and equipment are expected were analyzed.

The following systems were considered but not analyzed due to physical separation or absence of impact on equipment important to safety:

A. Extraction steam piping to heaters A, B, C, or D;

B. Heater drain piping from heaters C or D;

C. Condensate booster piping;

D. Moisture separator drain piping; and

E. Control rod drive (CRD) hydraulic piping.

3.6.1.1.3 Areas or Systems Affected by High Energy Line Breaks

An evaluation was conducted, as documented in Special Report No. 37, of the effects of HELBs on the following systems, components, and structures which would be necessary (in various combinations, depending on the effects of the break) to safely shut down the reactor and to maintain cold shutdown conditions.

3.6.1.1.3.1 Systems and Components

For pipe breaks outside primary containment, the following systems were analyzed:

A. Control and instrument cables and raceways;

B. Electrical distribution system;

C. Emergency dc power supply (batteries);

D. Emergency ac power supply (diesels);

E. Heating and ventilation systems (needed for long-term occupancy to maintain the reactor in a safe shutdown condition);

F. Reactor control systems and associated instrumentation;

G. Cooling and service water systems; and

H. Emergency core cooling system (ECCS) components.

3.6.1.1.3.2 Structures

The following structures, or portions thereof, which could be affected by the postulated breaks, were evaluated for pipe break protection outside containment:

A. Primary containment and torus;

B. Reactor building;

C. Main steam tunnel;

D. Control room;

E. Vital electric load centers and switchgear rooms;

F. Diesel-generator rooms; and

G. Auxiliary equipment rooms.

3.6.1.1.4 Specific Areas of Concern

Commonwealth Edison Company Special Report No. 37 (Reference 8) provided the results from examination of all postulated HELB locations outside primary containment and evaluated the break consequences. The NRC reviewed all of this information, including specific areas of concern such as compartment pressurization, pipe whip, compartment flooding, environmental effects, and control room habitability, where the potential consequences might be severe or where specific corrective action would further assure safe cold shutdown of the plant. Unless otherwise stated, the following information applies to both Units 2 and 3.

3.6.1.1.4.1 Compartment Pressurization

Large line breaks, including the double-ended rupture of the largest lines in a system and pipe cracks up to the critical size defined in Section 3.6.1.1.1.2, were considered for pipes in the main steam tunnel, outside primary containment, and in the turbine building. The compartment pressurization calculations included pressure plus impingement forces.

Each of the station's steam tunnels is divided into 2 compartments by a slab fitted with blowout panels which function to equalize pressure when a 2-psi differential pressure exists between the compartments. For each steam tunnel, a simultaneous rupture of one main steam line and two adjacent feedwater lines was assumed. Main steam isolation valve (MSIV) closure was assumed to occur 5.5 seconds after the rupture. For this case, the maximum steam tunnel pressure calculated was 20.9 psia. The tunnel walls can easily withstand this transient. However, the forces generated in such a transient could damage the blowout panels which could cause subsequent damage to cable trays located in the upper tunnel compartment. These cable trays carry safety-related cabling. The main steam line circumferential and longitudinal break points, identified in Table 12 of Reference 8, are those which could produce such damage. To prevent this damage to safety-related equipment, CECo provided the following improved support scheme for the blowout panels:

A. Installing an additional W36 x 135 beam on top of the existing W24 on the east-west wall of the tunnel, 11 feet south of column row "G";

B. Replacing the existing 6B x 12 north-south beams supporting the panels with five W21 x 55 beams; and

C. Upgrading the chains restraining the blowout panels.

In the reactor building, the consequences of HELBs and cracks in the HPCI, RWCU, and isolation condenser systems were evaluated. Damage to the torus could occur as a result of certain HPCI pipe longitudinal breaks identified in Table 13 of Reference 8. Commonwealth Edison Company provided pipe restraints at the critical break points to reduce the impingement loads. These restraints, which are anchored to the nearest structure, consist of U-shaped plates covering the break points.

Pressure calculations for the turbine building produced no areas of concern with respect to safety-related equipment.

3.6.1.1.4.2 Pipe Whip

The effects of pipe whip on structure walls and safety-related components were calculated for main steam and feedwater system pipe breaks in the steam tunnel; for HPCI, RWCU, and isolation condenser system line breaks in the reactor building; and for main steam and feedwater system pipe breaks in the turbine building. This evaluation included the double-ended rupture of the largest pipe in the main steam and feedwater systems. Break points were chosen in accordance with the guidelines identified in Section 3.6.1.1.1.1. Pipe whip calculations included pressure and impingement forces.

In the steam tunnel, whipping main steam pipes could cause damage to the blowout panels similar to that caused by pressure and impingement forces alone (see Section 3.6.1.1.4.1). Damage to safety-related cabling could occur as a result of circumferential breaks in the main steam system pipes indicated in Table 16, Reference 8. The modifications installed to mitigate the consequences of the pressure transient would also serve to mitigate the consequences of pipe whip.

In the reactor building, whipping HPCI piping, resulting from circumferential line breaks indicated in Table 17 of Reference 8, could damage the torus or the low pressure coolant injection (LPCI) valve operator (MO-2-1501-20A), both of which serve safety-related functions. The U-shaped restraints (described in Section 3.6.1.1.4.1) installed at the critical break points serve to mitigate the consequences of the pipe whip transient.

In the Unit 2 turbine building only, a whipping feedwater pipe, resulting from circumferential breaks indicated in Table 16 of Reference 8, could damage the wall adjacent to the emergency diesel generator. Subsequent damage to the diesel could occur. To prevent such damage, a frame was constructed of cross-braced columns anchored to the floor.

Postulated pipe whip calculations for HELBs in the Unit 3 turbine building produced no areas of concern with respect to safety-related equipment.

3.6.1.1.4.3 Compartment Flooding

Commonwealth Edison Company determined the effects of flooding for steam or feedwater line breaks in the steam tunnel, reactor building, and turbine building. Although the MSIV controls could be short circuited, should the steam tunnel fill with water, these valves would fail in the shut position and safe shutdown would not be impaired. No other safety-related equipment or wiring would be endangered by potential flooding caused by main steam, feedwater, HPCI, RWCU or isolation condenser system pipe breaks. The effects of flooding from loss of piping integrity, for other than the main steam and feedwater lines, and other internal sources, are covered in Section 3.4.

3.6.1.1.4.4 Environmental Effects

Electrical equipment was checked for possible adverse environmental effects which could be caused by HELBs. Adverse temperature, pressure, and humidity were the parameters which were used in the evaluation of safety-related equipment.

Commonwealth Edison Company assessed the consequences of environmental effects on safety-related electrical equipment. The equipment had been designed to limits in excess of postulated conditions which could arise from the HELB.

The environmental qualification of electrical equipment is covered in Section 3.11.

3.6.1.1.4.5 Control Room Habitability

The main control room is physically located away from and isolated from all high energy lines. Neither the control room equipment nor its ventilation system would be affected by environmental effects caused by a HELB. The control room would be habitable in the event of a HELB outside primary containment. Control room habitability is discussed in more detail in Section 6.4.

3.6.1.2 Control Rod Drive Hydraulic System Scram Discharge Piping

Piping in the scram discharge portion of the CRD hydraulic system was investigated in detail following the 1980 Browns Ferry 3 failure-to-scram event, as required in NUREG-0803.[4] One concern resulting from that effort was the potential for an unisolated rupture of this piping. This piping, however, is excluded from consideration for HELB because the probability of the rupture of this piping resulting in a loss-of-coolant accident is of such a small magnitude that the event is beyond the range of a credible occurrence. Disposition of this issue is covered in Section 4.6.4.

3.6.1.3 Instrument Line Break Outside Primary Containment

Following a review of the FSAR, the AEC raised several questions concerning instrument lines which originate at the reactor coolant pressure boundary and terminate outside primary containment. Since the main issues addressed were secondary containment integrity and potential offsite radiological consequences, these subjects are covered in Sections 6.2.3 and 15.6.2, respectively.

3.6.1.4 RBCCW Moderate Energy Line Break

In a draft evaluation of Systematic Evaluation Program (SEP) Topic III-5.B "Pipe Break Outside of Containment" dated January 17, 1980, the NRC identified a potential deviation from the criteria for postulated pipe breaks outside of containment. The NRC concern was that a properly oriented MELB in the RBCCW system could cause flooding of the redundant 4 kV switchgear 23-1 and 24-1 (on the 545' elevation of the reactor building) to the level of the curb surrounding the switchgear – about 5". If water rose above the curb, some of the breaker auxiliary components would be submerged and subject to failure.

The concern was evaluated and the results are documented in a letter to the NRC dated July 16, 1982. The flow rate from the MELB in 24 inch diameter RBCCW pipe was calculated at 623 gpm based on circular opening in the pipe of an area equal to that of a rectangle with one side equal to one-half the pipe diameter and the other side equal to one-half of the pipe thickness. At least 300 gpm will be collected by floor drains on elevation 545'-6". The remaining will flow through floor openings to elevation 517'-6". Nearly all the water reaching elevation 517'-6" will flow into floor drains on that elevation. A negligible amount of water will flow through floor openings on elevation 517'-6" to elevation 476'-6". No accumulation of water is expected on any elevation.

MCC's 28-1, 28-7, 29-1, 29-4, and 29-7 are located on elevation 517'-6". These MCC's are mounted on low pads but are not surrounded by curbing. However, since no water will accumulate on elevation 517'-6", the only possible hazard to an MCC is water falling directly onto the MCC from a floor opening above.

Only MCC 28-1 is under floor openings that could allow water to fall on it. Loss of MCC 28-1 would disable operation of the Isolation Condenser valves and disable safety systems powered by Division I. However, Division II would be unaffected and safe shutdown could be reached with the Division II HPCI and LPCI systems and the ADS.

Based on the above analysis, the NRC staff concluded in a Safety Evaluation transmitted on August 20, 1982 that the effects of a leak in the RBCCW system would not prevent reaching a safe shutdown condition.

3.6.1.5 Use of Isolation Condenser and Control Rod Drive Systems for Safe Shutdown Following a HELB

During the Systematic Evaluation Program in 1982, the NRC identified concerns with a two separate HELBs postulated in the feedwater system. A draft NRC Safety Evaluation for SEP Topic III-5.B – Pipe Break Outside of Containment (Reference 1) describes the following situations on page 8 of the conclusions section.

a. The consequences of a potential main feed system line break in the main feed regulatory valve area (turbine building 538' elev.) have not been determined with regard to possible damage to the engineered safeguards system electrical cables in that area. The licensee will be required to evaluate the consequences of a possible feed line break in this area and provide adequate protection as required.

b. A main feed system line break in the main feed pump area (turbine building 517' elev.) could damage the bus duct connecting diesel generator #2 to the 4 kV switchgear in the reactor building. Current criteria postulate a loss of offsite power (caused by a unit trip on loss of feedwater) and a potential single active failure of the #2/3 diesel generator. This scenario would result in loss of all AC power to the unit.

These issues were resolved in the following manner:

a. Commonwealth Edison provided a response to item a. above in a letter to the NRC dated July 16, 1982 (Reference 2). Commonwealth Edison stated that safe shutdown could be achieved using:

- CRD 2B pump for reactor vessel inventory control (including several manual operators actions to 1) start the Unit 2 diesel, 2) control the 4kV feed breakers, and, 3) transfer Division II 125 VDC control power from the main to reserve supply.
- A proposed CRD cross-tie from the other unit if the CRD 2B pump was assumed as the single failure
- Isolation condenser for decay heat removal including a manual action to reopen the inside containment isolation valves by jumpering contacts at the motor control centers if these valves spuriously closed.

The NRC accepted the above safe shutdown approach in a letter dated August 20, 1982 that closed out the open items from the draft Safety Evaluation on Topic III-5.B (Reference 3).

b. This item was resolved as discussed on pages 8 and 9 of the conclusions section in the draft NRC Safety Evaluation for SEP Topic III-5.B – Pipe Break Outside of Containment (Reference 1). Although item b. results in the total loss of AC power on the site, the likelihood of a unit trip resulting in the loss of offsite power is extremely small based on the operational history of this plant. Thus, AC power would be available to operate the safe shutdown systems, which were identified in the evaluation section of this report. Nevertheless, should a loss of offsite power be postulated, the isolation condenser would be available to remove reactor decay heat and commence a plant cooldown. In this case, makeup water to the isolation condenser would be provided by the fire protection system. The twenty minutes of shell side water in the condenser provides adequate time to start a diesel driven fire pump and assure a makeup flow path from the fire system to the isolation condenser. Decay heat removal using the isolation condenser can be maintained while repairs are made on the diesel generator which was assumed to fail (single active failure) or until offsite power is regained.

The above situations closely parallel a HPCI HELB in the HPCI Room, which could create a harsh environment in the Unit 2/3 diesel generator room. Both of the above feedwater HELBs result in the inability to use the ECCSs. The NRC accepted the isolation condenser and control rod drive systems under these circumstances as alternatives to the safety-related means of decay heat removal and reactor coolant inventory control using ADS and LPCI/CCSW. Therefore, in lieu of environmentally qualifying the Unit 2/3 diesel generator, the isolation condenser and control rod drive systems are credited as the means to safely shutdown following a HPCI HELB in the HPCI Room.

Since the NRC Safety Evaluation for SEP was issued, additional plant equipment has been installed that further increases the reliability of the isolation condenser and other mitigating equipment. First and foremost are diesel-driven isolation condenser make-up pumps that can supply clean demineralized condensate to the isolation condenser shells in lieu of the diesel-driven fire pumps. Second, diesel-generators installed for Station Blackout provide another means of powering the ECCS equipment. And last, the CRD cross-tie between units has been installed and its use is proceduralized in Dresden Safe Shutdown Procedures (DSSPs) and Emergency Operating Procedures (DEOPs).

3.6.2 Postulated Piping Failures in Fluid Systems Inside Primary Containment

Postulated pipe break interactions inside containment for Dresden Unit 2 were evaluated under SEP Topic III-5.A,[3] and later accepted by the NRC.[5] The final report submitted by CECo is contained in Reference 9. The following subsections briefly summarize the final report.

The discussion provided is also applicable to Dresden Unit 3.

This section addresses structural loading effects (pipe whip and jet impingement) resulting from postulated pipe breaks and cracks inside containment. Environmental considerations such as pressure, temperature, humidity, and flooding are covered in Section 3.11.

3.6.2.1 Criteria

The procedure for postulating break locations on a mechanistic basis in high and moderate energy piping systems inside the drywell is presented in this section. The approach reflects the guidance provided by the NRC Branch Technical Position MEB 3-1.[6]

3.6.2.1.1 Postulated Line Breaks

For the discussion which follows, a high energy piping system is defined as any system normally operating above the temperature or pressure limits presented in Section 3.6.1.1.1.1. Those piping systems that operate above these limits for only a relatively short portion (less than approximately 2%) of the period of time to perform their intended function are excluded from evaluation.

Breaks in piping systems inside the containment were to be postulated at the following locations in each piping (main or branch) run:

A. At terminal ends.

B. At intermediate locations selected by one of the following criteria:

1. At each pipe fitting (e.g., elbow, tee, cross, flange, and nonstandard fitting), welded attachment, and valve; or at a single location where the piping contains no fittings, welded attachments, or valves.

2. At each location where the stresses exceed 0.8 $(1.2S_h + S_A)$ but at no fewer than two separated locations chosen on the basis of highest stress. Where the piping consists of a straight run without fittings, welded attachment, or valves, and all stresses are below 0.8 $(1.2S_h + S_A)$, at a minimum, one location is chosen on the basis of highest stress. Two locations are selected with at least 10% difference in stress or, if stresses differ by less than 10%, two locations separated by a change of direction of the pipe run.

Circumferential breaks are postulated in fluid system piping and branch runs exceeding a nominal pipe size of 2½ inches, except where the maximum stress range exceeds the limit of 0.8 $(1.2S_h + S_A)$ but the circumferential stress range is at least 1.5 times the axial stress range.

Where break locations are selected without the benefit of stress calculations, breaks are postulated at the piping welds to each fitting, valve, or welded attachment. Alternatively, a single break location at the section of maximum stress range may be selected as determined by detailed stress analyses or tests on a pipe fitting.

Longitudinal breaks in fluid system piping and branch runs are postulated in nominal pipe sizes 4 inches and larger, except where the maximum stress range exceeds the limit 0.8 $(1.2S_h + S_A)$ but the axial stress range is at least 1.5 times the circumferential stress range.

Longitudinal breaks need not be postulated at terminal ends or at intermediate locations where the criterion for a minimum number of break locations must be satisfied.

3.6.2.1.2 Postulated Line Cracks

For the discussion which follows, a moderate energy piping system is defined as any system or portion of system where neither the maximum operating pressure exceeds 275 psig nor the maximum operating temperature exceeds 200°F during normal plant operating conditions.

The effects of moderate energy fluid systems were evaluated using the following criteria:

A. For piping systems that by plant arrangement and layout are isolated, physically separated, and remotely located from systems and components important to safety, through-wall leakage cracks need not be postulated;

B. For piping systems that are located in the same areas as high energy fluid systems which have postulated pipe break locations, through-wall leakage cracks need not be postulated;

C. For piping systems that are located in areas containing systems and components important to safety, but where no high energy fluid systems are present, through-wall leakage cracks should be postulated at the most adverse location to evaluate the effects of the resulting water spray and flooding; and

D. For those portions of piping from containment wall, to and including the inboard or outboard isolation valves, leakage cracks are postulated.

3.6.2.2 High Energy and Moderate Energy Systems

High energy systems, as defined in Section 3.6.2.1.1, inside the containment are as follows:

A. Core spray system;

B. Control rod drive system;

C. Feedwater system;

D. High pressure coolant injection system;

E. Isolation condenser system;

F. Low pressure coolant injection system;

G. Reactor recirculation system;

H. Reactor water cleanup system;

I. Shutdown cooling system; and

J. Main steam system.

Safety relief valve discharge piping and those segments of the core spray system beyond the normally closed valves are excluded from consideration since they operate less than 2% of the time.

The moderate energy systems inside containment greater than 2½ inches in diameter are the drywell equipment and floor drain discharge system, and the reactor building closed cooling water system.

3.6.2.3 Interaction with Structures and Components

Potential targets of pipe whip or jet impingement were identified from among all safety-related structures, systems, or components in the vicinity of postulated breaks.

Evaluation of interactions from postulated breaks was initially performed using simplified generic methods (generally by equivalent static structural analysis).

Interactions unresolved by generic target evaluations were all qualified on the basis of one of the following approaches:

A. Further system evaluation of electrical interactions;

B. Sophisticated nonlinear, dynamic (time-history) finite element analysis for pipe whip interactions with containment liner, reactor pressure vessel pedestal, and piping components; or

C. Rigorous and more refined development of jet pressure loading for jet interaction.

The results of the evaluations summarized in the following sections identified no potentially unacceptable break locations.

3.6.2.3.1 Drywell Shell

Criteria, design methods and evaluation results are detailed in this section for original pipe rupture protection of the drywell shell (including penetrations).

Pipe whip criteria are implemented in the design of various engineered safety features with particular emphasis on ECCS piping and instrumentation systems located within the drywell. Pipe restraints to prevent pipe whip are applied where deemed necessary to insure that:

A. Containment integrity is maintained;

B. At least one core spray system, including instrumentation, remains operable; and

C. At least one set of reactor pressure vessel level instrumentation remains operable.

It is felt that these criteria are met by the application of pipe restraints to the recirculation loop, physical separation of redundant ECCS piping and instrumentation, and physical separation of level instrumentation.

Under jet impingement loading, similar criteria are satisfied by containment and penetration design and the physical separation of ECCS components.

The primary containment vessel is completely enclosed in a reinforced concrete structure having a thickness of 4 - 6 feet. This concrete structure, in addition to serving as the basic biological shielding for the reactor system, also provides a major mechanical barrier for the protection of the containment vessel and the reactor system against potential missiles generated external to the primary containment. The space between the containment vessel and the concrete is controlled so that areas which are backed up by concrete can withstand jet forces which may occur upon failure of any system piping. Where concrete is not available, such as at the vent openings, barriers exist for jet protection.

All large pipes which penetrate the containment are designed so that they have anchors or limit stops located outside the containment to limit the movement of the pipe. These stops are designed to withstand the jet forces associated with the clean break of the pipe and thus maintain the integrity of the containment. Jet forces which may act upon the containment are assumed to be equal to reactor pressure acting directly on the containment over an area equal to the cross-sectional area of the largest local pipe or nozzle. The recirculation lines within the primary containment are provided with a system of pipe supports designed to limit excessive motion associated with a pipe split or circumferential break.

Piping connected to the reactor vessel, except the recirculation piping (see Section 5.4.1.2), has sufficient inherent protection so as to preclude the need for restraints.

Process piping primary containment penetration restraints fabricated for Dresden incorporated the design criteria established for the Oyster Creek Power Plant[7]. For use on the Dresden units the Oyster Creek design criteria were improved by changing the following aspects:

A. For Oyster Creek Case I, the load P_2 is less than the full break force, P. For Dresden the full break force P was utilized resulting in application of the greatest load possible to the drywell.

B. Values for PR and MS for Oyster Creek were reduced when penetrations were not radial to the sphere in either plan or elevation. For Dresden, values for PR and MS were reduced only when penetrations were not radial in plan. Therefore, these values were not always reduced as much as at Oyster Creek, resulting in conservatism in Dresden calculations.

C. For calculating stresses at Point 4, the Oyster Creek calculations treated the conditions as a solid attachment. For Dresden, the conditions were treated as a hollow attachment, which resulted in more realistic stress values.

D. For Case VI, in the Dresden criteria the torsional loads were calculated on the basis of combined bending and torsion in the process piping; whereas, at Oyster Creek it was assumed that the full strength of the pipe could be transferred in torsion.

The examination of other piping systems within the drywell has led to the conclusion that the main steam lines and the feedwater lines contain sufficient energy, should one of these lines suffer an instantaneous complete severance in certain specific locations, to penetrate the containment shell. Therefore, studies were made and tests conducted to determine the failure mode of this piping, i.e., to determine if the piping can sever completely and in a short enough time period to develop the energy that is required to penetrate the containment shell.

Tests conducted as part of the NRC-sponsored Reactor Primary Coolant Rupture Study demonstrate that a relationship exists between the size of a crack and the probability that the crack would propagate rapidly. The applicability of this study to the evaluation of the problem of pipe rupture in the drywell has been discussed in detail in Oyster Creek, Docket 50-219, Amendment 34. The results of the tests indicate that for a pipe which is cracked and for which the leak rate is approximately 5 gal/min, there is a probability of 1 in 1 million times that the crack would propagate rapidly enough to result in complete severance of the pipe. If the crack were of the size that resulted in a leakage of 15 gal/min, there would be a probability of rapid propagation of 10^{-4}. This crack size and leakage rate is well within the leak detection capability provided for the drywell. The leak detection capability in the drywell is discussed in Section 5.2.5.

The conditions for critical (or unstable) crack growth are based on the assumption that the cracks grow to critical size by mechanically or thermally induced cyclic loading, stress corrosion cracking, or some other mechanism characterized by gradual crack growth. From the tests conducted and the rate of crack growth, it can be concluded that the main steam lines and the feedwater lines will not suffer complete severance from gradual crack growth due to cyclic loading during the lifetime of the plant.

If an undetected fault could lead to rapid propagation or complete severance of any pipe in the drywell, it would occur in a small diameter pipe. There are no failures which can occur in small diameter pipes which will lead to a penetration of the containment.

Calculations show that pipe breaks which result in the maximum energy available for pipe whip are circumferential in nature and occur at the end of a long run of pipe in the vicinity of an elbow. The worst break is one in which the jet force from the elbow acts in a lateral direction such that the long run of pipe acts as a cantilever. Long pipe runs such as this encounter obstructions en route to the containment wall which minimize the probability of containment penetration. In addition, the smooth shape of a curved elbow and the relatively large area of impact imply that large deformation of the containment shell would be required before complete penetration would occur. Experiments conducted by CB&I have shown that for an impact area of 1.08 square feet, the containment shell would deform to the point of contacting the concrete (radial deformation of approximately 2□ inches) without causing the shell material to crack. Calculations made to determine if the other piping could possibly achieve the amount of energy necessary to penetrate the containment, using the restraint of other structural members (i.e., pipe, girders, beams, concrete, etc.) indicated that containment penetration was not possible.

3.6.2.3.2 Reactor Pressure Vessel

Jet reaction forces on the RPV were analyzed, with the RPV and support structures designed to withstand forces greater than those that would be created by full flow through any vessel nozzle at reactor design pressure. The 2 largest jet reaction forces would come from shearing a recirculation nozzle (658 kips) and shearing an outlet steam line (330 kips). Thus even if a line shears, the RPV would not be moved by jet reaction forces sufficiently to cause rupture of other connected pipes.

3.6.2.3.3 Reactor Pressure Vessel Pedestal and Reactor Shield Wall

The methodology used in arriving at initial evaluation criteria for postulated break interactions with the RPV pedestal and RPV shield wall was based on finite element analyses. Various magnitudes of pipe whip and jet impingement loads were applied to the finite element models to establish the allowable envelopes. The analyses produced allowable load values as a function of whipping or impinging pipe size for each structure. Only interactions producing loads which exceeded the initial load envelopes required more refined analysis, such as nonlinear analysis, for qualification.

3.6.2.3.4 Pipe-to-Pipe Interactions

Each target pipe was initially analyzed as a beam with variable end conditions to determine the deflection, plastic hinge formation, and tensile stress resulting from a load at the postulated target point. It was assumed that plastic hinges form successively at all points of maximum resisting moments under appropriate support conditions. Once all plastic hinges are formed and moments are eliminated, it was assumed that additional strain would occur until the total strain reached the allowable strain limit. The load applied on the given beam configuration that created this limit strain combination was calculated and designated as the limiting load. The limit load was then compared to the previously calculated applied load resulting from pipe whip or jet impingement. More sophisticated analysis was required only if an interaction produced a load higher than the limit load.

3.6.3 References

1. Letter from A. Giambusso (AEC) to Byron Lee (CECo), dated December 14, 1972, Concerning Effects of a Piping System Break Outside Containment, including errata issued on January 16, 1973.

2. Letter from D. L. Ziemann (NRC) to R. L. Bolger (CECo), dated May 12, 1976, Enclosing Amendment No. 16 and 14 to Facility License No. DPR-19 and DPR-25 for the Dresden Nuclear Power Station Units 2 and 3, respectively.

3. Integrated Plant Safety Assessment Report (IPSAR), Systematic Evaluation Program, Dresden Nuclear Power Station, Unit 2, NUREG-0823, February 1983.

4. Nuclear Regulatory Commission, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping," NUREG-0803, August 1981.

5. Letter from D.M. Crutchfield (NRC) to D.L. Farrar (CECo), October 27, 1983, IPSAR Section 4.7, Effects of Pipe Break on Structures, Systems and Components Inside Containment for Dresden Nuclear Power Station Unit 2.

6. Nuclear Regulatory Commission, Standard Review Plan, Section 3.6, NUREG-0800, July 1981, Branch Technical Position MEB 3-1, "Postulated Rupture Locations in Fluid Systems Inside and Outside Containment," Revision 1.

7. Oyster Creek Power Plant, Final Safety Analysis Report, Docket No. 50-219, Amendments 50 and 51.

8. Letter from J.S. Abel (CECo) to K.R. Goller (NRC), Dated February 18, 1975, Dresden Station Units 2 and 3, Quad – Cities Station Units 1 and 2, High Energy Line Break Analysis.

9. Letter from T.J. Rausch (CECo) to P. O'Connor (NRC), dated November 17, 1982, Dresden 2 SEP Topic III – 5.A, High Energy Break Inside Containment.

3.7 SEISMIC DESIGN

This section summarizes the seismic design bases for Dresden Station. Section 3.7.1 describes the seismic input motion applied in the analysis of structures, systems, and components at the Dresden site. Analytical methods are provided in Section 3.7.2 along with a description of the analyses performed for major plant structures. Section 3.7.3 summarizes seismic analysis methods and criteria applicable to piping systems, including alternatives implemented since the implementation of the original methods and criteria. Instrumentation used to measure and record seismic events at the Dresden site is addressed in Section 3.7.4.

It should be noted that seismic design of Dresden Unit 2 was examined by the NRC under systematic evaluation program (SEP), Topic III-6. The NRC evaluated the capability of Dresden Unit 2 to withstand a safe shutdown earthquake (SSE) by sampling review and confirmatory analysis. The NRC concluded that the majority of "safety related structures and structural elements of the Dresden 2 facility are adequately designed to resist the postulated seismic event." The remaining SEP Topic III-6 reviews have been closed out by the NRC (reference 36).

3.7.1 Seismic Input

The seismic design of structures and equipment at Dresden Units 2 and 3 was based upon the recommendations of seismologist Perry Byerly. John A. Blume and Associates, engineering consultants, reviewed the seismology, geology, and other pertinent data of the site and recommended the seismic design criteria. They also performed a dynamic analysis of the Class I structures.

Based upon the seismology report (in Volume III, Section 4 of the Dresden Unit 2 PDAR), an earthquake having an intensity of VII on the Modified Mercalli Scale is the maximum anticipated for the site.

The input for the seismic analysis of Class I equipment and structures was the north-south component of the El Centro earthquake of May 18, 1940, normalized to a maximum operating basis earthquake (OBE) ground acceleration of 0.10 g. The length of the earthquake record employed was 10 seconds. The maximum response occurred within the first 4 to 6 seconds. Longer intervals of motion have been used on similar structures, and the maximum response always occurred within 4 to 6 seconds regardless of the length of record used.

Figure 3.7-1 shows an unsmoothed response spectrum curve for the El Centro earthquake normalized to 0.10 g maximum OBE ground acceleration at 2% damping. The unsmoothed response spectrum was calculated from the El Centro record using the usual analytical methods. Also shown is the design OBE response spectrum curve for Dresden (Housner) at 2% damping which was applied to equipment and structures analyzed by the response spectrum method rather than the time-history method.

The actual El Centro response spectrum has not been used for any of the analyses. All OBE response spectrum calculations employed the smooth Housner response acceleration spectrum shown in Figure 3.7-2. When an OBE time-history analysis was made, the El Centro earthquake record was employed, normalized to a ground acceleration of 0.10 g.

The time-history method of analysis was used for the reactor-turbine building, the reactor pressure vessel, the chimney, and the drywell.

The response spectrum method of analysis was used for the following structures or systems:

A. Recirculation loop piping,

B. Suppression chamber ring header (suction),

C. Feedwater lines,

D. Main steam lines,

E. Isolation condenser,

F. Control room, and

G. Suppression chamber.

The actual El Centro unsmoothed spectrum (Figure 3.7-1, upper curve) was not used for response analyses in either of the methods (time-history or smooth response spectrum) above. It was used, however, to verify that when using the time-history method the maximum OBE loadings did not occur in valleys of the unsmoothed spectrum.

Since the unsmoothed curve is generated from the time-history record and the smooth response spectrum curve has lower accelerations for nearly all periods, it is concluded that the time-history method tends to overestimate the response when compared to the design criteria (smooth response spectrum).

The seismic consultant prepared the OBE acceleration response spectrum curves shown in Figures 3.7-2 and 3.7-3 based upon a ground acceleration of 0.10 g and the Housner response spectra shape. The seismic design of Class I structures and equipment was based upon a dynamic analysis using these curves. The natural periods of vibration were calculated for buildings which are vital to the proper shutdown of the plant. The damping factors given in Table 3.7-1 were used for strong vibrations within the elastic limit.

For the design of Class I structures and equipment the maximum horizontal acceleration and the maximum vertical acceleration were considered to act simultaneously. Where applicable the resulting seismic stresses for the two motions were combined linearly. The vertical OBE acceleration assumed was equal to 0.067 g, two-thirds the horizontal ground acceleration.

To assure that the plant can be shut down with containment and heat removal facilities intact, Class I structures have been designed to accommodate a ground motion of 0.2 g. Care was taken to assure that structures will not fail in a brittle manner. For this higher intensity earthquake, on a case by case basis where technically justified, higher percentages of critical damping factors from those shown in Table 3.7-1 have been used.

The results of the seismic analysis were used in the design of the associated Class I structures, systems, and components. For the seismic analysis of equipment, absolute acceleration was used at the points of support. Where a dynamic analysis was not performed, the horizontal seismic coefficients for rigid Class I equipment in the reactor turbine building were equal to or greater than the building acceleration at the installed elevation. The OBE vertical seismic coefficient is equal to two-thirds ground acceleration or 0.067 g. Flexible and rigid Class I piping systems are analyzed as described in Section 3.7.3.1.

The conclusion that the Class I equipment in the turbine building has adequate protection is based on the combined reactor turbine building dynamic analysis described in Section 3.7.2.2. All shear walls and seismic supporting elements have been designed using loads shown in the dynamic analysis. The resulting stresses are within the allowable stresses of the Uniform Building Code without the benefit of increased allowable stresses due to seismic loads.

Therefore, the design of these areas is conservative, providing adequate protection for the Class I equipment.

Class II items were designed following the normal practice for the design of power plants in the State of Illinois, but as a minimum the design met the standards given in the Uniform Building Code for Zone 1. The usual practice of determining the stress due to earthquakes, applying a static load based on a specified seismic coefficient, was followed. Allowable stresses for building materials in Class II structures are as specified in the Uniform Building Code. A one-third increase is allowed for combinations including seismic or wind loads.

3.7.2 Seismic System Analysis

This section provides an overview of the methods used to perform seismic system analyses at Dresden. Section 3.7.2.1 describes the general seismic analysis techniques employed for major Class I structures. Modeling and analysis details are provided for the reactor building (Section 3.7.2.2), control room (Section 3.7.2.3), and the chimney (Section 3.7.2.4).

3.7.2.1 Seismic Analysis Methods

Participation factors were not involved in the analyses of Class I structures, systems and components because either a time-history analysis was used, as outlined below, or a modal analysis was used, taking the square root of the sum of the squares (SRSS) with each mode participating equally. The mode shapes were not plotted in the analysis. The method employed in combining the modal values to obtain the design values of the acceleration, seismic force, shears and/or moments is given below for the respective methods.

3.7.2.1.1 Time-History Analysis

The response of each mass for each mode considered at each increment of time is retained in the computations, and total response for each increment of time is obtained through the algebraic sum of each mass point's modal contribution at that particular instant of time. Once displacement and inertia forces are calculated, time-histories are established, and a time-history of shears, moments, displacements, and accelerations is determined. These records are then scanned to determine the maximum values.

3.7.2.1.2 Modal Analysis

The total response for each mode is determined by taking the SRSS of the maximum response for each mode. Further explanation of this method is given in Section 3.7.3.1, which describes seismic analysis of Class I piping systems.

3.7.2.2 Reactor Building

This section summarizes the OBE seismic investigation of the combined reactor turbine building in two directions. Based on the recommended earthquake design criteria established for the station, envelopes of maximum acceleration, displacement, shear and overturning moment versus height have been developed and are presented in Section 3.8.4.1 for the two assumed earthquake directions.

For seismic purposes the building is considered to be a reinforced concrete and steel structure consisting of a reactor building and a turbine building. The reactor building is a reinforced concrete structure from its foundation at elevation 472'-6" to elevation 613'-0". A steel superstructure with lateral bracing has been placed at this level. The turbine building is a reinforced concrete structure from its foundation at elevation 513'-6" to elevation 561'-6". A steel-framed superstructure with lateral bracing has been placed at this level. The turbine building is connected to the reactor building at the operating floor at elevation 561'-6" and to its steel frame roof at elevation 622'-6" to form the combined reactor turbine building.

For the dynamic response investigation, an equivalent lumped mass system was selected to approximate the building in each direction. In Figures 3.7-4 and 3.7-5, the left portion of the model shown represents the reactor building and the right portion represents the turbine building. In a north-south direction these two portions of the model were connected at two points: at elevation 561'-6", representing the operating floor of the turbine building; and at elevation 622'-6", representing the roof of the turbine building. In the east-west direction, the left and right portions of the model were connected at one point only, elevation 561'-6" representing the operating floor of the turbine building.

The equivalent system consists of masses lumped at each floor level except that the top story steel frames were approximated by an equivalent multi-mass system as shown in Figures 3.7-4 and 3.7-5. Each story level mass represents the mass of concrete and equipment at each floor and the tributary mass of the equipment and concrete walls between adjacent floors. The top story masses are similarly developed but include the tributary mass of the walls, frame, and the mechanical equipment of the story. The average area and moment of inertia of the concrete between floors was used to determine the stiffness characteristics between masses. The steel-framed top portion was investigated separately; however, an equivalent frame stiffness was developed for each direction. A value of 3×10^6 psi was assigned as the elastic modulus of concrete, and 3×10^7 psi was assigned as the elastic modulus of the steel structure.

The design modulus of 3×10^6 psi is in accordance with the ACI "Building Code Requirements for Reinforced Concrete" (ACI 318-63, Section 1102), which is standard design practice. However, it is recognized that the modulus of elasticity of concrete increases with age following the 28-day period, but it is difficult to evaluate the amount of increase. The following factors affect the strength of concrete:

A. Curing temperature,

B. Initial temperature,

C. Variations in mixes, and

D. Amount of hydration.

The elastic modulus is not directly proportional to the strength of concrete; nevertheless, the effect of increasing the strength causes an increase in the modulus. However, the increase in the modulus due to age is not believed to be significant in light of all the uncertainties affecting the modulus of concrete. Whatever the small change in the modulus may be, this effect is partially accounted for by cracks in the concrete structure due to shrinkage and temperature. Such cracks tend to make the structure more flexible, which tends to compensate for the increased modulus. Also the percentage change in the modulus is small compared to other inputs in the analysis such as dimensions, areas, cross sections, mass grouping, etc. Hence the effect of an unknown modulus change on the validity of the dynamic analysis is considered to be negligible.

The mathematical model of the reactor turbine building (Figures 3.7-4 and 3.7-5) includes the mass of the drywell. In the analysis of the drywell, discussed in Section 3.7.2.2.1.1, the mass and properties of the drywell are taken out of the reactor turbine building model. The drywell lumped mass model was considered fixed at elevation 500'-0-⅝" and laterally supported at elevation 572'-2".

The mathematical model of the reactor pressure vessel (Figure 3.7-6) gives the support conditions which are fixed at the base of the pedestal, lateral support at the stabilizer elevation connecting the reactor pressure vessel to the sacrificial shield, and a horizontal pipe truss system connecting the sacrificial shield to the building. The reactor pressure vessel seismic analysis is affected by seismic motions at the support points only. Therefore, the reactor pressure vessel could be decoupled from the rest of the building and analyzed separately, taking into consideration the effects of input motion at all supports and conservatively combining the individual effects.

A rebaselined seismic model of the RPV, internals and interrelated portions of the main plant structures was developed in 1994 to address issues associated with the seismic analysis of the core shroud repair as well as other RPV internals component repair designs. This model was modified to incorporate the core shroud repair hardware and was used to perform the seismic analysis for the repair design. The revised seismic analysis of the RPV was performed using a combined RPV and main power block model to directly account for the affects of the coupling. The El Centro and Housner spectrum compatible time histories were used to perform the seismic analysis for the core shroud repair design. The enveloping values (accelerations displacements, shears and moments) from the two time histories were used for the evaluations and design of the repair hardware. A core shroud repair designed to structurally replace circumferential core shroud welds H1 through H7 was installed in Unit 2 during D2R14, and is scheduled to be installed in Unit 3 during D3R14.

The ground motion utilized in determining the dynamic response of the reactor building has a maximum base OBE acceleration of 0.10 g. A constant vertical OBE acceleration of 0.067 g was assumed to act simultaneously with the horizontal OBE design acceleration.

The computer program used in this analysis was specially designed to solve the dynamic response of structures subjected to arbitrary ground motions. Since the program was written to cover as many structural configurations as possible, the structural member input data for the program is in the form of member moments of inertia, areas, and effective shear areas. The effects of axial and shear deformation are included in the calculation of the stiffness matrix. The input and output data are shown in Tables 3.7-2 and 3.7-3. A simplified block diagram of the computer program is shown in Figure 3.7-7.

Section 3.8.4 presents the results of the reactor building seismic analysis. In the north-south direction, the coupled periods of vibration determined for the subject structure were 0.39, 0.23, 0.20, 0.17, and 0.14 seconds for the first five modes respectively. In the east-west direction coupled periods of vibration determined for the subject structure were 0.37, 0.34, 0.25, 0.15 and 0.12 seconds for the first five modes respectively.

The structure has been designed to resist the OBE shears and moments presented without any increase in stress for short-term loadings. In addition, the structure was reviewed to assure that it would resist a safe shutdown earthquake (SSE), equal to twice the OBE seismic shears and moments presented without hindering the ability of the plant to be shut down safely.

Calculations were performed with the aid of a digital computer and five modes were considered in the analysis. Since the predominant response of the building is due to the first mode of vibration while the fifth mode has a period of vibration approaching that of a rigid system, the fifth mode has a negligible participation on the overall response.

3.7.2.2.1 Primary Containment Seismic Analysis

The seismic study of the drywell conducted by John A. Blume and Associates is summarized in this section.

3.7.2.2.1.1 Drywell Seismic Analysis

For seismic purposes, the drywell containment structure is considered to be a bulb-shaped steel structure free to move except where it is attached to the concrete around it at two points: at the bottom fixed in the reactor building at elevation 500'-0-⅝" and supported laterally by the reactor building at elevation 575'-2".

A 2-inch gap for thermal expansion between the steel drywell and the concrete containment shield wall was ensured by cementing elastic polyurethane sheets to the steel drywell and to the sleeves for the drywell penetrations prior to pouring the containment shield concrete wall. This process is described more fully in Section 6.2.1.2.3.6. Since the foam is compressible, the drywell is considered free to move except at the two attachment points.

The drywell is embedded integrally within the reactor building concrete mass substructure. The reactor pressure vessel pedestal was cast integrally with the concrete mass above the bottom of the drywell. The relative mass and stiffness of the concrete substructure assures the fixity of both the drywell and reactor pressure vessel.

In order to perform the mathematical analysis, the drywell structure was idealized as a lumped mass system supported on elastic columns fixed at the elevations indicated above.

Twenty-one lumped masses were chosen to represent the structure. Since the drywell was analyzed as coupled to the reactor turbine building, the interaction between building elements was automatically considered. In a coupled model all interaction forces are determined by one analysis. The mathematical models are shown in Figures 3.7-8 and 3.7-9 for the north-south and east-west directions, respectively, as correlated with the reactor and turbine buildings. In Figure 3.7-8 the spring numbered 15 represents three 14WF167 beams connecting the turbine building superstructure to the concrete portion of the reactor building; each beam has a stiffness value of 105,000 kips/ft. The connecting link number 16 represents the rigid connection of the reinforced concrete operating floor of the turbine building to the reactor building and is not considered to act as a spring. In Figure 3.7-10 the drywell model is shown by itself only to allow for a larger scale for the drawing. The model is physically tied to the other buildings for the analysis as indicated. Properties of the elastic column were determined by cutting a horizontal section through the drywell between each mass point and calculating the moment of inertia and effective shear area. This calculation was performed by a computer using as an input the average diameter and shell thickness between the lump mass points in question. For the dynamic response investigation, equivalent mass systems were selected to approximate the coupled combined reactor turbine building and drywell in each direction. The natural frequencies and mode shapes and the dynamic response of an equivalent lump mass system were determined and six modes were considered, with the damping values assigned at 5% for all modes. The horizontal ground motion utilized in determining the dynamic response of the reactor building had a maximum base acceleration of .10 g (OBE) in accordance with established criteria.

The computer program used in this analysis was specially designed to solve the dynamic response of structures subjected to arbitrary ground motions. Since the program was written to cover as many structural configurations as possible, the structural member input data for the program is in the form of member moments of inertia, area, and effective shear area. Effects of axial deformations and shear deformations are included in the calculation of the stiffness matrix. The computer retains the response of each mass and each individual mode at each increment of time, and the total response for each increment of time is obtained by adding together the responses of each mass point for each mode at a particular instant of time. This results in an exact combination of mode participation without the necessity of using approximate methods such as the root mean square method.

In the north-south direction, the periods of vibration determined for the coupled system were 0.39, 0.23, 0.20, 0.17, 0.14, and 0.077 seconds for the empty condition, and 0.39, 0.23, 0.20, 0.17, 0.14, and 0.12 seconds for the flooded condition, for the first six modes, respectively. Evidently, the flooded drywell has an effect on the sixth mode of the coupled system only since the masses of the flooded drywell are small compared to the masses of the combined reactor turbine building.

In the east-west direction, the periods of vibration determined for the subject structure were 0.37, 0.34, 0.25, 0.15, 0.12, and 0.065 seconds for the empty condition, and 0.37, 0.34, 0.25, 0.15, 0.12, and 0.116 seconds for the flooded condition, for the first six modes, respectively. The natural period of vibration of the drywell is about 0.07 seconds for the empty condition and 0.12 seconds for the flooded condition.

Depending on the period of vibration of the system in question, the response of the system to the earthquake is sensitive to the degree of damping used. For rigid structures such as the drywell, the response is relatively insensitive to the damping values used. The interaction forces between the drywell and reactor building would be sensitive to the damping assumed for the reactor building. Since this damping was conservatively assumed to be 5%, the calculated interaction forces presented are conservative.

Viscous damping was used in all the analyses. Damping coefficients for Class I items are given in Section 3.7.1.

The torsional rigidity of the drywell/reactor turbine building is so great that torsional vibrations are negligible. Also the system can easily accommodate possible torsional shears due to lack of symmetry.

Based upon experience with the analysis of other drywell structures, the resulting effect of hydrodynamics is to reduce the total seismic forces. Thus, the effects of the dynamic response of the fluid in the drywell are conservatively neglected in this dynamic analysis.

The drywell has been designed on the basis of the results presented. Each direction of the earthquake was considered separately and the most severe case was used in the final design. In addition, the structure was designed to resist a constant vertical acceleration equal to 0.067g (OBE) acting simultaneously with the horizontal accelerations presented. The final design was reviewed for twice the OBE design parameters presented, including a constant vertical SSE acceleration of 0.133g acting simultaneously with twice the OBE horizontal accelerations presented.

3.7.2.2.1.2 Torus Seismic Analysis

The torus seismic analysis, including ring header, was updated as part of the Mark I Containment Long-Term Program. The results are documented in the Plant Unique Analysis Report[1] (PUAR) which has been reviewed and accepted by the NRC.[2] The NRC safety evaluation concluded, "CECo's PUAR analysis verified that the containment modifications made have restored the original design safety margin to the Mark I containments at Dresden Units 2 and 3." See Sections 3.8 and 6.2 for additional details about the Mark I Program.

3.7.2.3 Control Room

The control room is located in the turbine building with a separate air conditioning system complete with heating and cooling equipment and fresh air intake which can be isolated.

As with the reactor building seismic analysis, an equivalent mass system was used as a model for the dynamic analysis of the control room structure. Using a preliminary response acceleration curve and the results of a dynamic analysis, the absolute maximum acceleration was calculated for each level, and from this the seismic forces "applied" to each floor were calculated and the resulting shear and moment diagrams plotted. This analysis considered flexibility about the minor principal axis only. Section 3.8.4.2 summarizes results of the control room seismic analysis.

3.7.2.4 Chimney

For seismic study purposes the concrete wall thickness of the tapered chimney was considered to be 7 inches for the top 100 feet and 18 inches for the bottom 30 feet, with an internal diameter of 11.0 feet at the top and 22 feet, 7½ inches at the base. This configuration is indicated in Figure 3.7-11. However, in addition to its primary purpose, the chimney base is also used as a means of providing the gland seal hold up volume. This then modified the study dimensions of the chimney such that the internal diameter at the base became 20 feet, 1½ inches and the concrete wall thickness became 33 inches for the bottom 43 feet, 6 inches. This configuration is shown in Figure 3.3-2.

The addition of the concrete at the base served to stiffen the structure as it was previously analyzed for seismic conditions.

In developing a mathematical model for dynamic analysis the chimney was treated as a flexible cantilever system with the base fixed at the top of the foundation at grade level, elevation 517'-0". The foundation is fixed at the bottom, founded on and surrounded by rock at elevation 500'-0". This condition results in a shorter period for the stack than if the base were allowed to rotate; thus, the corresponding response acceleration will be higher. Therefore, base rotation was conservatively neglected.

An OBE seismic time-history analysis was made on the chimney using the time-history record of the El Centro earthquake of May 18, 1940, north-south component normalized to 0.10 g ground motion. The damping value used was 5%.

For the mathematical model, 33 mass points were considered to be supported by weightless elastic matrices for the cantilever system, the periods and mode shapes were calculated, displacement and inertia force time-histories were established, and a time-history of shears, moments, displacements, and accelerations was determined.

The computer program used in this analysis was specially designed to solve the dynamic response of structures subject to arbitrary ground motions. Member input data for the program was in the form of moments of inertia, areas and effective shear areas. The effects of axial and shear deformation were included in the formation of a stiffness matrix. The process logic of the computer solution is summarized in Figure 3.7-13. Calculated data used as input to the computer is shown in Figure 3.7-12. The first seven natural periods of vibration are 1.381, 0.372, 0.158, 0.089, 0.058, 0.042 and 0.033 seconds. Results of the seismic analysis are summarized in Section 3.8.4.3.

3.7.3 Seismic Subsystem Analysis

This subsection addresses seismic subsystem analysis. Section 3.7.3.1 summarizes seismic qualification of piping systems, and Section 3.7.3.2 provides details of refinements to seismic criteria and analysis methods implemented as alternatives to those described in the original Dresden FSAR.

3.7.3.1 Piping

Generally three methods have been used for seismic analysis of piping systems at Dresden:

A. Mode superposition using floor response spectra;

B. Static analysis using conservative static coefficients; or

C. Lateral deflection and force evaluation curves.

Each of these three analysis methods is described in the following subsections. Additional information regarding which method was originally applied to various systems may be found in Section 3.9.3.1 and Table 3.9-6.

3.7.3.1.1 Dynamic Analysis of Class I Piping

The subsequent paragraphs provide a discussion of the design bases of mode superposition using floor response spectra, as applied in the original analysis of the main steam lines, feedwater lines, and recirculation piping.

3.7.3.1.1.1 Mathematical Model

Each piping run was idealized as a mathematical model consisting of lumped masses separated by elastic members. Lumped masses were located at selected critical points as required to adequately represent the dynamic and elastic characteristics of the pipe system. Using the elastic properties of the pipe between successive mass points, the flexibility matrix for each pipe run was determined. The flexibility calculations included the effects of torsional, bending, shear, and axial deformations. Also included was the change in flexibility due to the curved members.

After the flexibility and mass matrices of the mathematical model were obtained, the frequencies and mode shapes for several modes of vibration were determined, usually five or six. The effect of higher modes was found to be negligible. The mode shapes and frequencies were determined by solving the following equation:

$$(K - \omega_n^2 M)\phi_n = 0$$

where:

K = square stiffness matrix of the pipe loop

M = mass matrix for the pipe loop

ω_n = frequency for the n^{th} mode

ϕ_n = mode shape matrix of the n^{th} mode

After the frequency was determined for each mode, the corresponding spectral acceleration was read from the floor response spectra for the reactor building. These floor response spectra were given for two directions of earthquake motion for each floor level in the building. The response spectrum used for the original response spectrum analyses was for Mass 6 north-south direction with 0.005 damping ratio. This spectrum was chosen because it resulted in the maximum response for the pipe systems. Using these spectral accelerations, the response for each mode was found by solving the following equation:

$$Y_n \max = R_n Sa_n D / M_n \omega_n^2$$

where:

$Y_n \max$ = response of the n^{th} mode

R_n = participation factor for the n^{th} mode = $\Sigma[\ M_i \phi_{in}\ ]$

Sa_n = spectral acceleration for the n^{th} mode

D = earthquake direction matrix

M_n = generalized mass matrix for the n^{th} mode = $\Sigma[\ M_i \phi_{in}^2\ ]$

ω_n = frequency for the n^{th} mode

Using these results the maximum displacements for each mode were determined for each mass by the following relationship:

V_{in} = $\phi_{in} Y_n \max$

where:

V_{in} = maximum displacement of mass i for mode n

The total response for each mass was determined by taking the SRSS of the maximum deflection for each mode:

$$V_i = \sqrt{\Sigma V_{in}^2}$$

where:

V_i = maximum displacement of mass i due to all modes calculated

The inertia forces for each direction of earthquake were then determined from:

Q = K V

where:

Q = inertia force matrix for all masses

V = maximum displacement matrix

The internal forces, moments, and stresses for the pipe loops can be determined from standard structural analysis methods using the inertia forces for each direction of earthquake as a loading condition.

3.7.3.1.1.2 Computation of Stresses

All values for forces and moments are for the global coordinate system. After converting these values to the member coordinate system, the pipe stresses were determined in accordance with the following equation:

$$S_e = \sqrt{S_b^2 + 4S_t^2}$$

where:

S_e = equivalent resultant stress for pipe

S_b = resultant bending stress at joint = B M_b/Z

S_t = resultant torsional stress at joint = M_t/2Z

M_b = resultant bending moment

M_t = resultant torsional moment

Z = section modulus of pipe

B = stress intensification factor (see below)

3.7.3.1.1.3 Effect of Curved Members

The flexibility matrix of the pipe system included the effects of curved members and elbows. This curvature effect depends on the bend characteristic shown below:

$$h = t\,R/r_m^2$$

where:

h = bend characteristic

t = thickness of pipe

r_m = mean radius of pipe

R = radius of bend

The flexibility factor gives the change in flexibility due to the curved members and is given by:

$K = 1.65/h$

where:

K = flexibility factor

h = bend characteristic

The stresses in curved pipes differed from those calculated for straight pipes with equal bending moments. This stress increase is given by the stress intensification factor:

$i = 0.90/h^{2/3} \geq 1.00$

where:

i = stress intensification factor

h = bend characteristic

3.7.3.1.1.4 Description of Computer Program

All of the calculations outlined above were performed with the aid of a digital computer. The computer program employed was written specifically for the analysis of three-dimensional piping systems.

The input data for this program consisted of the coordinates of all critical joints and valves in the pipe system including the coordinates of the joints selected as lumped masses. Additional input consisted of the thickness, diameter, weight, and elastic modulus of the pipe runs. The computer then calculated the stiffness and force transformation matrices, mode shapes, frequencies, and inertia forces. Using the inertia forces as loading conditions, the internal forces and moments, displacements and stresses were then computed and printed out.

For the OBE dynamic response of the pipe runs, an analysis was made for both the X and Z direction earthquake. For each of these loading conditions, a constant vertical Y direction acceleration of 0.067 g (two-thirds horizontal ground acceleration) was combined with the horizontal.

The results are in the form of coordinates, internal forces, moments, displacements, reactions at intermediate supports, and stresses.
The results were divided in two cases for each pipe. Case 1 consisted of the results due to the appropriate spectral accelerations for the X-direction earthquake combined with two-thirds of the ground spectral accelerations for the Y-direction (vertical) earthquake. Case 2 consisted of the results due to the appropriate spectral accelerations for the Z-direction earthquake combined with two-thirds of the ground spectral acceleration for the Y-direction earthquake.

3.7.3.1.2 Class I Piping Analyzed Statically Using Conservative Static Coefficients

A detailed, sophisticated dynamic response spectra analysis of a complex piping system where frequencies, mode shapes, and inertia forces are calculated and then from which shears, moments, and displacements are determined is extremely complicated.

In lieu of a complete dynamic analysis, a less complicated static analysis was made in some cases, utilizing conservative static seismic coefficients. These static coefficients were determined in the following manner.

Horizontal static coefficients were determined by using the average of the peak values from the unsmoothed ground spectral curve of the normalized earthquake selected for the site.

This average acceleration was then multiplied by the ratio of the building response acceleration at the installed elevation of the piping to maximum ground acceleration:

$$C_p = A_{peak} (A_{building}/A_{ground})$$

where:

C_p = static coefficient for the piping system

A_{peak} = average acceleration of the peak the ground spectral curve

$A_{building}$ = acceleration of the building at the installed elevation of the piping system

A_{ground} = acceleration of the ground

The vertical coefficient was taken at a constant value equal to two-thirds of the maximum base ground acceleration, (e.g., .067 g for OBE).

These seismic loads were applied uniformly in both the horizontal direction (perpendicular to each other) and the vertical direction. The results from one horizontal direction were combined with the vertical direction and then compared with the results from the other horizontal direction combined with the vertical direction. The larger of the two results was then used for design.

This procedure resulted in horizontal seismic coefficients that range in value as shown on Table 3.7-4.

The coefficient used on an entire piping system was the coefficient corresponding to the elevation zone where the majority of the system mass is located. It is emphasized that these coefficients are highly conservative.

The design approach used for hangers, restraints, snubbers, etc., for statically analyzed piping systems was the same as that used for dynamically analyzed systems; the design approach is described more fully in the following subsection and Section 3.9.3.

3.7.3.1.2.1 Static Analysis Application

3.7.3.1.2.1.1 Location of Trial Restraints

The first consideration in analyzing a piping system for seismic loading was to establish a trial span between restraints (on a stress basis):

The piping was assumed to be a uniformly loaded fixed-fixed beam of length corresponding to the allowable distance between restraints. The fixed-fixed beam model is justified because the slope and deflection can be assumed to be zero at the restraint attachment points. The maximum moment in the pipe span on this basis is equal to:

$$M = w\,l^2/12$$

where:

l = length of span, feet

w = weight, per foot

M = moment, foot-pounds

This maximum moment occurs at the ends of the span. If a valve or other concentrated mass is located in the span (see Figure 3.7-14), there is an additional moment at the span ends of:

$$M_1 = (Pab^2/l^2) \quad \text{and} \quad M_2 = (Pa^2b/l^2)$$

The moment to be allocated to the seismic contribution was then determined from the USAS B31.1[20] requirements (detailed discussion of B31.1 requirements presented later) as follows:

From B31.1 paragraph 102.3.2(d),

$S_h \geq S_{weight} + S_{pressure} + S_{seismic}$

$S_h = 1{,}000 + PD/4t + M/Z$ or

$M = [S_h - 1{,}000 - (PD/4t)]Z$

where:

$M = M + M_1$ or $M + M_2$, inch-pounds

S_h = allowable stress B31.1, psi

1,000 = Assumed weight stress, psi

$\frac{PD}{4t}$ = Axial pressure stress, psi

Z = Pipe section modulus, cubic inches

The trial span, l, was found from the expressions for M, M_1, and M_2.

Once the trial span was established, the piping layout was reviewed. Engineering judgement played a major role in applying the trial span equations developed above. For example, the pipe leg a-b shown in Figure 3.7-14 is considered as a concentrated mass.

The piping was reviewed and an evaluation made that the trial span equations were satisfied for both horizontal and vertical loading. Horizontal restraints are usually added to comply with the trial span requirement. Since the vertical seismic load is small compared to the weight, usually no additional vertical supports were needed over those required for weight. However, no seismic restraint credit was taken for variable and constant support hangers or for sliding supports.

3.7.3.1.2.1.2 Seismic Load Analysis

Once the trial restraints were located, the piping layout and building structural plans were reviewed to determine if the trial restraint locations were physically possible. Usually some minor adjustments were necessary. A uniform load analysis was then performed as follows:

An internal load distribution was determined for each of two horizontal 1 g uniform loadings and for a 1 g vertical uniform loading. The direction for one horizontal loading was taken normal to the axis of the most slender piping profile and the other direction was taken along that axis
.

In this load analysis, no restraint credit was taken from variable and constant support hangers or for sliding supports.

The boundaries of the system model used in the analysis extend well beyond the stress analysis boundaries set by the first normally closed valve. This provides confidence that the loading influence of piping outside (but attached to) the critical portion of the system was adequately accounted for.

3.7.3.1.2.1.3 Code Stress Analysis

Once the internal moment distribution of the system was determined, the stress analysis was then performed by calculating the longitudinal stress, σ, at a section per Paragraph 102.3.2(d) of B31.1-1967:

Stress, $\sigma = P\,(A_f/A) + M_w/Z + M_e/Z$ (1)

where:

P = internal design pressure, psi

A_f = flow area of pipe, square inches

A = metal area of pipe, square inches

Z = section modulus, cubic inches

M_w = Resultant bending moment due to weight, inch-pounds

M_e = Resultant earthquake moment as defined below, inch-pounds

The earthquake moment, M_e, is determined by the larger of the following equations:

M_e = $(q\,M_1 + K\,M_2)$ or $(q\,M_3 + K\,M_2)$

where:

M_1 = resultant bending moment due to 1 g seismic load in X direction (see Figure 3.7-15) (Sliding supports not included.)

M_2 = resultant bending moment due to 1 g seismic load in Y direction (Sliding supports not included.)

M_3 = resultant bending moment due to 1 g seismic load in Z direction (Sliding supports not included.)

K = two-thirds of maximum base ground OBE acceleration ($K = .067$ g)

q = seismic static coefficient, as defined in Table 3.7-4

The value of stress, σ, must not exceed $1.2S_h$ from Table A-1 of B31.1.

The above procedure is conservative because it is assumed that M_w, M_1, M_2, and M_3 are collinear. Another conservatism that is introduced is to use the combined stress, S_e, from the static load analysis computer printout. This stress value contains the torsional shear stress in addition to the bending stress. On this basis Equation 1 becomes

$$\sigma = \left(\frac{Af}{A}\right)P + Se_W + S_e \qquad (2)$$

Se_w = combined deadweight stress

Se_e = combined earthquake stress

Se_e = $qSe_1 + KSe_2$ or $qSe_3 + KSe_2$

All other terms are as defined previously.

The systems were checked by Equation 2 because of the convenience of application. If a point did not pass this check, refinements were made. If refinements did not correct the overstress, restraints were added or other corrective action taken.

The B31.1 bending stress intensification factors for elbows and branches were applied full value to both the weight and seismic analysis stresses. This is a conservative practice. The B31.1 Code stress intensification factors are based in part on fatigue tests of actual piping components. Inherently then, the stress intensification factors include the effect of peak stress. Since weight and seismic stresses are primary stresses, the factors are conservative when used in primary stress calculations. Because the stress factors associated with unreinforced branch connections are so large, it was sometimes necessary to reinforce a pipe-to-pipe branch connection to reduce stress to an acceptable level. This was done when the piping had adequate restraint to maintain seismic stress at a relatively low level throughout except at the branch point.

The piping layout drawings were also reviewed to determine if the seismic deflections would cause the piping to strike any structure, component, etc., which may damage the piping.

3.7.3.1.2.1.4 Design Loads on Attached Equipment and Restraints

After satisfactory pipe stresses were obtained, the piping drawings were modified to include the restraints and the details of the restraint were developed.

Seismic design loads for rigid deadweight supports and seismic restraints were based on combining the worst restraint load produced by a horizontal uniform seismic load with the restraint load produced by vertical seismic and weight.

In the case of variable or constant supports, the hanger manufacturer was asked to certify that seismic deflections at hangers can be absorbed without failing the hanger.

At equipment connections (pumps, piping designed by others, etc.) or pipe anchors, six components of load (F_x, F_y, F_z, M_x, M_y, M_z) derived in the same manner as the restraint loads, were presented to the responsible designer for review. Where pipe anchors were designed for pipe ultimate or limit loads, no additional anchor analysis was required.

After the seismic restraints were designed, the stress analyst reviewed the details to insure that the restraints behaved as assumed in the uniform load analysis.

3.7.3.1.2.1.5 Summary of Static Analysis

Piping drawings were reviewed and trial span equations coupled with engineering evaluation were used to establish the need for seismic restraints. After the trial restraints were located, they were checked for compatibility with the building structure.

The adequacy of the trial restraints was checked by making a static uniform seismic load analysis and combining pressure, weight and seismic stress in accordance with B31.1 criteria. Static acceleration factors were based on the ground spectral curve with building response verification included.

All restraints, hangers, and anchors were designed for the seismic loads. Connecting equipment seismic loads were reviewed to insure that the loads could be safely sustained.

3.7.3.1.3 Class I Piping Analyzed Statically Using Lateral Deflection and Force Evaluation Curves

Lateral deflection and force evaluation curves for piping systems were developed by John A. Blume and Associates for Class I piping, based on the natural period of the piping system as a function of pipe size (diameter and schedule) and span. The curves provided guidelines for the evaluation of the lateral supports in a piping system. The pipes were considered filled with water and the wall thickness or schedule number was shown on the graph. The modulus of elasticity is 29 x 10^6 psi. The curves were based on a single span with pinned-pinned end conditions.

The use of the piping curves was as follows:

A. Given the period of the supporting building or structure, the period of the piping, as to when it is rigid, flexible, or resonant, was established.

$$\text{Rigid:} \quad \frac{\text{Period of Structure } (T_b)}{\text{Period of Piping } (Tp)} > 2.0$$

$$\text{Flexible:} \quad \frac{\text{Period of Structure } (T_b)}{\text{Period of Piping } (Tp)} < 0.7$$

$$\text{Resonant: } 0.7 \leq \frac{\text{Period of Structure } (T_b)}{\text{Period of Piping (Tp)}} \leq 2.0$$

B. The maximum spans for various diameters of piping to carry a lateral load of 0.5 g with stresses no more than 1500 psi were established. Table 121.1.4 in Power Piping USAS B31.1.0, 1957 shows maximum spans for various pipe sizes that will support a 1.0 g load, including contents with stresses no more than 1500 psi. When these spans are increased by a factor of 1.41 (the square root of two), the supported load is reduced to 0.5 g (see Table 3.7-5).

Since most of the small lines are supported in a manner to be classified as rigid, they will see the same acceleration as the floor of the reactor building, which has an acceleration of less than 0.4 g below elevation 589'-0". Thus, if the piping line is classified as being rigid, the seismic stress is less than 1500 psi.

C. The resonant limits were established for various diameters of piping by using Blume's curves giving natural periods as a function of pipe size and span.

D. After the span was selected, the maximum deflection and the reaction on the supports was determined.

The span between lateral supports was checked against a deflection limit of L/480. This deflection limit restriction may govern in small pipe sizes. Deflection under weight or seismic loads is generally of secondary importance. In all Class I piping systems, the deflection of the line was kept within reasonable bounds in order to minimize pocketing and to avoid contact with other lines or objects. Limiting the maximum deflection to L/480 is a practical limit for average piping in process units.

When spans are limited to support a lateral load of 0.5 g or are limited so that the line is classified as rigid, the maximum deflection of L/480 will not exceed 1 inch. Also, the L/480 is associated with the use of lateral deflection and force evaluation curves; these curves, in general, are limited to piping less than 10 inches in diameter. For piping lines this size, the spans used result in deflections considerably less than 1 inch.

E. The displacement and support reactions for equipment located at higher elevations and having flexible spans were increased by a factor of 3 due to amplification of equipment accelerations over the ground acceleration.

F. Spans were reduced by a factor of 2 to account for valves or branch lines. For 90° bends, either leg is not more than L/2 where L is three-quarters of the allowable span.

Supports in general were located in a manner to be out of the resonant range. If in the resonant range, the static coefficients given in Table 3.7-4 were used. If in the flexible range, the ground response spectra for the site was used.

Additional information concerning seismic analysis of piping may be found in Section 3.9.3.

3.7.3.2 Equipment

Seismic qualification of mechanical equipment is addressed in Section 3.9.2.2 and seismic qualification of electrical equipment is addressed in Section 3.10.

3.7.3.3 Other Structural Elements

3.7.3.3.1 Masonry Walls

Masonry walls are utilized at Dresden as firewalls, partition walls, radiation shielding, opening blockouts, and exterior walls. All masonry walls at Dresden are unreinforced, except in the bed-joint of every other course; however, bed-joint reinforcement is not considered in the evaluation of the walls. The evaluation and qualification of the Class I masonry walls (i.e., those walls supporting Class I equipment or components) or non-Class I walls whose structural failure may affect Class I equipment or components was performed in response to IE Bulletin 80-11.

The masonry walls are modeled in one of three ways, depending on the supporting configuration. Walls supported from the floor only are modeled as cantilevers. Walls supported at floor and ceiling are modeled as one-way strips. Walls supported on two or more adjacent sides are modeled as two-way plates. The material is considered to be isotropic and elastic.

The wall models are used to determine the fundamental period of the walls. Adjustments to the periods are made to account for openings in the walls. The final periods are then used to select the proper seismic acceleration from the building floor response spectra. The damping level used for the analyses is 2% for both the OBE and SSE. For the generation of attachment loads, the damping levels used for piping are 0.5% for OBE and 2.0% for SSE, for other systems, 1.0% for OBE and 2.0% for SSE. Equivalent static analyses are performed for the inertial considerations. The results of one horizontal analysis and the vertical analyses are combined by absolute summation for the total response.

In addition to inertial loads described in the previous paragraph, interstory drift is also evaluated. In-plane shear strains due to relative displacement between the top and bottom of the walls is calculated. Since none of the masonry walls at Dresden are effectively fixed at the top or bottom, out-of-plane drift is not considered.

Overturning and sliding of cantilever walls are also evaluated to ensure maintaining a factor of safety of 2.0 for OBE and 1.5 for SSE. Movements are evaluated to ensure that safety-related items are not affected.

Masonry walls with mortared joints at both the top and bottom boundaries that do not meet the acceptance criteria for allowable stresses are investigated for arching effects. The wall's capability of resisting horizontal loads, after ultimate tension stresses are exceeded, is developed when the wall jams at the top and bottom against the supporting structural members. The center of the wall cracks due to tension stresses, and a three-hinged arch is formed to resist the loads through compression stresses only.

Design seismic loads generated by the safe shutdown earthquake are based on the peak acceleration of the applicable response criteria and a damping factor of 10% of critical.

The stiffnesses of the supporting structural elements are accounted for in the analysis. Also, the deflection at the center hinge must be less than or equal to one third of the wall thickness. If an arching wall meets the above requirement, it is considered acceptable when the compression stress developed in the arch is less than or equal to the allowable flexural compression stress.

3.7.4 Seismic Instrumentation

A strong motion seismograph is installed at Dresden. The unit is self-contained and is battery-operated from rechargeable batteries.

The seismograph is located in the auxiliary electric equipment computer room on elevation 516'-0". The unit is mounted directly onto the floor and is placed out of the way of normal traffic.

3.7.5 References

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33. Galambos, T.V. and Ravindra, M.K., "Properties of Steel for Use in the LRFD," Journal of the Structural Division, ASCE, September 1978.

34. "Final Report on Bar Tests for the Committee of Reinforcing Bar Producers, AISI," by Wiff, Janney, Elstner and Associates, April 1970.

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36. U. S. Nuclear Regulatory Commission (NRC) Letter from Mr. J. F. Stang, To Mr D. L. Farrar - Commonwealth Edison, Subject - "Systematic Evaluation Program, Topic lll-6, Structural Integrity of Reactor Pressure Vessel - Dresden Nuclear Power Station, Unit 2", (TAC NO. M72906), dated September 10, 1993.

37. NUREG/CR-0891, "Seismic Review of Dresden Unit 2 For the Systematic Evaluation Program", April 1980.

Table 3.7-1

DAMPING FACTORS FOR STRONG VIBRATIONS WITHIN THE ELASTIC LIMIT

Item (Note 1)	Percentage of Critical Damping
Reinforced Concrete Structures	5.0
Steel Frame Structures	2.0
Welded Assemblies	1.0
Bolted and Riveted Assemblies	2.0
Vital Piping Systems	0.5

Notes

1. The damping factors for the core shroud, guide tubes, CRD housing and RPV stabilizer are considered proprietary information and are provided in GENE-771-84-1194, Revision 2, "Dresden Units 2 and 3, Shroud Repair Seismic Analysis".

Table 3.7-2

COMPUTER INPUT DATA FOR SEISMIC ANALYSIS PROGRAM

A. Geometry of Model

1. Vertical distances between mass points

2. Mass point identification (i.e.: Mass 1, Mass 2, etc.)

B. Cross-Section Properties

1. Moments of inertia of structural members

2. Areas of structural members

3. Shear areas of structural members

C. Weights and Masses

1. Weight of each mass point

2. Mass of each mass point

D. Input Earthquake Data

1. Input earthquake – time in seconds and acceleration in gravity units

2. Integration interval to be used in step-by-step solution of Duhamel Integral (0.005 s)

Table 3.7-3

COMPUTER OUTPUT DATA FOR SEISMIC ANALYSIS PROGRAM

A. Maximum absolute displacements of each mass points.

B. Maximum absolute accelerations of each mass points.

C. Maximum absolute shears at each mass point.

D. Maximum absolute overturning moments at each mass point.

E. Natural period of vibration of each mode calculated.

Table 3.7-4

HORIZONTAL SEISMIC COEFFICIENTS

Horizontal Static Coefficient (g)	Piping Located in Building Elevation Zone
1.60	Above 588'-0"
1.25	545'-6" to 588'-0"
1.00	517'-6" to 545'-6"
.70	Below 517'-6"

Table 3.7-5

SPANS FOR VARIOUS DIAMETERS OF PIPING

Horizontal Seismic Loading Conditions 1.0 g and 0.5 g
Seismic Stress Limited to 1500 psi

Nominal Pipe Size (in.)	1/4	3/8	1/2	3/4	1	1 1/2	2	2 1/2	3	4	6	8	10
Span (ft) for 1.0 g	5	5	6	7	7	9	10	11	12	14	17	19	22
Span (ft) for 0.5 g	7	7	9	10	10	13	14	15	17	20	24	28	31

Table 3.7-6

Intentionally Deleted

3.8 DESIGN OF CLASS I STRUCTURES

3.8.1 Concrete Containment

The primary containment function for the Dresden Mark I design is provided by three interconnected steel structures: the drywell, vent system, and pressure suppression chamber (torus or wetwell) as shown in Figure 3.8-1. The concrete reactor building structure, which houses the primary containment for both units, serves as a radiation shield and fulfills a secondary containment function. Structural design of the concrete reactor building is addressed in Section 3.8.4. Functional design of both primary and secondary containment is covered in Section 6.2.

3.8.2 Steel Containment

This section presents key aspects of the structural design of the Dresden primary steel containment. Design elements for the major steel containment components are addressed in Sections 3.8.2.1, 3.8.2.2 and 3.8.2.3 and include:

A. Physical layout,

B. Codes and standards,

C. Loading conditions,

D. Design and analysis procedures,

E. Structural evaluation, and

F. Testing and inservice inspection.

The Mark I primary containment system is designed to condense the steam released during a postulated loss-of-coolant accident (LOCA), to limit the release of fission products associated with such an accident, and to serve as a source of water for the emergency core cooling system (ECCS).

Each Mark I primary containment consists of a drywell, which encloses the reactor vessel, reactor coolant recirculation system, and branch lines of the reactor coolant system; a toroidal-shaped pressure suppression chamber containing a large volume of water; and a vent system connecting the drywell to the water space of the suppression chamber (see Figures 3.8-1, 6.2-1 and 6.2-2).

The drywell is a steel pressure vessel with a spherical lower section and a cylindrical upper section, as shown in Figure 6.2-2. A portion of the lower spherical drywell section is embedded in concrete. This embedment, in combination with lateral restraints which are attached to the cylindrical section, forms the drywell support system. The suppression chamber is a steel pressure vessel, shaped like a torus, encircling and located below the drywell. The suppression chamber is mounted on support structures which transmit loads to the concrete foundation of the reactor building (see Figures 3.8-1, 6.2-1, and 6.2-2).

The drywell and suppression chamber are interconnected by a vent system. Eight main vents connect the drywell to a vent ring header, which is located within the suppression chamber airspace. A bellows assembly is located at the junction where each main vent penetrates the suppression chamber shell to permit differential movement of the suppression chamber and drywell/vent system. Projecting downward from the vent ring header are downcomer pipes, arranged in 48 pairs around the vent header circumference, terminating below the surface of the suppression chamber water volume (Figure 6.2-4).

The original design of the Mark I containment system considered various postulated accident loads associated with the containment design as input into the analysis (see Section 6.2 for a functional description of the containment system). These included pressure and temperature loads resulting from a LOCA, seismic loads, dead loads, jet impingement loads, hydrostatic loads due to water in the suppression chamber, and pressure test loads. Subsequently, while performing large-scale testing for the Mark III containment system and in-plant testing for Mark I primary containment system, new suppression chamber hydrodynamic loads were identified. These hydrodynamic loads are related to the postulated LOCA and safety relief valve (SRV) actuation.

The additional loads result from dynamic effects of drywell atmosphere and steam being rapidly forced into the suppression pool during a postulated LOCA and from suppression pool response to SRV operation generally associated with plant transient operating conditions. Additional details regarding the origin and nature of these hydrodynamic loads are presented in Section 6.2.1. Because these hydrodynamic loads had not been considered in the original design of the containment, a detailed reevaluation was undertaken. This reevaluation, referred to as the Mark I Program, involved tasks performed to restore the originally intended design safety margins for the Dresden plant. The Mark I Program culminated in the issuance of the Plant Unique Analysis Report (PUAR)[1] for Dresden, followed by review and acceptance by the NRC.[2]

The following subsections address structural design of the drywell (Section 3.8.2.1), vent system (Section 3.8.2.2), and suppression chamber (Section 3.8.2.3).

3.8.2.1 Drywell

Due to the physical layout of the drywell, in which the main vent junctions are immediately above the drywell's concrete embedment (see Figure 6.2-2), the main vents are anchored to the drywell shell. With the exception of the main vent junctions, the drywell was not reevaluated under the Mark I Program. The following subsections thus describe original drywell structural design.

3.8.2.1.1 Description of Structure

Drywell dimensions were dictated by the need to enclose the reactor vessel and associated auxiliary equipment. The governing thermal-hydraulic aspects for containment sizing are addressed in Section 6.2.1.

The drywell spherical section is 66 feet in diameter and varies in thickness from $\frac{13}{16}$ inches to $1\frac{1}{16}$ inches. The cylindrical neck section is 37 feet in diameter and varies in thickness from $\frac{3}{4}$ inches to $1\frac{1}{2}$ inches. The spherical to cylindrical transition is $2\frac{3}{4}$ inches thick. The removable top head ranges from $1\frac{1}{4}$ inches to $1\frac{7}{16}$ inches in thickness. The drywell stands 111 feet, 11 inches tall. Drywell materials are described in Section 6.2.1.

As noted in the arrangement drawings, the drywell bottom is filled with concrete. Beneath the drywell is concrete fill from the spring line down. These concrete fills are in contact with an internal continuous steel ring on the interior and the steel support skirt on the exterior. These shear rings transmit the seismic shear loads into the building foundation and result in the drywell base and the reactor building acting as a unit under seismic loads. The upper portion of the drywell is supported by stabilizers and a truss arrangement to the reactor vessel and shield wall at elevation 575'-2". These systems transmit the upper lateral seismic loads to the reactor building. Thus all the vertical and seismic loads are transmitted directly to the reactor building and do not require additional support structures.

3.8.2.1.2 Applicable Codes, Standards, and Specifications

The primary containments were designed, erected, pressure-tested, and N-stamped in accordance with the ASME Code Section III, 1965 Edition including addenda through Summer 1965.

Repairs/Replacements of primary containment components (class MC) are performed in accordance with ASME Code Section XI, Subsection IWE, 2001 Edition with 2003 Addenda.

3.8.2.1.3 Loads and Load Combinations

The loads applicable to the design of the drywell are defined as follows:

D = dead load of structure and equipment plus any other permanent loads contributing stress, such as soil or hydrostatic loads or operating pressures, and live loads expected to be present when the plant is operating

P = pressure due to LOCA

R = jet force or pressure on structure due to rupture of any one pipe

H = force on structure due to thermal expansion of pipes under operating conditions

T = thermal load on containment, reactor vessel, and internals due to LOCA

E = operating basis earthquake (OBE) load (0.10 g horizontal ground acceleration, 0.067 vertical ground acceleration)

E' = safe shutdown earthquake (SSE) load (0.20 g horizontal ground acceleration, 0.133 g vertical ground acceleration)

The load combinations and allowable stresses applicable to the design of the drywell are defined as follows:

Load Combination	Allowable Stress
D + P + H + T + E	ASME Section III, Class B, without the usual increase for seismic loadings.
D + P + R + H + T + E	Same as the preceding, except local yielding is permitted in the area of the jet force where the shell is backed up by concrete. In areas not backed up by concrete, primary local membrane stresses at the jet force do not exceed 0.90 of the material yield point at 300°F.
D + P + R + H + T + E′	Primary membrane stresses, in general, do not exceed the yield point of the material. If the total stress exceeded the yield point, an analysis was made to determine that the energy absorption capacity exceeded the energy input from the earthquake. The same criteria as the preceding are applied to the effect of jet forces for this loading condition.

3.8.2.1.4 Design and Analysis Procedures

Design requirements for the drywell include provisions for resisting dead, live, and operating loads and additional special loads. Potential seismic loads are addressed in Section 3.7. Section 6.2 describes the drywell design provisions for thermal expansion loading.

3.8.2.1.5 Structural Evaluation

The drywell bottom is fixed in the foundation concrete by tension in the drywell skirt and compression on the concrete base. The drywell skirt is a 5/8-inch plate connected to the drywell by a full penetration weld. The bearing of the skirt plate on the concrete is 45 psi for the operating basis earthquake and 90 psi for the safe shutdown earthquake with the drywell flooded. For uplift, the stress in the skirt plate and weld is 1030 psi for the operating basis earthquake and 2060 psi for the safe shutdown earthquake. The tension in the anchor bolts connecting the skirt plate to the foundation is 9800 psi for the operating basis earthquake and 19,600 psi for the safe shutdown earthquake. The weight of the water in the flooded drywell was neglected for the uplift calculations. The anchor bolts are embedded in the concrete foundation with a bearing plate on the end, at a distance sufficient to develop the full tensile capability of the bolt.

The drywell plates are restrained in the area of the skirt connection by the concrete inside and outside of the drywell. Therefore, this plate has membrane stresses only. For the operating basis earthquake (OBE), this stress is 2750 psi; for the safe shutdown earthquake (SSE), the stress is 5300 psi. Both of these values are within the working stress allowables.

The concrete structure inside the drywell that supports the reactor pedestal is prevented from sliding by a 6-inch x 1-inch continuous shear plate. This shear plate stands vertically, is welded to the inside of the drywell, and is located on the same diameter as the drywell skirt. The shear in the two 3/8-inch fillet welds attaching this shear plate to the drywell under the OBE is 3550 psi. The bearing on the concrete for this condition is 315 psi. These values are within the allowable of 15,800 psi for the weld metal and 935 psi for the concrete. For the SSE, the stress in the weld is 7000 psi and in bearing on the concrete 615 psi. The weld stress and the concrete allowable bearing stress for the SSE is still below the working stress allowables.

Absolute seismic acceleration curves were developed to give an envelope of the maximum absolute accelerations with respect to height. Moment, shear, and displacement curves were also developed. Instead of directly using the absolute acceleration curves, the moment, shear, and displacement curves were used in the seismic design of the drywell. Figures 3.8-2 through 3.8-7, present the maximum displacements, shears, and moments in both the east-west and north-south directions.

In the drywell/reactor building/turbine building analysis, some minor discrepancies in the displacements of the interconnecting elements resulted due to:

A. The plotting procedures, whereby a displacement is calculated for each mass point and a continuous curve drawn through these points, and

B. The deflection of the spring connections shown between the reactor and turbine buildings.

The NRC performed an independent review, under Systematic Evaluation Program Topic III-7.B, of the Dresden Unit 2 drywell and concluded that the drywell will perform its intended function when subjected to combined seismic-LOCA loads.

3.8.2.1.6 Testing and Inservice Inspection Requirements

Pressure and leakage rate testing of the containment system are addressed in Section 6.2.6.

Primary containment repairs, replacements, and examinations are done in accordance with ASME Section XI, Subsection IWE, 2001 Edition with 2003 Addenda.

3.8.2.1.7 Flued Head Penetrations

Fluid pipe penetrations are of two general types; i.e., those which accommodate thermal movement and those which experience relatively little thermal stress. Fluid piping penetrations for which movement provisions are made are high-temperature lines such as the main steam line and certain other reactor auxiliary and cooling system lines. Typical penetrations of this type are shown in Figures 3.8-8, 3.8-9A and 3.8-9B. These penetrations have a guard pipe between the hot line and the penetration nozzle and a two-ply expansion bellows between the penetration nozzle and the flued head. This configuration permits the penetration to be vented to the drywell should a rupture of the hot line occur within the penetration.

The guard pipes are designed to the same pressure and temperature as the fluid line and are attached to a multiple flued head fitting, a forging with integral flues or nozzles. This fitting was designed to conform to ASME Section VIII. The penetration sleeve is welded to the steel drywell and extends through the concrete containment shield wall where it is welded to a bellows, which in turn is welded to the guard pipe. The bellows accommodates the thermal expansion of the steam pipe and steel drywell relative to the steam pipe. A double bellows arrangement permits remote leakage testing of the penetration seal. The lines are anchored at one end of the penetration assembly to limit the movement of the line relative to the containment yet permit pipe movement parallel to the penetration.

The only lines which connect to a high-pressure system which do not have a double seal penetration sleeve are the hydraulic lines to the control rod drives. These comprise 354 small, stainless steel lines, shop-welded to three sections of the drywell plate. The mechanical problems involved with this number of small penetrations in a relatively small area make it impractical to provide individual penetration sleeves. The pipes are designed to deflect with the drywell shell. They are not individually testable but are tested as part of the overall containment leakage rate test.

Penetration details of cold piping lines are shown on Figure 3.8-10. The pipe sleeve which attaches to the drywell is designed for 62 psig but can withstand a substantially higher pressure due to the use of heavy wall pipe. No bellows are required, since thermal expansion is minimal. A tabulation of the type of penetration used for each service is shown in Tables 3.8-1 and 3.8-2 for Unit 2 and 3, respectively.

Lines which open directly to the containment do not have a separate penetration sleeve and are welded directly to the containment shell.

Modifications have been made to replace flued head anchors for containment penetrations X-113, X-108A, and X-109A with new flued head anchor structures with increased load capacity. The new anchor structures resist pipe loads due to pipe breaks or seismic events.

3.8.2.1.8 Electrical Penetrations

Electrical penetration seals were designed to accommodate the electrical requirements of the plant. These are functionally grouped into low-voltage power and control cable penetration assemblies, high-voltage power cable penetration assemblies, and shielded cable penetration assemblies. Each penetration seal has the same basic configuration shown in Figure 3.8-11. An assembly is sized to be inserted in the 12-inch Schedule 80 penetration nozzles which are furnished as part of the containment structure. Installation of the penetration assembly is accomplished by inserting it from either side of the containment into the penetration nozzle. Three field welds are required to complete the installation of the assembly in the penetration nozzle.

Headerplates conforming to the inner diameter of the penetration nozzle are provided at each end of the penetration assembly, forming a double pressure barrier. Radiation shielding is attached to the penetrations on the drywell side to provide external access to the electrical connections during plant operation.

The design and fabrication of each type of penetration assembly is in accordance with the requirements of ASME Section III, Class B, and materials of construction are self-extinguishing in accordance with ASTM-D635. The electrical penetrations were designed to withstand environmental conditions present during a postulated LOCA, as well as to maintain containment integrity for extended periods of time with a post-accident environment. These conditions, including the normal operating environmental conditions, are shown in Table 3.11-1, 3.11-2, and 3.11-3.

The installed assemblies are designed to withstand a continuous internal pressure of 125 psi during normal environmental conditions and to meet a leakage rate of 1.16 x 10^{-6} cc/s when pressurized to 63 psig with dry helium at an ambient temperature of 175°F. The latter condition is verified prior to installation in the primary containment.

The low-voltage assembly is suitable for voltages 600V or less and is designed for conductors varying in size from 18 to 4/0 AWG. The cables are grouped and passed through openings in the headerplates as shown in Figure 3.8-12. A potting compound is applied at each end of the penetration to seal the assemblies. A cable lead is terminated at either a splice or an environment-resistant connector. The maximum wire density is restricted to 42% of the end flange cross-sectional area.

Shielded signal cables are provided to interconnect low-noise circuits between the reactor and the control room; in particular, the reactor neutron monitoring channels. Figure 3.8-13 shows a cutaway view of the containment penetration assembly for shielded signal cables. One type of circuit uses coax connectors mounted directly on the headerplates and isolated from ground. Another type of circuit uses connectors mounted on the penetration assembly auxiliary structure. The cable density is restricted to one circuit per three square inches of headerplate surface for the first type and approximately 80 circuits of the latter type for each 12-inch penetration nozzle.

A sectional view of the high-voltage power cable penetration assembly is shown on Figure 3.8-14. The penetration assembly accommodates voltages up to 5 kV and cables as large as 1000 MCM and is designed to maintain low gas leakage rates and high insulation resistance. The high-voltage cables are passed through openings in the headerplates, and potting compound is applied to both sides of the headerplates to effect a pressure seal. The headerplates are constructed of stainless steel, a nonmagnetic material, in order to eliminate the possibility of eddy current heating.

3.8.2.1.9 Instrument Line Penetrations

Instrument line penetrations for the drywell are shown in Table 3.8-4. The following discussion, including descriptions and quantities, is applicable to Unit 3; the descriptions are generally applicable to Unit 2 also.

Dresden Unit 3 contains 16 penetration assemblies which are used for primary system instrumentation. Each of these assemblies is configured to carry six instrument pipes through the concrete containment shield wall. Of the total of 96 penetrating pipes, 77 are active lines and 19 are spares. Each of the active penetrating pipes are equipped with stop valves and excess flow check valves located outside the containment as indicated in Table 3.8-4.

"In addition to the stop and excess flow check valves on the Reactor Vessel Water Level A-Loop and B-Loop (penetrations X-209 and X-108B respectively), these lines have Reactor Vessel Water Level Instrumentation System (RVWLIS) Backfill interfaces upstream of the stop valves. Each of the RVWLIS Backfill lines has two simple 3/8-inch tubing, Type 316 stainless steel check valves in series acting as primary containment isolation valves."

The penetrating lines are 1-inch Schedule 80, Type 304 stainless steel pipe. Each of the lines is welded to a stainless steel pipe which is welded to the drywell penetration housing. A detail of a typical multiple pipe instrument penetration is shown in Figure 3.8-15.

Within the secondary containment are 1-inch process stop valves, flow check valves, and 1/2-inch Schedule 80, Type 304 stainless steel piping to the instrument rack. Piping or stainless steel tubing is used within the rack to the sensor. All welds have been dye-penetrant tested. Analyses have been performed to assure that the installation from the penetration to the instrument rack meet seismic Class I requirements.

The two reactor recirculation pump No. 1 seal cavity instrument lines are interconnected with the reactor recirculation pump seal purge lines between the excess flow check valves and the instruments. Redundant, safety related check valves are installed in each seal purge line in close proximity to the containment penetrations. The piping between the excess flow check valves and the safety related seal purge line check valves is seismically designed, consistent with the design of the piping to the instruments. See Section 4.6.4.6. for further discussion of the seal purge line check valves.

Each process stop valve and excess flow check valve is Type 304 or Type 316 stainless steel. The excess flow check valves permit a maximum flow of 2 gal/min. A detail of a typical penetrating pipe installation is shown in Figure 3.8-16. There are three exceptions where the process stop valves are more than 12 inches from the penetration. These are for penetrations X-130, X-131, and X-135, all of which are located below grade.

No special protection has been provided for any of the instrument lines within the secondary containment. An analysis has been performed of the consequences of a 1-inch instrument line break in the Dresden Unit 3 plant. The break was assumed to occur outside the primary containment but upstream of the excess flow check valve in the line. A manually operated stop valve located outside the containment wall upstream of the break was not assumed to be closed until after the reactor was shut down and depressurized. The reactor was assumed to be shut down manually by the operator upon detection of the break, i.e., by audible disclosure or by detection of increased radiation level in the reactor building or water level increase in the reactor building sumps. The results of this analysis, including secondary containment integrity and radiological consequences, are addressed in Section 15.6.2.

The instrumentation required to either shutdown the reactor or initiate core cooling functions are separated so that a single event cannot jeopardize these functions, as discussed in Section 7.6.2.

3.8.2.2 Vent System

3.8.2.2.1 Description of Structure

The vent systems for Units 2 and 3 are constructed from cylindrical shell segments joined together to form a manifold-like structure connecting the drywell to the suppression chamber. Figures 3.8-1, 6.2-2, and 6.2-3 show the configuration of the vent system. The major components of the vent system include the vent lines, vent line/vent header spherical junctions, vent header, and downcomers. Figures 3.8-17, 3.8-18, 6.2-4 and 6.2-5 show the proximity of the vent systems to other containment components.

The eight vent lines connect the drywell to the vent header in alternate mitered cylinders or bays of the suppression chamber. The vent lines are nominally 1/4 inch thick and have an inside diameter (ID) of 6 feet, 9 inches. The upper ends of the vent lines include conical transition segments at the penetration to the drywell (Figure 3.8-24). The drywell insert plate around each vent line/drywell penetration is 2¼ inches thick, with a 3⅝-inch thick cylindrical nozzle. The vent lines are shielded from jet impingement loads at each vent line/drywell penetration location by jet deflectors which span the openings of the vent lines. The eight vent line/vent header spherical junctions connect the vent lines and the vent header (Figure 3.8-25). Each spherical junction is constructed from six shell segments with thicknesses varying from ¼ inch to ¾ inches. The spherical junctions all have a 1-inch diameter drain line extending from the bottom of the spherical junction to below the suppression pool surface.

The safety relief valve discharge lines (SRVDLs) are routed from the drywell through the vent lines and penetrate the vent lines inside the suppression chamber. Section 3.9.3.1 provides a discussion of the analysis of SRVDL piping.

The vent header is a continuous assembly of mitered cylindrical shell segments joined together to form a ring header (Figure 6.2-3). The vent header is ¼-inch thick and has an ID of 4 feet, 10 inches.

Ninety-six downcomers penetrate the vent header in pairs (Figures 3.8-26 and 6.2-3). Two downcomer pairs are located in each vent line bay; four pairs are located in each nonvent line bay. Each downcomer consists of an inclined segment which penetrates the vent header and a vertical segment which terminates below the surface of the suppression pool (Figures 3.8-26, 3.8-27, and 3.8-28). The inclined segment is ½ inch thick and the vertical segment is ¼ inch thick. The inside diameters of the inclined and vertical portions of the downcomer are 2.0 feet.

Full penetration welds connect the vent lines to the drywell, the vent lines to the spherical junctions, the spherical junctions to the vent header, and the downcomers to the vent header. Therefore, the connections of the major vent system components are capable of developing the full capacity of the associated major components themselves.

The intersections of the downcomers and the vent header are reinforced with a system of stiffener plates and bracing members (Figures 3.8-26, 3.8-27, and 3.8-28). In the plane of the downcomer pairs, the intersections are stiffened by a pair of ½-inch stiffener plates located between each set of the downcomers and a pair of lateral bracing pipe members at the bottom of each set of two downcomers. The stiffener plates are welded both to the tangent points of the downcomer legs and to the vent header. The lateral bracing members are welded to the downcomer rings near the tangent points. The system of stiffener plates is designed to reduce local intersection stresses caused by loads acting in the plane of the downcomers. The system of lateral bracing ties the downcomer legs together in a pair; therefore, separation forces on the pair of downcomer legs will be taken as axial forces in the bracing.

In the direction normal to the plane of the downcomer pair, the downcomers are braced by a longitudinal bracing system. In Dresden Unit 2 these bracings are located in those vent line bays which house the SRV discharge line, and which extend to midlength of the neighboring nonvent bays (Figure 3.8-26). In this manner, 62% of all the downcomers are braced longitudinally. However, in Dresden Unit 3 all 96 downcomers are braced longitudinally. The longitudinal bracing patterns for the two Dresden units are shown in Figures 3.8-29 and 3.8-30. The ends of the horizontal pipe members near miter joints and centerlines of the nonvent bays are welded to the downcomer rings. The 3-inch x 1-inch diagonal members and their adjacent horizontal pipe members are connected to lugs which are welded to the downcomers.

This bracing system provides an additional load path for the transfer of loads acting on the submerged portion of the downcomers and results in reduced local stresses in the downcomer/vent header intersection regions. The system of downcomer/vent header intersection stiffener plates and lateral bracing members provides a redundant mechanism for the transfer of loads acting on the downcomers, thus reducing the magnitude of loads passing directly through the intersection. The longitudinal bracing also ties together several pairs of downcomers in the longitudinal direction, causing an increase in stiffness to the overall system that minimizes the dynamic effect of several loads, including SRV loads on submerged structures. This also results in load sharing among the downcomers for the SRV loads on submerged structures.

A bellows assembly is provided at the penetration of the vent line to the suppression chamber (Figure 3.8-24). The bellows allows differential movement of the vent system and suppression chamber to occur without developing significant interaction loads. Each bellows assembly consists of a stainless steel bellows unit connected to a $2\frac{1}{8}$-inch thick nozzle. The bellows unit has a 7-foot, 5-inch ID and contains five convolutions which connect to a $\frac{1}{2}$-inch thick cylindrical sleeve at the vent line and a 1-inch thick cylindrical sleeve at the torus nozzle end. A $1\frac{1}{2}$-inch thick annular plate welded to the vent line connects to the upper end of the bellows assembly by full penetration welds. The lower end of the bellows assembly is a $2\frac{1}{8}$-inch thick nozzle, already described, which is connected to the suppression chamber shell insert plate by full penetration welds. The overall length of the bellows assembly is 3 feet, $2\frac{3}{4}$ inches.

Vent header deflectors are provided in both the vent line bays and the nonvent line bays (Figures 3.8-18, 3.8-27, and 3.8-28). The deflectors shield the vent header from pool swell impact loads which are postulated to occur during the initial phase of a LOCA event. The vent header deflectors are constructed from 20-inch diameter, Schedule 100 pipe. The vent header deflectors are supported by 1-inch thick connection plates that are welded to the vent header support collar plates near each miter joint.

The drywell/wetwell vacuum breakers are nominal 18-inch units. There are two vacuum breakers in each vacuum breaker header. There are six vacuum breaker headers on the suppression chamber. The headers originate as a 30-inch outside diameter (OD) vertical penetration at the upper outside quadrant of six different vent line bays. This penetration is reinforced by a $1\frac{1}{2}$-inch thick insert plate at each location. The header then leaves the 30-inch penetration as two separate, horizontal 18-inch OD lines where the vacuum breakers are contained. After the two vacuum breakers, the two 18-inch OD lines come together again into a 24-inch OD line. A 24-inch diameter bellows assembly immediately follows this intersection. The header continues as a 24-inch diameter line from the bellows to the vent line/drywell penetration. This 24-inch diameter vent line penetration is reinforced with a 33-inch diameter by $\frac{3}{4}$-inch thick insert plate.

The vent system is supported vertically by two column members at each miter joint location (Figures 3.8-31, 3.8-32, and 6.2-6). The support column members are constructed from 6-inch diameter, Schedule 80 pipe. The upper ends of the support columns are connected to the 1-inch thick vent header support collar plates by

2¾-inch diameter pins. The support collar plates are attached to the vent header with 5/16-inch fillet welds. The support column loads are transferred at the upper pin locations by ¾-inch thick pin plates. The lower ends of support columns are attached to 1-inch thick ring girder pin plates with 2¾-inch diameter pins and ¾-inch thick pin plates. The support column assemblies are designed to transfer vertical loads acting on the vent system to the suppression chamber ring girders, while simultaneously resisting drag loads on submerged structures.

The vent system is supported horizontally by the vent lines which transfer lateral loads acting on the vent system to the drywell at the vent line/drywell penetration locations. The vent lines also provide additional vertical support for the vent system, even though the vent system support columns provide primary vertical support. Since the relative stiffness of the bellows with respect to other vent system components is small, the support provided by the vent line bellows is negligible.

The vent system also provides support for a portion of the SRVDL piping inside the vent line and suppression chamber (Figures 3.8-24 and 6.2-4). Loads acting on the SRVDL piping are transferred to the vent system by the penetration assembly and internal supports on the vent line. Conversely, loads acting on the vent system cause motions to be transferred to the SRVDL piping at the same support locations. Since the relative stiffness of the SRVDL piping with respect to other vent system components is small, the support provided by the SRVDL piping to the vent system is negligible.

3.8.2.2.2 Applicable Codes, Standards, and Specifications

The primary containment, including the vent system described herein, was originally designed, erected, pressure-tested, and N-stamped in accordance with the ASME Code Section III, 1965 Edition including addenda through Summer 1965.

For the Mark I Program reevaluation, the acceptance criteria generally followed the ASME Code Section III, 1977 Edition with addenda up to and including Summer 1977 for metal containment (Class MC) components and component supports. Further detail regarding structural acceptance criteria may be found in Section 3.8.2.2.5.

Repairs/Replacements of primary containment components (class MC) are performed in accordance with ASME Code Section XI, Subsection IWE, 2001 Edition with 2003 Addenda.

3.8.2.2.3 Loads and Load Combinations

The loads for which the vent system was evaluated are defined in NUREG-0661[3] on a generic basis for all Mark I plants. Mark I program loads have been defined in a separate report known as the Plant Unique Load Definition (PULD)[4] for Dresden. The PULD essentially implemented the methodologies defined in NUREG-0661.

The loads acting on the vent system are categorized as follows:

A. Dead weight loads,

B. Seismic loads,

C. Pressure and temperature loads,

D. Vent system discharge loads,

E. Pool swell loads,

F. Condensation oscillation loads,

G. Chugging loads,

H. SRV discharge loads,

I. Piping reaction loads, and

J. Containment interaction loads.

Dead weight, seismic, and pressure and temperature loads were considered in the original containment design. The Mark I Program identified additional pressure and temperature loads as a result of postulated LOCA and SRV discharge flows. Section 3.8.2.1.3 describes the design pressure loads applicable to the vent system.

Not all of the loads defined in NUREG-0661 have been evaluated in detail since some are enveloped by others or have a negligible effect on the vent system. Only those loads which maximize the vent system response and lead to controlling stresses have been fully evaluated and are described in the PUAR.[1]

3.8.2.2.4 Design and Analysis Procedures

With the exception of a few minor differences, the vent system geometry for Dresden Units 2 and 3 is identical to that of Quad Cities Units 1 and 2. These differences are as follows:

A. The vent line angle of inclination at Dresden is approximately 1° higher than at Quad Cities.

B. The Dresden units' drywell/vent line penetrations include a 1/2-inch thick conical transition segment connected to a 3 5/8-inch thick cylindrical nozzle at the drywell ends. The Quad Cities' penetrations include a 1/2-inch thick spherical transition segment connected to a 3-inch thick nozzle at the drywell ends.

C. The inclined portion of the downcomer is 1/2 inch thick in the Dresden units; whereas, in Quad Cities it is 3/8 inches thick.

D. The vacuum breakers in the Dresden units are located outside of the suppression chambers, and their headers penetrate the vent lines near the drywell ends; in the Quad Cities units they are attached to the vent line/vent header spherical junctions.

The effect of these differences in the overall vent system analysis was investigated and found to be insignificant. Therefore, the analyses were performed on analytical models which are based on plant unique geometry for Quad Cities Units 1 and 2. Various models used in the analysis are described in the following paragraphs.

With the exception of the nonrepetitive pattern of the downcomer longitudinal bracing system in Dresden Unit 2, the repetitive nature of the vent system geometry is such that the vent system can be divided into 16 identical segments which extend from midbay of the vent line bay to midbay of the nonvent line bay (Figure 6.2-3). To account for the nonrepetitive pattern of the longitudinal bracing system in Dresden Unit 2, two conditions were idealized. First, it was assumed the bracing system is included in the 1/16 segment. In this assumption, all 96 downcomers were assumed to be braced longitudinally (100% bracing condition). Second, it was assumed that the 1/16 segments do not include any bracing system. With this assumption, a nonbracing condition was developed. These two idealized conditions bound any particular bracing condition which might exist in any particular 1/16 segment of the two Dresden vent systems. The analysis of the vent system for the majority of the governing loads was therefore performed for the two 1/16 segments described previously.

Two beam models of the 1/16 segment reflecting the preceding conditions were used to obtain the response of the vent system to all loads, except those resulting in asymmetric effects on the vent system. The resulting responses from the two models were compared and the more severe was selected for code evaluation. The models included the vent line, the vent header, the downcomers, the support columns, and the downcomer lateral bracings. The longitudinal bracing was also included in one model.

The local stiffness effects at the penetrations and intersections of the major vent system components (Figures 3.8-24 through 3.8-28) were included by using stiffness matrix elements of these penetrations and intersections. A matrix element for the vent line/drywell penetration, which connects the upper end of the vent line to the transition segment, was developed using the finite difference model of the penetration. A matrix element which connects the lower end of the vent line to the beams on the centerline of the vent header and to the beams on the centerline of the vacuum breaker nozzles, was developed using the finite element model of the vent line/vent header spherical junction.

Finite element models of each downcomer/vent header intersection were used to develop matrix elements which connect the beams on the centerline of the vent header to the upper ends of the downcomers at the downcomer miters.

The node spacings used in the two analytical models were identical and were refined to ensure adequate distribution of mass, to determine component frequencies and mode shapes, and to facilitate accurate load application. The stiffness and mass properties used in the two models were identical and were based on the nominal dimensions and densities of the materials used to construct the vent system. Small displacement linear-elastic behavior was assumed throughout. Further details concerning the vent system models and boundary conditions are provided in the PUAR.[1]

Dynamic analyses using the two 1/16 beam models of the vent system were performed for the pool swell loads and condensation oscillation loads specified in
Section 3.8.2.2.3. The analyses consisted of a transient analysis for pool swell loads

and harmonic analysis for condensation oscillation loads. The modal superposition technique with 2% damping was utilized in both the transient and harmonic analyses. The pool swell and condensation oscillation load frequencies were enveloped by including vent system frequencies to 100 Hz and 50 Hz, respectively.

The remaining vent system load cases specified in Section 3.8.2.2.3 involve either static loads or dynamic loads, which were evaluated using an equivalent static approach. For dynamic loads, conservative dynamic amplification factors were developed and applied to the maximum spatial distributions of the individual dynamic loadings.

The two 1/16 beam models were also used to generate loads for the evaluation of stresses in the major vent system component penetrations and intersections. Beam end loads, distributed loads, reaction loads, and inertia loads were developed from the two models and the critical cases were applied to the detailed analytical models of the vent system penetrations and intersections.

A beam model of a 180° segment of the vent system, based on the Dresden Unit 2 downcomer longitudinal bracing configuration (Figure 3.8-29) was used to obtain the response of the vent system to asymmetric loads. The Dresden Unit 2 bracing pattern was selected since a maximum number of unbraced downcomers are grouped together in one area, thus enveloping the other unit's configuration. The plane of symmetry due to the uniqueness of the bracing pattern is at a 45° clockwise rotation from true north (Figure 3.8-29). Another 180° beam model based on the Dresden Unit 3 downcomer longitudinal bracing configurations (Figure 3.8-30) is also used to obtain the response of the vent system to asymmetric loads. The resulting responses from the two beam models are compared and the more severe of the two is selected for code evaluation. The two models include the vent lines, the spherical junctions, the vent header, downcomers, downcomer lateral bracing, the downcomer longitudinal bracing, and the vent header deflector.

Many of the modeling techniques used in the 180° beam model, such as those used for local mass and stiffness determination, are the same as those utilized in the 1/16 beam model of the vent system. The local stiffness effects at the vent line-drywell penetrations, vent line/vent header spherical junctions, and the downcomer-vent header intersections were included using stiffness matrix elements for these penetrations and intersections. Pin conditions were modeled at the attachments of the support columns to the suppression chamber.

The asymmetric loads which act on the vent system are horizontal seismic loads and asymmetric chugging loads. An equivalent static analysis was performed for each of the loads using the 180° beam model.

The penetrations and intersections of the major components of the vent system were evaluated using refined analytical models of each penetration and intersection. These include the vent line/drywell penetration, the vent line-vent header spherical junction, and the downcomer/vent header intersections.

Each of the penetration and intersection analytical models includes mesh refinement near discontinuities to facilitate evaluation of local stresses. The stiffness properties used in the model are based on the nominal dimensions of the materials used to construct the penetrations and intersections. Small displacement linear-elastic theory was applied throughout.

The analytical models were also used to evaluate stresses in the penetrations and intersections. Stresses were computed by idealizing the penetrations and intersections as free bodies in equilibrium under a set of statically applied loads. The applied loads, which were extracted from either of the two 1/16 beam model results or from either of the two 180° beam models results, consist of loads acting on the penetration and intersection model boundaries and of loads acting on the interior of penetration and intersection models. The loads acting on the penetration and intersection model boundaries are the beam end loads taken from the vent system at nodes coincident with the penetration or intersection model boundary locations.

3.8.2.2.5 Structural Evaluation

The NUREG-0661[3] acceptance criteria on which the vent system analysis is based follow the ASME Code Section III, 1977 Edition, including the Summer 1977 Addenda for Class MC components and component supports. The corresponding service level assignments, jurisdictional boundaries, allowable stresses, and fatigue requirements are consistent with those contained in the applicable subsections of the ASME Code and the Plant Unique Analysis Applications Guide (PUAAG).[5]

The items evaluated in the analysis of the vent system are the vent lines, the spherical junctions, the vent header, the downcomers, the downcomer ring plates, the support columns and associated support elements, the drywell shell near the vent line penetrations, the vent header deflectors, the downcomer/vent header intersection stiffener plates, the downcomer bracing systems, the vent header support collar, and the vent line bellows assemblies.

The vent lines, the vent line/vent header spherical junctions, the vent header, the downcomers, the drywell shell, the downcomer/vent header intersection stiffener plates, the downcomer ring plates, and the vent header support collars were evaluated in accordance with the requirements for Class MC components contained in Subsection NE of the ASME Code. Fillet welds and partial penetration welds joining these components or attaching other structures to them were also examined in accordance with the requirements for Class MC welds contained in Subsection NE of the ASME Code.

The support columns, the downcomer bracing members, and the associated connecting elements and welds were evaluated in accordance with the requirements contained in Subsection NF of the ASME Code for Class MC component supports. The vent header deflectors and associated components and welds were also evaluated in accordance with the requirements for Class MC components supports, with allowable stresses corresponding to ASME Code, Subsection NF Service Level D limits.

The allowable stresses for all the major components of the vent system, such as the vent line, the spherical junctions, the vent header, and the downcomers, were determined at the maximum DBA temperature of 284°F per the Plant Unique Load Definition (Reference 4). Table 3.8-5 shows the allowable stresses for the load combinations with ASME Code Service Level B and C limits.

The portion of the SRVDL within the limits of reinforcement normal to the vent line penetration (both above and below the vent shell) is classified as a Class MC component for analysis purposes. This segment of piping was evaluated as part of the SRVDL, which was analyzed according to the methods summarized in Section 3.9.3.1.3.

As permitted in ASME Code Subsection NCA, Class 1 piping rules were employed in the stress analysis of this section of the SRVDL piping. Class MC material stress allowable were used, however. Acceptance criteria are therefore based on the requirements of Code Subsection NB, and are summarized in Table 3.8-6.

Table 3.8-7 shows the maximum stresses and associated design margins for the major vent system components, component supports, and welds for the controlling load combinations.

Table 3.8-8 summarizes the Class MC SRVDL stress and fatigue results. The calculated and Code allowable stresses are given for each applicable Code equation for each service level. The calculated and allowable fatigue usage factor is also given for the applicable service level.

As demonstrated in the results, after completion of the modifications described in Section 6.2, all of the vent system results are within acceptance limits.

3.8.2.2.6 Testing and Inservice Inspection Requirements

Pressure and leakage rate testing of the containment system is addressed in Section 6.2.6.

Primary containment repairs, replacements, and examinations are performed in accordance with ASME Section XI, Subsection IWE, 2001 Edition with 2003 Addenda.

3.8.2.3 Suppression Chamber

3.8.2.3.1 Description of Structure

The suppression chamber is constructed from 16 mitered cylindrical shell segments joined together in the shape of a torus. Figure 6.2-3 illustrates the configuration of the suppression chamber. Figures 3.8-1 and 3.8-17 through 3.8-19 show the proximity of the suppression chamber to other components of the containment.

The suppression chamber is connected to the drywell by eight vent lines which, in turn, are connected to a common vent header within the suppression chamber. Attached to the vent header are downcomers which terminate below the surface of the suppression pool. The vent system is supported vertically at each miter joint by two support columns which transfer reaction loads to the suppression chamber (Figure 6.2-5). A bellows assembly is provided at the penetration of the vent line to the suppression chamber to allow differential movement of the suppression chamber and vent system to occur (Figure 6.2-4).

The major radius of the suppression chamber is 54.5 feet, measured at midbay of each mitered cylinder. The ID of the mitered cylinders which make up the suppression chamber is 30.0 feet. The suppression chamber shell thickness is typically 0.585 inches above the horizontal centerline and 0.653 inches below the horizontal centerline, except at penetrations, where it is locally thickened (Figure 6.2-4).

The suppression chamber shell is reinforced at each mitered joint location by a T-shaped ring girder (Figures 6.2-5, 3.8-19 and 3.8-21). A typical ring girder is located in a plane 4 inches from the miter joint and on the nonvent line bay side of each miter joint. As such, the intersection of a ring girder web and the suppression chamber shell is an ellipse. The inner flange of a ring girder is rolled to a constant inside radius of 12 feet, $10\frac{3}{4}$ inches. Thus the ring girder web depth varies from 24 to $27\frac{1}{2}$ inches and has a constant thickness of 1 inch. The ring girder flange is attached to the ring girder web with $\frac{5}{16}$-inch fillet welds. The ring girder web is attached to the suppression chamber shell with $\frac{3}{8}$-inch fillet welds (Figures 3.8-21 and 3.8-22).

The ring girders are laterally reinforced at the base of the vent header support columns by 1-inch thick plate assemblies (Figure 3.8-14). Both units have five such assemblies in the bays with SRVDLs. In the non-SRVDL bays, there are no such assemblies in Unit 2, and two assemblies in Unit 3. In addition to these lateral stiffeners, the ring-girder-web plate-to-torus-shell fillet weld was increased from $\frac{3}{8}$ inches to $\frac{5}{8}$ inches over a 12.0-foot long arc near the outside torus support column (Figure 3.8-19).

The suppression chamber is supported vertically at each miter joint by inside and outside columns and by a saddle support (Figures 6.2-5, 3.8-19, and 3.8-20). The connection web plate is parallel to the mitered joint. During construction, the support columns were jacked radially outward before being bolted to the basemat, thus imposing a $\frac{3}{16}$-inch preset in the inside column and $\frac{11}{16}$-inch in the outside column. The saddle supports are located parallel to the associated miter joint and in the plane of the ring girder web. At each miter joint, the ring girder, the columns, the column connections, and the saddle support form an integral support system, which takes vertical loads acting on the suppression chamber shell and transfers them to the reactor building basemat. The support system provides full vertical support for the suppression chamber, at the same time allowing radial movement and thermal expansion to occur.

Figure 6.2-5 shows that the vertical support system provides a load transfer mechanism which acts to reduce local suppression chamber shell stresses and to more evenly distribute reaction loads to the basemat. The vertical support system also acts to raise the suppression chamber natural frequencies beyond the critical frequencies of most hydrodynamic loads, thereby reducing dynamic amplification effects.

The inside and outside column supports are pipe members. The inside column is an 8-inch diameter pipe reinforced by two, 160° segments of 10-inch diameter pipe. The outside column is a $10\frac{3}{4}$-inch outside diameter pipe with a $2\frac{1}{4}$-inch wall thickness. The column base plate assemblies consist of base plates, a pin, and anchor bolts (Figure 3.8-20).

The connection of the column supports to the suppression chamber shell is a column stub assembly consisting of a $1\frac{1}{2}$-inch thick column web plate, 1-inch thick stiffener plates, and either a $1\frac{1}{2}$-inch thick wing plate at the outside column or a $1\frac{1}{4}$-inch thick flange plate at the inside column (Figure 3.8-19).

The column connection web plates and saddle support web plates are connected with fillet welds and partial penetration welds.

Each saddle support consists of a 1¼-inch thick web plate, a 1¼-inch thick lower flange plate and saddle base plate assemblies (Figures 3.8-19 and 3.8-20). The saddle base plate assemblies consist of a 2⅞-inch thick base plate, a ½-inch thick lubrite plate, and a 1-inch thick bearing plate. This assembly allows for radial growth due to thermal loads as do the column base plate assemblies. The saddle is reinforced with 4¼-inch thick stiffener plates to ensure that buckling does not occur during peak loading conditions.

The anchorage of the suppression chamber saddle to the basemat consists of eight, 1¾-inch diameter, epoxy-grouted anchor bolts provided at each saddle base plate location. Four, 1½-inch, epoxy-grouted anchor bolts and two embedded in the original basemat pour are provided at each outside column base plate location, and two epoxy-grouted anchor bolts and two embedded in the original basemat pour are provided at each inside column location. The saddle anchor bolts are anchored through a 3 13/16-inch long slotted hole in the base plate to allow for thermal growth. A total of 26 anchor bolts at each miter joint provides the principal mechanism for transfer of uplift loads on the suppression chamber to the basemat.

A sway rod assembly at the outside columns provides lateral support for the suppression chamber (Figure 3.8-23). This seismic sway rod consists of 3½-inch diameter sway rods and 3¾-inch diameter turnbuckles to provide restraint for movement along the torus centerline resulting from lateral loads acting on the suppression chamber. The sway rods are joined to the 1½-inch thick wing plate at the top of the column by 4-inch diameter pins. The lower ends of the sway rods are joined to a 2-inch thick seismic tie plate at the column base (Figure 3.8-23).

Each unit has five vent bays with T-quenchers. The ramsheads of the T-quenchers are located near midbay, with the associated quencher arms oriented horizontally parallel to the longitudinal axis of the vent bay. The quencher arms are supported by a horizontal pipe beam which spans the miter joint ring girders.

The suppression chamber provides support for many other containment-related structures, such as the vent system, and the catwalk. Loads acting on the suppression chamber cause motions at the points where these structures attach to the suppression chamber. Loads acting on these structures also cause reaction loads on the suppression chamber. These containment interaction effects were evaluated in the analysis of the suppression chamber.

3.8.2.3.2 Applicable Codes, Standards, and Specifications

The primary containment, including the suppression chamber described herein, was originally designed, erected, pressure-tested, and N-stamped in accordance with the ASME Code Section III, 1965 Edition with addenda up to and including Summer 1965.

For the Mark I Program reevaluation, the acceptance criteria generally follow the ASME Code Section III, 1977 Edition with addenda up to and including Summer 1977 for Class MC components and component supports. Further detail regarding structural acceptance criteria may be found in Section 3.8.2.3.5.

Repairs/Replacements of primary containment components (class MC) are performed in accordance with ASME Code Section XI, Subsection IWE, 2001 Edition with 2003 Addenda.

3.8.2.3.3 Loads and Load Combinations

The loads acting on the suppression chamber are categorized as follows:

A. Dead weight loads,

B. Seismic loads,

C. Pressure and temperature loads,

D. Column preset loads,

E. Pool swell loads,

F. Condensation oscillation loads,

G. Chugging loads,

H. SRV discharge loads, and

I. Containment interaction loads.

Design pressure loads applicable to the suppression chamber are described in Section 3.8.2.1.3.

Not all of the loads defined in NUREG-0661[3] have been evaluated in detail, because some are enveloped by others or have a negligible effect on the suppression chamber. Only those loads which maximize the suppression chamber response and lead to controlling stresses have been fully evaluated and are described in the PUAR.[1]

Not all of the possible suppression chamber load combinations were evaluated, since many were enveloped by others and do not lead to controlling suppression chamber stresses. The enveloping load combinations were determined by examining the possible suppression chamber load combinations and comparing the respective load cases and allowable stresses as described more fully in the PUAR.[1]

3.8.2.3.4 Design and Analysis Procedures

The repetitive nature of the suppression chamber geometry is such that the suppression chamber can be divided into 16 identical segments, which extend from midbay of the vent line bay to midbay of the nonvent line bay (Figure 6.2-3). The suppression chamber can be further divided into 32 identical segments extending from the miter joint to midbay, provided the offset ring girder and vertical supports are assumed to lie in the plane of the miter joint. The effects of the ring girder and vertical supports offset have been evaluated and found to have a negligible effect on the suppression chamber response. The analysis of the suppression chamber, therefore, was performed for a typical 1/32 segment.

A finite element model of a 1/32 segment of the suppression chamber was used to obtain the suppression chamber response to all loads except those on submerged structures. This analytical model includes the suppression chamber shell, the ring girder modeled with beam elements, the column connections and associated column members, the saddle support and associated base plates, and miscellaneous stiffener plates.

The suppression chamber shell has a circumferential node spacing of 8° at midbay, with additional mesh refinement near discontinuities to facilitate examination of local stresses. Additional refinement is also included in modeling of the column connections and saddle support at locations where higher local stresses occur. The stiffness and mass properties used in the model are based on the nominal dimensions and densities of the material used to construct the suppression chamber. Small displacement linear-elastic behavior is assumed throughout.

A second finite element model was developed to obtain detailed ring girder responses to suppression chamber shell hydrodynamic loads and ring girder-torus shell interaction responses to loads on submerged structures. This model consisted of a detailed plate model of the ring girder and ring girder stiffeners, a partial 1/32 segment torus shell model on each side of the miter joint, the column connections and associated column members, the saddle support with associated flanges, and the stiffener plates. The column, column connection, and saddle support were positioned 4 inches from the miter joint in this analytical model to accurately represent the as-built torus support system.

The model reflects the modified ring girders, reinforced to withstand Mark I loads. These modifications are lateral reinforcement stiffeners to prevent ring girder bending due to out-of-plane loads. Upon installation of the final Mark I related modifications, both units at Dresden have five ring girder stiffeners in the SRVDL bays (Figure 6.2-5); however, they differ in the number of ring girder stiffeners in the non-SRVDL bays. Unit 2 has zero; Unit 3 has two. Two analytical models were generated to address the submerged structure loads, one each for the SRVDL and non-SRVDL bays. These are the five-stiffener model and the zero-stiffener model. The zero stiffener ring girder configuration was conservatively chosen for analysis of the non-SRVDL bay loads.

For each of the hydrodynamic torus shell loads, a displacement set was statically applied to the ring girder-torus shell intersection on the ring girder model, along with appropriate dynamic amplification factors. This displacement set was selected from the response time-history at the time of maximum strain energy. These loads thus applied determine the state of stress in the ring girder due to hydrodynamic torus shell loads.

For each of the submerged structure loads, a set of forces was applied to the ring girder below the pool surface in the out-of-plane direction. A dynamic load factor was developed for each load, depending upon the natural frequency of the ring girder and that of the load itself. With the application of this factor, the state of stress was determined in the ring girder, the ring girder stiffener plates, and the local torus shell due to the submerged structure loads.

When computing the response of the suppression chamber to dynamic loading, the fluid-structure interaction effects of the suppression chamber shell and contained fluid (water) were considered. This was accomplished through use of a finite element model of the fluid. The analytical fluid model was used to develop a coupled mass matrix, which was added to the submerged nodes of the suppression chamber analytical model to represent the fluid. A water volume corresponding to a water level 3½ inches below the suppression chamber horizontal centerline was used in this calculation. This was the average water volume expected during normal operating conditions.

A frequency analysis was performed using the suppression chamber analytical model from which all structural modes in the range of 0 to 50 Hz were extracted.

Using the analytical model of the suppression chamber, a dynamic analysis was performed for each of the hydrodynamic torus shell load cases specified in Section 3.8.2.3.3. The analysis consisted of either a transient or harmonic analysis, depending on the characteristics of the torus shell load being considered. The modal superposition technique with 2% of critical damping was utilized in both transient and harmonic analyses.

The remaining suppression chamber load cases specified in Section 3.8.2.3.3 involved either static or dynamic loads which were evaluated using an equivalent static approach. For the latter, conservative dynamic amplification factors were developed and applied to the maximum spatial distributions of the individual dynamic loadings.

In addition to vertical loads, a few of the governing loads acting on the suppression chamber result in net lateral loads. These lateral loads are transferred to the reactor building basemat by the torus seismic restraints.

The general methodology used to evaluate the effects of lateral loads consists of establishing an upper bound value of the lateral load for each applicable load case. The results for each load case were then grouped in accordance with the controlling load combinations and the maximum total lateral load acting on the suppression chamber was determined.

The direction of each lateral load acting on the suppression chamber is taken as the azimuth causing the maximum tensile stress in the seismic sway rods. Depending on the load, the direction of the azimuth is aligned either with a miter joint or with the midbay of a 1/16 sector of the suppression chamber. A 360° beam model of the torus, supports, and seismic sway rods was used in this determination of the distribution of the lateral loads. Once the seismic restraint loads were known, these values were compared with the allowable seismic restraint loads.

Tensile loads in the seismic sway rods result in concentrated forces acting on the suppression chamber. These forces act in the direction of the sway rods at the point of attachment to the outside column wing plates. The effect of these forces on the suppression chamber shell are evaluated using the analytical model described previously as the ring girder model. The resulting shell stresses are then combined with the other loads contained in the controlling load combination being evaluated, and the shell stresses in the vicinity of the seismic restraints are determined. The resulting shell stresses were then combined with the other loads contained in the controlling load combination being evaluated, and the shell stresses in the vicinity of the seismic restraints were determined.

3.8.2.3.5 Structural Evaluation

The NUREG-0661[3] acceptance criteria on which the suppression chamber analyses are based follow the ASME Code, Section III, 1977 Edition, including the Summer 1977 Addenda for Class MC components and component supports. The corresponding service level assignments, jurisdictional boundaries, allowable stresses, and fatigue requirements are consistent with those contained in the applicable subsections of the ASME Code and the PUAAG.[5]

The items examined in the analysis of the suppression chamber include the suppression chamber shell, the ring girder, and the suppression chamber horizontal and vertical support systems.

The suppression chamber shell and ring girder were evaluated in accordance with the requirements for Class MC components contained in Subsection NE of the ASME Code. Fillet welds and partial penetration welds in which one or both of the joined parts includes the suppression chamber shell and the ring girder were also evaluated in accordance with the requirements for Class MC component attachment welds contained in Subsection NE of the ASME Code.

The allowable stresses for each suppression chamber component and vertical support system component were determined at 165°F. The allowable stresses for the vertical support system base plate assemblies were determined at 100°F. Table 3.8-9 shows the resulting allowable stresses for the load combinations with ASME Code Service Level B, C, and D limits.

Table 3.8-10 summarizes the maximum stresses and associated design margins for the major suppression chamber components and welds for the controlling load combinations.

The components of the suppression chamber, which are specifically designed for the loads and load combinations used in this evaluation, exhibit the margins of safety inherent in the original design of the primary containment after completion of the modifications described in Section 6.2. The intent of the NUREG-0661[3] requirements is therefore considered to be met.

3.8.2.3.6 Testing and Inservice Inspection Requirements

Pressure and leakage rate testing of the containment system is addressed in Section 6.2.6.

Primary containment repairs, replacements, and examinations are performed in accordance with ASME Section XI, Subsection IWE, 2001 Edition with 2003 Addenda.

3.8.3 Internal Structures of Steel Containment

Class I structures located within the primary containment include the reactor concrete pedestal and the concrete reactor shield wall. Structural evaluation for the reactor pedestal and reactor shield wall are addressed in Section 3.9.3.

3.8.4 Other Class I Structures

The major structures covered in this section are the reactor building, turbine building (including control room), and the 310-foot concrete chimney. The reactor and turbine buildings are constructed as one integral structure and were analyzed as one composite model as explained in Section 3.7.2.2. The integral structure will be referred to as the reactor-turbine building in the discussions which follow. The turbine building portion of the structural complex is a Class II structure as explained in Section 3.8.5.

The plant structures and equipment are divided into two categories as related to safety. These categories are Class I and Class II, as defined in Section 3.2, and repeated below:

A. Class I - those structures and equipment of which a failure thereof could cause significant release of radioactivity or are vital to a safe plant shutdown.

B. Class II - all other structures and equipment which are utilized in the station operation but are not essential to a safe shutdown.

Implementation of these definitions has resulted in specific structures, systems, and components being classified as Class I. These items are listed in Section 3.2.

For all Class I structures, general discussion of the applicable codes, loads and load combinations, structural acceptance criteria, materials, and inspection requirements is provided in the following subsection under reactor turbine building. These general requirements are not repeated in the subsequent subsections unless specific exceptions are described.

3.8.4.1 Reactor-Turbine Building

3.8.4.1.1 Description of Structure

The description of the reactor-turbine building is provided in Section 3.7.2.2.

3.8.4.1.2 Applicable Codes, Standards, and Specifications

The design code which was used to govern the construction documents was the "Uniform Building Code," 1964 Edition (UBC). Floor live loads are as specified in documents transmitted to the architect-engineer by GE and the UBC. Roof live loads are from the UBC. All structures are designed to withstand the maximum potential loadings resulting from a wind velocity of 110 mph. This is considerably in excess of UBC requirements.

3.8.4.1.3 Loads and Load Combinations

General requirements for the design of all structures and equipment include provisions for resisting the dead loads, live loads, and wind or seismic loads with impact loads considered part of the live load. Selection of materials to resist these loads is based on standard practice in the power plant field. Their use is governed by the building codes valid at the site of construction and the experience and knowledge of the designers and builders.

The loads of concern include the following:

D = dead load of structure and equipment plus any other permanent loads contributing stress, such as soil or hydrostatic loads or operating pressures, and live loads expected to be present when the plant is operating

P = pressure due to LOCA

R = jet force or pressure on structure due to rupture of any one pipe

H = force on structure due to thermal expansion of pipes under operating conditions

T = thermal loads on containment due to LOCA

E = OBE load (0.10 g horizontal ground acceleration, 0.067 g vertical acceleration)

$E^{/}$ = SSE load (0.20 g horizontal ground acceleration, 0.133 g vertical acceleration)

3.8.4.1.4 Design and Analysis Procedures

The criteria for Class I structures and equipment with respect to stress levels and load combinations for the postulated events are noted below:

D + R + E Normal allowable code stresses (AISC for structural steel, ACI for reinforced concrete). The customary increase in design stresses, when earthquake loads are considered, is not permitted.

D + R + $E^{/}$ Stresses are limited to the minimum yield point as a general case. However in a few cases, stresses may exceed yield point. In this case an analysis, using the Limit-Design approach, is made to determine the energy absorption capacity which should be such that it exceeds the energy input. This method has been discussed in the NRC publication TID-7024, "Nuclear Reactor and Earthquakes," Section 5.7. The resulting distortion is limited to assure no loss of function and adequate factor of safety against collapse.

The maximum allowable stresses used for various loading conditions are given for Class I structures in Table 3.8-11.

3.8.4.1.5 Structural Evaluation

Absolute OBE seismic acceleration curves were established for the reactor-turbine building which give the envelope of the maximum absolute acceleration with respect to height. These curves were used for the seismic design of equipment elements rigidly attached to the building and are shown by Figures 3.8-33 and 3.8-34. Since these curves were of absolute accelerations, the shear, moment, and displacement curves were used in the design of the building. The maximum OBE envelopes of building design shears, moments and displacements are shown graphically in Figures 3.8-35 to 3.8-40.

In Figure 3.8-33, it is noted that in the turbine building an acceleration as high as 2.4 g is calculated to occur at about elevation 580'-0". The maximum shears and moments produced by the acceleration (2.4 g) were distributed among the turbine building superstructure columns. A further seismic study with 59% of the crane load acting on one turbine building frame indicates a maximum acceleration of 0.80 g acting at the elevation of the crane. The period of the system is 0.54 seconds. Note that the turbine building is a Class II structure and that the degree of analysis which was performed exceeds the requirements for a Class II structure.

All Class I structures and systems have been investigated for SSE loading, and only in insignificant, isolated locations in the reactor building's superstructure were the yield or ultimate stresses slightly exceeded. Because of the redistribution of forces and the additional structural systems, such as the diaphragm action of the roof truss, the design is considered adequate with the slight overstress in isolated locations, and an energy absorption analysis is not warranted.

3.8.4.2 Control Room

Acceleration, shear, and moment curves calculated as a result of the OBE analysis of the control room are included as Figures 3.8-41, 3.8-42, and 3.8-43.

Although the analysis of the reactor-turbine building was a coupled time history analysis, the control room was independently modeled and dynamically analyzed. A comparison of accelerations for the concrete portion of the turbine building structure versus the control room portion alone is given in Table 3.8-12.

The difference in accelerations of the superstructure is due to the modeling assumptions and method of analysis of the superstructure and distribution of the crane loads.

A review of potential hazards which could jeopardize control room integrity was conducted. It was concluded that the worst case condition which could occur would be if the turbine room crane were to fall from its rails. This potential problem was analyzed in detail and it was concluded this situation would not occur. A summary of the analysis follows.

The cranes consist basically of a bridge section, the bridge trucks, and the bridge wheels. The bridge section is of the box section girder type with an internal diaphragm stiffener. The bridge trucks are also constructed as girders and are connected rigidly to the bridge girder section. The bridge wheels are of the double flanged type, the flanges being large enough to completely envelope the crane rail head, i.e., the flange tips extend below the bottom of the rail head. The wheels are press fitted to the axle, which is mounted on rubber bearings encased in bearing capsules on both sides of the axle. The capsules are rigidly attached to the bridge truck. There are four wheels at each end of the bridge on each crane. The entire assembly behaves as a rigid unit such that the crane will not weave or twist during normal operations. Due to vertical earthquake excitation, the entire bridge structure will not experience a vertical acceleration which is in excess of 1.0 g; hence, the wheel tread will never lose contact with the supporting rail. Also, because all of the wheels are of the double flange type and because of the rigidity of the bridge unit, it would not be possible for the bridge assembly to become derailed due to lateral seismic excitation.

3.8.4.3 Concrete Chimney

Design and construction of the chimney column was in accordance with applicable requirements of ACI "Standard Specification for the Design and Construction of Reinforced Concrete Stacks" (ACI-505) with the following exceptions and with those described in Section 3.3.1.1:

A. Earthquake loading

1. Total lateral static coefficients: 10% of total gravity modes.

2. Distribution of lateral forces.

    a. Lateral forces have been distributed along the chimney column in accordance with the following UBC formula:

$$F_x = \frac{VW_x h_x}{Ewh}$$

    b. Cantilever moments were determined using this load distribution.

3. Allowable unit stresses: Allowable stresses for the combined effects of wind and dead load, per Sections 500 and 501 of ACI-505, were satisfied.

The chimney structure was designed to resist the seismic shears and moments presented without any increase in stress for short term loadings. In addition, analysis of the structure indicates that it could resist twice the OBE seismic shears and moments presented without hindering the ability of the plant to safely shut down and without collapse or failure which could cause damage to other structures or components whose integrity is necessary to assure a safe shutdown of the plant.

The results of the OBE seismic analysis in the form of design shears, design moments and relative displacement envelopes are shown in Figures 3.8-44 through 3.8-46.

The chimney working stress design was controlled by the OBE from an elevation beginning at 160'-0" above grade to the top. Critical stresses occur at an elevation of 208'-0" above grade. At this elevation the steel stress is 10,550 psi and the concrete stress is 440 psi.

The chimney was investigated for the critical stress point for SSE. This critical point is at an elevation of 208'-0" above grade. At this elevation the steel stress is 21,000 psi and the concrete stress is 660 psi.

Refer to Section 3.3.1.1 for additional information pertaining to the analysis of the chimney, including wind loading.

3.8.4.4 Masonry Walls

As a result of IE Bulletin 80-11, a reevaluation of 96 masonry walls which are in proximity to safety-related piping or support safety-related piping has been performed. This analysis is described in more detail in Section 3.7.3.4.1. The analysis considered the function of the wall, the construction materials used, and the construction techniques used in the walls. The walls were analyzed for dead load, live load, attachment loads, wind load, tornado load, earthquake load, thermal loads, and high energy line break loads. The calculated stresses were compared with building code allowable stresses per ACI 531-79 to determine adequacy of the walls. Walls which were found to be inadequate by this analysis were modified accordingly.

3.8.4.5 Embedded Plates

On February 2, 1987, while performing piping system hanger and support inspections in accordance with the In-Service Inspection (ISI) Program, CECo personnel discovered that the embedment plate for support M1150D-62 was pulled away from the ceiling approximately 1/8 to 1/4 inch along one surface.

Investigation into the degraded embedment plate revealed a discrepancy between the shop drawing used for its fabrication and the design specifications. Although the design specified 9-inch hold down strap spacings, the shop drawing indicated 18-inch hold down strap spacings. The 18-inch spacings were confirmed on the degraded embedment plates by performing ultrasonic (UT) tests. Further UT tests performed on other embedment plates also revealed the existence of some 24-inch hold down strap spacings.

As result of the discrepancy between the design and the installed strap spacing, a further investigation of embedment plates on all related piping installations at Dresden Units 2 and 3 was initiated.

CECo conducted a two-part review program. The first part used interim acceptance limits - with an upper limit to ascertain that plant operation under as-found conditions does not present a safety hazard to the public. The second part consisted of a long term program to conduct modifications as needed, such that all embedment plates were confirmed to meet Final Safety Analysis Report (FSAR) stress allowables.

The embedded plate program resulted in modifications to only three plates out of approximately 1100 at Dresden to satisfy FSAR allowable stress commitments. In spite of discrepancies between design and fabrication of the embedment plates, anchor spacing for all embedment plates met operability criteria stresses.

3.8.4.6 Concrete Expansion Anchors IE Bulletin 79-02 Program

A mixture of wedge- and self-drilling-type concrete expansion anchors have been used in safety-related areas. The minimum embedment depth for wedge type expansion anchors is 4½ anchor diameters. Self-drilling anchors were predominately used prior to 1977. All concrete expansion anchors were specified to be installed in accordance with manufacturer's recommendations.

Commonwealth Edison Company inspected wedge- and self-drilling-type expansion anchors supporting safety-related piping in which the calculated factor of safety (ultimate anchor capacity divided by the calculated applied load) is less than or equal to 10. This was done to assure conformance to manufacturer's installation recommendations. Wedge-type expansion anchors were inspected to verify the following items:

A. Minimum test torque level,

B. Minimum embedment depth, and

C. Expansion anchor size.

Wedge-type expansion anchors which did not meet the required test torque value were retorqued to the installation torque value and reinspected within 7 days to assure that relaxation did not exceed the required torque level. Wedge-type expansion anchors which did not have the correct embedment length or size were reanalyzed and, when inadequate to support the design loads, were replaced, or the expansion anchored plate assembly modified accordingly to carry the design loads.

Manufacturers of self-drilling concrete expansion anchors typically have not specified initial installation torque values. The torquing of a self-drilling anchor does not seat the anchor in the concrete hole, and thereby, minimize anchor displacement (as in the case of wedge-type anchors). Commonwealth Edison Company, however, performed a test program for self-drilling type expansion anchors under the direction of an independent testing laboratory to determine appropriate test torque levels to assure that the preload in the self-drilling expansion anchors is greater than or equal to the design loads. Self-drilling expansion anchor assemblies supporting safety-related piping were inspected by applying the test torque to the individual anchors and verifying the correct size. The self-drilling expansion anchors were inspected subsequent to the application of the test torque to assure that the shell of the self-drilling expansion anchor was not in contact with the back of the expansion anchor baseplate. Self-drilling expansion anchors which were in contact with the back of the expansion anchor baseplate were either replaced with a wedge-type anchor, or the expansion anchored plate assembly was modified to support the design loads.

Future expansion anchor installations will consist of wedge-type anchors only, with an embedment length equal to eight anchor diameters. These anchors will be installed in accordance with approved QA/QC procedures, and the design load for these anchors will be less than the specified anchor preload.

3.8.4.7 Reactor Building Superstructure

The reactor building superstructure is designed for the following loading conditions and acceptance criteria (Reference 6):

a) Normal load condition (includes full crane lifted load)
Acceptance criteria: Normal AISC allowables.

b) Wind load condition (includes full crane lifted load)
Acceptance criteria: UFSAR Table 3.8-11; allowable stresses for loading condition 2.

c) OBE seismic condition (includes full crane lifted load)
Acceptance criteria: UFSAR Table 3.8-11; allowable stresses for loading condition 1.

d) SSE seismic condition (includes full crane lifted load)
Acceptance criteria: UFSAR Table 3.8-11; allowable stresses for loading condition 3.

The design calculation DRE 98-0020 shows that the reactor building superstructure is acceptable for all of the above load combinations, including the load combinations of OBE and SSE with full crane loads.

The design calculation considered the travel paths of the reactor vessel head and the spent fuel cask that bound the paths described in response to NUREG-0612. For the concrete shield plugs, while the calculation considered the travel path similar to that described in response to NUREG-0612, it limited the distance the concrete shield plugs can move relative to the outer columns (References 7 & 8).

The information regarding the travel paths of the heavy loads (reactor vessel head, spent fuel cask and concrete shield plugs) is being incorporated into the corporate procedures related to heavy load movement (References 9 & 10).

3.8.5 Non-Class I Structures

Class II structures supporting Class I structures, systems and components were designed to Class II requirements and have been investigated to assure that the integrity of the Class I items is not compromised. Class I structures, systems and components located in Class II structures include the control room, standby gas treatment system, and the standby electrical power systems comprising of the station batteries, diesel generators, essential busses, and other electrical gear for power to critical equipment.

The following structures and systems were designed for Class II rather than Class I because none of them are required for safe shutdown of the plant under conditions of the DBA: the crib house, radioactive waste building and waste disposal system, new storage building for the old steam dryers, condensate storage tanks and pumps, reactor building crane, auxiliary power buses, shutdown cooling system, the standby coolant supply system, service water system, fire protection system, and air compressors and receivers.

The containment cooling service water pumps and the emergency diesel generator cooling water pumps are located in Class II structures, but have been afforded Class I protection. The containment cooling service water pumps are located in the turbine building below grade on a reinforced concrete floor above the condensate and condensate booster pumps. The grade floor slab above these pumps protects them from debris and missiles during tornado-type conditions and the floors and surrounding structure in this area have been calculated to be earthquake resistant. The emergency diesel generator cooling water pumps are located at elevation 490'-8" in the crib house. This is the same floor that the circulating water pumps are located on and is below the reinforced concrete slab at grade. The concrete structure of the crib house would not be affected by tornado or earthquakes.

The auxiliary power buses are not required for a safe shutdown of the plant. The diesel generators supply power to the emergency buses which are Class I. The diesel generators and the emergency buses are both totally redundant.

Equipment which requires air from the air compressors and receivers are designed for fail-safe operation should a loss of air occur. Therefore, the air compressors and receivers are not designed to Class I.

3.8.6 References

1. "Dresden Nuclear Power Station Units 2 and 3 Plant Unique Analysis Report," Revision 0, May 1983.

2. Letter from J.A. Zwolinski (NRC) to D.L. Ferrar (CECo), September 18, 1985, "Mark I Containment Long-Term Program."

3. NUREG-0661, July 1980, "Safety Evaluation Report Mark I Containment Long-Term Program," U.S. Nuclear Regulatory Commission.

4. "Mark I Containment Program Plant Unique Load Definition, Dresden Station Units 2 and 3, General Electric Company, NEDO-24566, Revision 2, April 1982.

5. "Mark I Program Structural Acceptance Criteria Plant Unique Analysis Applications Guide, Task Number 3.13," Mark I Owners Group, General Electric Company, NEDO-24583, Revision 1, July 1979.

6. Calculation DRE 98-0020 (revisions 0 & 1) “Evaluation of Reactor Building Superstructure”

7. Letter from ComEd to NRC dated June 22, 1981 regarding “Control of Heavy Loads at Nuclear Power Plants”

8. NRC letter to ComEd dated July 11, 1983 “NUREG-0612; Control of Heavy Loads at Nuclear Plants”

9. Letter No. RS-02-082 from K. R. Jury (Exelon) to U. S. NRC, “Response to Request for Additional Information Regarding Heavy Loads Handling,” dated April 12, 2002

10. Letter No.RS-02-125 from Patrick R. Simpson (Exelon) to U. S. NRC, “Supplement to Response to Request for Additional Information Regarding Heavy Loads Handling,” dated July 8, 2002.

Table 3.8-1

DRYWELL MAJOR PENETRATION CLASSIFICATION
UNIT 2

Penetration	Quantity	Service	Type[1]	Size (in.)
X-105A, B, C, D	4	Main steam	1	20
X-106	1	Main steam drain	1	2
X-107A, B	2	Feedwater	1	18
X-108A	1	Isolation condenser steam supply	1	14
X-109B	1	Isolation condenser return	1A	12
X-111A, B	2	Shutdown cooling supply	1A	16
X-113	1	Cleanup system supply	1A	8
X-115	1	HPCI steam supply	1	10
X-116A	1	LPCI pump discharge	1A	16
X-116B	1	LPCI pump discharge	1A	16
X-117	1	Drywell floor drain sump discharge	2	3
X-118	1	Drywell equipment drain sump discharge	2	3
X-119	1	Demineralized water supply	2	3
X-120	1	Service air supply	2	1
X-121	1	Instrument air supply	2	1
X-123	1	Reactor building closed cooling water supply	1A	6
X-124	1	Reactor building closed cooling water return	1A	6
X-130	1	Standby liquid control	-	1½
X-144[2]	1	CRD system return[2]	1[2]A	4
X-145, X-150	2	Containment spray	-	10
X-147	1	Reactor head cooling	1A	2½
X-149A, B	2	Core spray	1A	10

Notes:

1. Penetration types are illustrated in Figures 3.8-8 (Type 1), 3.8-9 A (Type 1A), 3.8-9B (Type 1B), and 3.8-10 (Type 2).
2. The Unit 2 CRD return line was removed inside the drywell in 1993. The penetration and the CRD return line were both capped on the inboard side of the penetration.

Table 3.8-2

DRYWELL MAJOR PENETRATION CLASSIFICATION
UNIT 3

Penetration	Quantity	Service	Type(1)	Size (in.)
X-105A	1	Main steam	1	20
X-105B, C, D	3	Primary steam	1	20
X-106	1	Main steam drain	1	2
X-107A	1	Feedwater	1	18
107B	1	Feedwater	1A	18
X-108A	1	Isolation condenser steam supply	1	14
X-109A	1	Isolation condenser return	1A	12
X-111A	1	Shutdown cooling supply	1A	16
X-111B	1	Shutdown cooling supply	1A	16
X-113	1	Cleanup system supply	1	8
X-113B	1	Shutdown cooling supply	1	8
X-128	1	HPCI steam supply	1	10
X-116A, B	2	LPCI pump discharge	1A	16
X-117	1	Drywell floor drain sump discharge	2	3
X-118	1	Drywell equipment drain sump discharge	2	3
X-119	1	Demineralized water supply	2	3
X-120	1	Service air supply	2	1
X-121	1	Instrument air supply	-	1
X-123	1	Reactor building closed cooling water supply	1A	6
X-124	1	Reactor building closed cooling water return	1A	6
X-138	1	Standby liquid control	-	1½
X-109B(2)	1	CRD system return(2)	1(2)A	4
X-145, X-150	2	Containment spray	-	10
X-147	1	Reactor head cooling	1A	2½
X-149A	1	Core spray	1A	10
X-149B	1	Core spray	1A	10

Notes:

1. Penetration types are illustrated in Figures 3.8-8 (Type 1), 3.8-9 (Type 1A), and 3.8-10 (Type 2).
2. The Unit 3 CRD return line was removed inside the drywell during the 1985-86 Unit 3 Recirculation Pipe Replacement Outage. The drywell penetration was capped on the inside, and the line itself was capped on the outboard side of the penetration.

Table 3.8-3

ELECTRICAL PENETRATIONS ENVIRONMENTAL DESIGN CONDITIONS
(Historical Data)*

Normal Operating Environment — Each penetration assembly shall be capable of continuous operation at the environmental conditions listed below:

Parameter	Inside Primary Containment	Outside Primary Containment
Temperature	150°F	125°F
Pressure	-2 to +2 psig	0 psig
Relative Humidity (RH)	20% to 100%	20% to 100%
Limits of RH vs. Lifetime	50% RH < 2% time 70% RH < 1% time 90% RH < 0.5% time	50% RH < 2% time 70% RH < 1% time 90% RH < 0.5% time
Radiation Dose (without shielding)	10 R/hr	Less than 0.1 R/hr

Maximum Emergency Environment — Each penetration assembly shall be capable of maintaining containment integrity for not less than 2 hours when subjected to the environmental conditions listed below. The canister leakage rate limits are established by the Primary Containment Leakage Rate Testing Program.

Parameter	Inside Primary Containment
Temperature	320°F
Pressure	125 psig
Relative Humidity	100% RH

Maximum Long-Term Emergency Environment — Each penetration assembly shall be capable of maintaining containment integrity for at least 10 days when subjected to the environmental conditions listed below. The canister leakage rate limits are established by the Primary Containment Leakage Rate Testing Program.

Parameter	Inside Containment
Temperature	281°F
Pressure	62 psig
Relative Humidity	100% RH

* NOTE: See Tables 3.11-1, 3.11-2, 3.11-3 for Environmental Parameters

Table 3.8-4

PRIMARY INSTRUMENT LINE PENETRATIONS

Penetration No. Unit 2	Penetration No. Unit 3	Function	Type Isolation(1)
X-131A,C	X-115A, X-127D	Drywell pressure trip for LPCI pressure initiation	1
X-131B,D	X-115B, X-127E	Drywell pressure trip for containment spray pressure permissive	1
	X-115C-F, X-127A-C,F	Spares	
X-115B	X-135B,C	A Main steam line flow, pressure (C penetration), and flow restrictor dP indication.	1
X-115B	X-135E,F	B Main steam line flow, pressure (F penetration), and flow restrictor dP indication.	1
X-115B	X-129B,C	C Main steam line flow, pressure (C penetration), and flow restrictor dP indication.	1
X-115B	X-129E,F	D Main steam line flow, pressure (F penetration), and flow restrictor dP indication.	1
	X-129A,D X-135A,D	Spares	
X-132B	X-130A	Pressure indication for A recirculation pump cavity 2 seal leak	1
X-132A	X-130D	Pressure indication for A recirculation pump cavity 1 seal leak	1
X-132D	X-131A	Pressure indication for B recirculation pump cavity 2 seal leak	1
X-132C	X-131D	Pressure indication for B recirculation pump cavity 1 seal leak	1

Table 3.8-4 (Continued)

PRIMARY INSTRUMENT LINE PENETRATIONS

Penetration No. Unit 2	Unit 3	Function	Type Isolation(1)
X-134A	X-130E	A recirculation pump head (LPCI)	1
X-134B	X-130F	A recirculation pump discharge pressure	1
X-134C	X-131E	B recirculation pump head (LPCI)	1
X-134D	X-131F	B recirculation pump discharge pressure	1
X-133A,B	X-130B,C	A recirculation flow APRM converter	1
X-133C,D	X-131B,C	B recirculation flow APRM converter	1
X-141A	X-132E,F	dP for isolation condenser supply flow	1
X-141A	X-132A,D	dP for isolation condenser return flow	
	X132B,C	Spares	
	X-133A	RPV flange leak detection	1
X-129A-D X-128A,B X-129E,F	X-133B,C,E,F X-134B,C,E,F	RPV level and pressure including core plate dP	1
X-127A	X-133D	Wide range level transmitter	1
	X-134A,D	Spares	
X-136	X-136	TIP flux monitor tubes	2
X-136E	X-136F	Purge check valves	2
X-136J	X-136C	TIP ball valve A	2
X-136F	X-136B	TIP ball valve B	2
X-136E	X-136D	TIP ball valve C	2
X-136H	X-136F	TIP ball valve D	2
X-136G	X-136E	TIP ball valve E	2
X-146A-D, X-208E-H	X-141A-D X-151A-D	Recirculation riser dP	1

Table 3.8-4 (Continued)

PRIMARY INSTRUMENT LINE PENETRATIONS

Penetration No. Unit 2	Penetration No. Unit 3	Function	Type Isolation[1]
	-141E,F X-151F	Spares	
X-142A,B	X-142A-D	Jet pump flow	1
X-131C	X-151E	Jet pump instrument	1
X-209[2]	X-209[2]	Reactor vessel water level (A-loop)	1
X-108B[2]	X-108B[2]	Reactor vessel water level (B-loop)	1

Notes:

1. Type 1 isolation includes an excess flow check valve and stop valve outside of containment.

2. The Unit 2 and Unit 3 A-Loop and B-Loop interface with the RVWLIS Backfill lines upstream of the stop valves. The RVWLIS Backfill lines provide containment isolation with two simple check valves.

Table 3.8-5

ALLOWABLE STRESSES FOR VENT SYSTEM COMPONENTS AND COMPONENTS SUPPORTS

Item		Material[1]	Material Properties[2]	Stress Type	Allowable Stress (ksi)	
					Service Level B	Service Level C
Components	Drywell Shell	SA-516 Grade 70	S_{mc} = 19.30 S_{ml} = 22.61 S_y = 33.87	Local Primary Membrane	28.95	50.81
				Primary + Secondary[3] Stress Range	67.83	N/A
	Vent Line	SA-516 Grade 70	S_{mc} = 19.30 S_{ml} = 22.61 S_y = 33.87	Primary Membrane	19.30	33.87
				Local Primary Membrane	28.95	50.81
				Primary + Secondary[3] Stress Range	67.83	N/A
	Vent Line/ Vent Header Spherical Junction	SA-516 Grade 70	S_{mc} = 19.30 S_{ml} = 22.61 S_y = 33.87	Primary Membrane	19.30	33.87
				Local Primary Membrane	28.95	50.81
				Primary + Secondary[3] Stress Range	67.83	N/A
	Vent Header	SA-516 Grade 70	S_{mc} = 19.30 S_{ml} = 22.61 S_y = 33.87	Primary Membrane	19.30	33.87
				Local Primary Membrane	28.95	50.81
				Primary + Secondary[3] Stress Range	67.83	N/A

Table 3.8-5 (Continued)

ALLOWABLE STRESSES FOR VENT SYSTEM COMPONENTS AND COMPONENTS SUPPORTS

Item		Material[1]	Material Properties[2]	Stress Type	Allowable Stress (ksi)	
					Service Level B	Service Level C
Components	Downcomer	SA-516 Grade 70	$S_{mc} = 19.30$ $S_{ml} = 22.61$ $S_y = 33.87$	Primary Membrane	19.30	33.87
				Local Primary Membrane	28.95	50.81
				Primary + Secondary[3] Stress Range	67.83	N/A
	Support Collar Plate	SA-516 Grade 70	$S_{mc} = 19.30$ $S_{ml} = 22.61$ $S_y = 33.87$	Primary Membrane	19.30	33.87
				Local Primary Membrane	28.95	50.81
				Primary + Secondary[3] Stress Range	67.83	N/A
Component Supports	Support Columns[4]	SA-333 Grade 1	$S_y = 28.27$	Bending	18.66	24.88
				Tensile	16.96	22.61
				Combined	1.00	1.00
				Compressive	11.84	15.79
				Interaction	1.00	1.00

Table 3.8-5 (Continued)

ALLOWABLE STRESSES FOR VENT SYSTEM COMPONENTS AND COMPONENTS SUPPORTS

Item		Material[1]	Material Properties[2]	Stress Type	Allowable Stress (ksi)	
					Service Level B	Service Level C
Welds	Support Collar Plate to Vent Header	SA-516 Grade 70	S_{mc} = 19.30 S_y = 33.87	Primary	15.01	26.42
				Secondary	45.03	N/A

Notes:

1. The head and shell of the Unit 2 drywell were fabricated of SA-212 Grade B plate and manufactured to A-300 requirements. This material specification no longer exists and was replaced by SA-516 Grade 70, carbon steel.
2. Material properties taken at maximum event temperatures.
3. Thermal bending stresses are excluded when evaluating primary-plus-secondary stress range.
4. Stresses due to thermal loads may be excluded when evaluating component supports.

Table 3.8-6

CLASS MC PIPING ACCEPTANCE CRITERIA

Code Equation[1]	Service Level	Stress/Usage Limit	Allowable Stress (ksi)
9	Design	1.5 S_m	24.75
9	C	2.25 S_m	37.16
10	A, B	3.0 S_m	49.50
12[2]	A, B	3.0 S_m	49.50
13[2]	A, B	3.0 S_m	49.50
Fatigue[3]	A, B	1.0	N/A

Notes:

1. See NB-3652 and NB-3653 of the ASME Code.
2. Required only if Equation 10 is not satisfied.
3. Cumulative fatigue usage calculation per NB-3653.

Table 3.8-7

MAXIMUM VENT SYSTEM STRESSES FOR CONTROLLING LOAD COMBINATIONS

Item		Stress Type	Load Combination Stresses (ksi)(1)									
			SBA II(1)		IBA I(1)		DBA I(1)		DBA II(1)		DBA III(1)	
			Calculated Stress	Ratio to Allowable(2)	Calculated Stress	Ratio to Allowable(2)	Calculated Stress	Ratio to Allowable(2)	Calculated Stress	Ratio to Allowable(2)	Calculated Stress	Ratio to Allowable(2)
Components	Drywell Shell	Local Primary Membrane	17.07	0.59	12.68	0.44	18.56	0.49	17.39	0.46	20.35	0.40
		Primary + Secondary Stress Range	61.09	0.90	47.44	0.70	N/A	N/A	58.26	0.86	N/A	N/A
	Vent Line	Primary Membrane	10.15	0.94	16.18	0.80	17.03	0.88	16.94	0.88	25.57	0.75
		Local Primary Membrane	9.86	0.34	8.69	0.30	5.39	0.14	9.09	0.24	10.21	0.20
		Primary + Secondary Stress Range	30.82	0.45	26.91	0.40	N/A	N/A	27.75	0.41	N/A	N/A
	Vent Line/ Vent Header Spherical Junction	Primary Membrane	9.47	0.49	7.91	0.41	7.39	0.38	8.13	0.42	10.07	0.30
		Local Primary Membrane	15.91	0.55	13.35	0.46	13.67	0.47	14.52	0.50	20.04	0.39
		Primary + Secondary Stress Range	48.23	0.71	35.32	0.52	N/A	N/A	39.15	0.58	N/A	N/A
	Vent Header	Primary Membrane	17.46	0.91	14.66	0.76	18.68	0.97	17.85	0.93	25.98	0.77
		Local Primary Membrane	20.93	0.72	9.27	0.32	18.96	0.50	18.59	0.49	19.87	0.39
		Primary + Secondary Stress Range	51.67	0.76	29.27	0.43	N/A	N/A	47.38	0.70	N/A	N/A
	Downcomer	Primary Membrane	8.52	0.44	3.80	0.20	11.88	0.62	5.67	0.29	16.25	0.48
		Local Primary Membrane	20.05	0.69	9.96	0.34	16.63	0.44	16.92	0.45	18.96	0.37
		Primary + Secondary Stress Range	34.70	0.51	10.85	0.16	N/A	N/A	34.81	0.51	N/A	N/A

Table 3.8-7 (Continued)

MAXIMUM VENT SYSTEM STRESSES FOR CONTROLLING LOAD COMBINATIONS

Item		Stress Type	Load Combination Stresses (ksi)(1)									
			SBA II(1)		IBA I(1)		DBA I(1)		DBA II(1)		DBA III(1)	
			Calculated Stress	Ratio to Allowable(2)	Calculated Stress	Ratio to Allowable(2)	Calculated Stress	Ratio to Allowable(2)	Calculated Stress	Ratio to Allowable(2)	Calculated Stress	Ratio to Allowable(2)
Components	Support Collar Plate	Primary Membrane	1.89	0.10	1.14	0.06	3.12	0.16	1.43	0.07	3.20	0.09
		Local Primary Membrane	6.28	0.22	5.07	0.18	9.97	0.26	5.48	0.15	10.22	0.20
		Primary + Secondary Stress Range	57.50	0.85	3.41	0.50	N/A	N/A	49.20	0.73	N/A	N/A
Component Supports	Support Columns	Bending	9.70	0.50	6.73	0.35	3.07	0.16	11.73	0.60	6.93	0.27
		Tensile	3.86	0.22	5.44	0.20	13.32	0.75	5.23	0.30	13.50	0.57
		Combined	0.72	0.72	0.55	0.55	0.91	0.91	0.90	0.90	0.84	0.84
		Compression	5.14	0.42	3.56	0.45	3.48	0.29	3.39	0.28	4.42	0.27
		Interaction	0.99	0.99	0.88	0.88	0.46	0.46	0.91	0.91	0.58	0.58
Welds	Column Ring Plate to Vent Header	Primary	6.79	0.45	4.45	0.30	10.64	0.71	6.00	0.40	10.99	0.42
		Secondary	11.29	0.25	7.14	0.16	N/A	N/A	9.50	0.21	N/A	N/A

Notes:

1. See Dresden Plant Unique Analysis Report for discussion of load combinations; SBA = small break accident; IBA = intermediate break accident; DBA = design basis (recirculation line break) accident.
2. See Table 3.8-4 for allowable stresses.
3. Local stresses are reported at the vent line/vent header junction.

Table 3.8-8

STRESS ANALYSIS RESULTS — CLASS MC PIPING

ASME Code Paragraph	Code Equation	Service Level	Stress (ksi)/Usage	
			Calculated	Allowable
NB-3652	9	Design	2.60	24.75
	9	C	7.50	37.13
NB-3653	10	A, B	69.26[1]	49.50
	12	A, B	3.98	49.50
	13	A, B	44.18	49.50
	Fatigue usage	A, B	0.09	1.0

Note:

1. This is acceptable in accordance with the ASME Code, as long as equations 12 and 13 (from NB-3653) are satisfied.

Table 3.8-9

ALLOWABLE STRESSES FOR SUPPRESSION CHAMBER COMPONENTS AND SUPPORTS

Item		Material(1)	Material Properties (ksi)(2)	Stress Type	Allowable Stress (ksi)		
					Service Level B	Service Level C	Service Level D
Components	Shell	SA-516 Grade 70	S_{mc} = 19.30 S_{ml} = 23.17 S_y = 35.86 S_u = 70.00	Primary Membrane	19.30	35.86	41.65
				Local Primary Membrane	28.95	53.79	62.48
				Primary + Secondary Stress Range (3)	69.51	N/A	N/A
	Ring Girder	SA-516 Grade 70	S_{mc} = 19.30 S_{ml} = 23.17 S_y = 35.86 S_u = 70.00	Primary Membrane	19.30	35.86	41.65
				Local Primary Membrane	28.95	53.79	62.48
				Primary + Secondary Stress Range (3)	69.51	N/A	N/A
Component Supports	Column Connection(4)	SA-516 Grade 70	S_y = 35.86	Membrane	21.52	28.69	43.04
				Extreme Fiber)	26.90	35.87	53.80
	Saddle(4)	SA-516 Grade 70	S_y = 35.86	Membrane	21.52	28.69	43.04
				Extreme Fiber	26.90	35.87	53.80

Table 3.8-9 (Continued)

ALLOWABLE STRESSES FOR SUPPRESSION CHAMBER COMPONENTS AND SUPPORTS

Item		Material[1]	Material Properties (ksi)[2]	Stress Type	Allowable Stress (ksi)		
					Service Level B	Service Level C	Service Level D
Welds	Ring Girder to Shell	SA-516 Grade 70	S_{mc} = 19.30 S_{ml} = 23.17 S_y = 35.86	Primary	15.02	27.89	32.04
				Primary + Secondary	54.07	N/A	N/A
	Column Connection to Shell	SA-516 Grade 70	S_{mc} = 19.30 S_{ml} = 23.17 S_y = 35.86	Primary	15.02	27.89	32.04
				Primary + Secondary	54.07	N/A	N/A
	Saddle to Shell	SA-516 Grade 70	S_{mc} = 19.30 S_{ml} = 23.17 S_y = 35.86	Primary	15.02	38.03	44.17
				Primary + Secondary	61.42	N/A	N/A

Notes:

1. The head and shell of the Unit 2 drywell were fabricated of SA-212 Grade B plate and manufactured to A-300 requirements. This material specification no longer exists and was replaced by SA-516 Grade 70, carbon steel.
2. Material properties are taken at the maximum event temperature.
3. Thermal bending stresses may be excluded when comparing primary-plus-secondary stress range values to allowables.
4. Stresses due to thermal loads may be excluded when evaluating component supports.

Table 3.8-10

MAXIMUM SUPPRESSION CHAMBER STRESSES FOR CONTROLLING LOAD COMBINATIONS

Item		Stress Type	Load Combination[4] Stresses (ksi)							
			IBA III		IBA IV		DBA III		DBA IV	
			Calculated Stress	Ratio to Allowable	Calculated Stress	Ratio to Allowable	Calculated Stress	Ratio to Allowable	Calculated Stress	Ratio to Allowable
Components	Shell	Primary Membrane	16.96	0.88	17.28	0.89	15.93	0.82	22.18	0.62
		Local Primary Membrane	21.24	0.73	20.22	0.70	15.09	0.52	27.04	0.50
		Primary plus Secondary Stress Range	63.05	0.91	64.22	0.92	30.18	0.43	N/A	N/A
	Ring Girder	Primary Membrane	19.24	0.99	18.84	0.98	19.09^(1)^	0.99	24.62	0.69
		Local Primary Membrane	25.53	0.88	23.24	0.80	28.17^(1)^	0.97	26.02	0.48
		Primary plus Secondary Stress Range	45.92	0.66	47.20	0.68	57.41[1]	0.83	N/A	N/A
Supports	Column	Membrane	17.25	0.80	17.62	0.82	8.72	0.41	19.24	0.67
	Connection	Extreme Fiber	20.09	0.75	20.61	0.77	8.72	0.32	20.87	0.58
	Saddle	Membrane	18.29	0.85	18.25	0.85	10.16	0.47	27.47	0.96
		Extreme Fiber	18.29	0.68	18.25	0.68	10.16	0.38	27.47	0.77

Table 3.8-10 (Continued)

MAXIMUM SUPPRESSION CHAMBER STRESSES FOR CONTROLLING LOAD COMBINATIONS

Item		Stress Type	Load Combination[4] Stresses (ksi)							
			IBA III		IBA IV		DBA III		DBA IV	
			Calculated Stress	Ratio to Allowable	Calculated Stress	Ratio to Allowable	Calculated Stress	Ratio to Allowable	Calculated Stress	Ratio to Allowable
Welds	Ring Girder to Shell	Primary	19.70[3]	0.87	17.76[3]	0.79	21.62[2]	0.96	24.40	0.87
		Secondary	45.94	0.83	42.57	0.79	50.30[2]	0.93	N/A	N/A
	Column Connection to Shell	Primary	20.25	0.91	21.90	0.97	11.92	0.79	40.59	0.97
		Secondary	27.88	0.52	29.39	0.54	14.21	0.26	N/A	N/A
	Saddle to Shell	Primary	19.05	0.93	18.36	0.90	11.81	0.58	30.65	0.81
		Secondary	19.05	0.31	18.36	0.30	11.81	0.19	N/A	N/A

Notes:

1. A stress intensification factor of 1.10 has been used to account for pitting in the torus shell.
2. These results are governed by the zero ring girder stiffener model.
3. This local primary membrane stress has an allowable based on 1.5 S_{mc}.
4. IBA = intermiediate break accident; DBA = design basis (recirculation line break) accident.

Table 3.8-11

ALLOWABLE STRESSES FOR CLASS I STRUCTURES

Loading Conditions	Reinforcing Steel Maximum Allowable Stress	Concrete Maximum Allowable Stress: Compression	Shear	Bearing	Tension on Net Section	Structural Steel Shear on Gross Section	Compression on Gross Section	Bending
Dead, live, operating, and OBE seismic (0.1 g)	$0.5\ F_y$	$0.45\ f_{c'}$ 2	$1.1\sqrt{f_{c'}}$ 3	$0.25\ f_{c'}$ 4	$0.60\ F_y$	$0.40\ F_y$	Varies with slenderness ratio(2)	$0.66\ F_y$ to $0.60\ F_y$
Dead, live, operating, and wind	$0.667\ F_y$	$0.60\ f_{c'}$ 5	$1.467\sqrt{f_{c'}}$ 6	$0.333\ f_{c'}$ 7	$0.80\ F_y$	$0.53\ F_y$	Varies with slenderness ratio(2)	$0.88\ F_y$ to $0.80\ F_y$
Dead, live, operating, and SSE seismic (0.2 g)	---	[Safe shutdown of the plant can be achieved](1)				---	---	---
(Extended Power Uprate Loads)		Refer to Section 3.9.3.1.3.5						

F_y = minimum yield point of material $\quad$ $f_{c'}$ 8 = compressive strength of concrete

Notes:

1. The structure was analyzed to assure that a proper shutdown can be made during ground motion having twice the intensity of the spectra shown in Figure 3.7-1 even though stresses in some of the materials may exceed the yield point.
2. The slenderness ratio for compression members in ceiling mounted supports for cable trays, conduits, and HVAC ductwork is limited to 300.

Table 3.8-12

COMPARISON OF ACCELERATIONS FOR THE CONCRETE PORTION OF THE TURBINE BUILDING AND CONTROL ROOM

Elevation	Control Room	Turbine Building
620'-0"	0.46 g	0.90 g
561'-5"	0.24 g	0.28 g
549'-0"	0.19 g	0.20 g
534'-0"	0.14 g	0.15 g
515'-5"	0.10 g	0.10 g

3.9 MECHANICAL SYSTEMS AND COMPONENTS

This section addresses the design of mechanical systems and components. The design transients (thermal cycles) applicable to the reactor vessels and vessel fatigue evaluation are described in Section 3.9.1.

Section 3.9.2 describes dynamic testing and analysis for mechanical systems and components. Included in Section 3.9.2 are a brief discussion of equipment qualification, and results of tests and analyses demonstrating the acceptability of the reactor vessel internals under flow-induced vibration loads.

The qualification of the reactor vessel and supports, pressure-retaining equipment, piping, and piping supports is the focus of Section 3.9.3. For each of these component types, acceptance criteria, loading conditions, and design evaluations are presented.

Section 3.9.4 references Section 4.6 for a discussion of the design of the control rod drive (CRD) system.

Section 3.9.5 summarizes the layout, design bases, and qualification of the reactor vessel internals.

Finally, the inservice testing program for pumps and valves is described in Section 3.9.6.

Throughout this Section 3.9, references and discussions are made to original criteria and current acceptance criteria for the USAS B31.1 and applicable sections of the ASME Section III and Section VIII Boiler and Pressure Vessel Codes (BPVC). For analysis purposes, more current editions (2004 with 2005 addenda for B31.1 and 2007 with 2008 addenda for the ASME BPVC) may be used for allowable stress values. This is applicable to ASME Section III Class 2 and 3 piping and components and ASME Section VIII, Div 1 pressure vessels. Pressure-retaining items under the jurisdiction of the Illinois Emergency Management Agency (IEMA) must also follow the rules of the National Board Inspection Code (NBIC), Part 3, Section 3.4.2. This is based on a code reconciliation (Reference 47).

3.9.1 Special Topics for Mechanical Components

This section presents the design transients and addresses the fatigue evaluation of the reactor vessel and considerations for the evaluation of the faulted condition. Overall acceptance criteria, loading conditions, and a discussion of the design evaluation of the vessels and supports are presented in Section 3.9.3.1.1. The evaluation of reactor vessel internals is covered in Sections 3.9.2 (vibration loadings) and 3.9.5 (other design loadings).

3.9.1.1 Design Transients

The construction permits for Dresden Units 2 and 3 were issued on January 10, 1966, and October 14, 1966, respectively. At that time, the ASME Code covered only reactor vessels and associated piping up to and including the first isolation or stop valve. Pumps, piping, and valves were built primarily to the USAS B31.1 Power Piping Code rules.

The following subsection provides a description of the design transients applicable for the fatigue evaluation of the reactor pressure vessel.

3.9.1.1.1 Reactor Pressure Vessel Fatigue Evaluation

The reactor pressure vessel (RPV) was originally designed for fatigue to a set of thermal cycles, or design allowables. The original RPV stress report showed that the vessel and its components could withstand the designated number of cycles with a fatigue usage factor less than 1.0, as required by the ASME Code. Based on a GE thermal cycle counting procedure,[1] operating transient cycles through October 1989 were redefined using actual plant data for Units 2 and 3.[2] The method for predicting cycles through 40 years resulted in some cycle estimates at year 40 that were higher than the original design basis.

The comparison of predicted cycles and design allowable cycles for Unit 2 showed six events of concern: heatup, cooldown, reduction of power, turbine roll, scram and safety relief valve (SRV) blowdown. The allowable for turbine trips was also exceeded, but the turbine trip cycles were for information only, as those events were also counted as either a loss of feedwater (FW) heaters or a scram. All other events have substantial margin between the 40-year prediction and the allowable. Furthermore, the fact that the Unit 2 feedwater nozzles experience relatively few batch feedwater injections in hot standby is a significant benefit, as this is the more severe event for that nozzle.

The comparison of predicted and allowable cycles for Unit 3 showed different results compared to Unit 2. There were only three cases where predicted events exceeded design allowables: heatup, cooldown and scram. Other events have adequate margin. As with Unit 2, batch feedwater injection during hot standby shows relatively few cycles, providing a source of analysis margin for the feedwater nozzle.

The new design basis allowable cycles shown in Table 3.9-1 do not necessarily indicate the number of cycles at which the vessel components' fatigue usages reach the allowable of 1.0. For many components, the usage for the design cycles is very low (less than 0.1). Therefore, it is a very simple matter to increase the allowable numbers of the events of concern mentioned above. For components like the feedwater nozzle, which have a relatively high design fatigue usage, more involved analyses can demonstrate that additional margin is available. In fact, it is possible to reduce the feedwater nozzle fatigue usage by accounting for the few batch injection cycles predicted. This provides margin for life extension purposes or for longer intervals between feedwater thermal sleeve seal refurbishments for rapid cycling fatigue considerations.

Fatigue reanalysis with revised cycle allowables was completed for Quad Cities. In that analysis, the support skirt was the critical location for fatigue. New finite element stress analysis of the skirt was used to determine usage for 245 cooldowns, 12 SRV blowdowns (same effect on skirt as a cooldown) and 250 heatups. The resulting usage is shown on the table below. The Quad Cities and Dresden fatigue analyses for the support skirt and closure studs have identical results, so the Quad Cities analysis can be applied to Dresden.

All vessel components have acceptable fatigue usage for 40 years for the usage factors shown in the table below.

All vessel components have acceptable fatigue usage for 40 years when the following usage factors are applied to the limiting components:

Component	EPU Fatigue Usage Factor	Code Allowable
Shroud Support	0.820	1.0
Support Skirt	0.862	1.0
Feedwater Nozzle	0.748	1.0
Closure Studs	0.750	1.0

Structural Integrity Associates (SIA) was contracted early in 1998 to independently review the General Electric analyses (reference 1) which updated the original Babcock & Wilcox Stress Report. The SIA report (reference 2) identified several conservatisms that had been applied in the General Electric analysis that resulted in high fatigue usage factors for the closure head studs. General Electric has re-evaluated the fatigue usage to the closure head studs (reference 3). The projected fatigue usage of the closure head studs after forty years of service life is now projected to be 0.79 for Dresden 2 and 0.74 for Dresden 3. As a result, the closure studs are not the vessel fatigue life-limiting component.

3.9.1.2 Considerations for the Evaluation of the Faulted Condition

For a discussion of dynamic analysis methods applicable to seismic evaluation of piping, see Section 3.7.

3.9.2 Dynamic Testing and Analysis

The following subsections discuss testing performed to evaluate the effects of piping vibration and the type of analysis performed to qualify safety-related mechanical equipment.

3.9.2.1 Piping Vibration, Thermal Expansion, and Dynamic Effects

The reactor vessel internals have undergone vibration testing, as discussed in Section 3.9.2.4.

As a consequence of the extended power uprate (EPU), the main steam and feedwater piping experience higher flow velocities than at the original licensed power. In order to demonstrate acceptability of operating at EPU power levels, a vibration monitoring program was implemented during the EPU power ascension. Accelerometers were mounted at various points on the main steam and feedwater piping to assess vibration. These points were monitored at increments from below original licensed power to full EPU power. The vibration results were compared to acceptance criteria developed per ASME OM-S/G-1997, Standards and Guides for Operation and Maintenance of Nuclear Power Plants.

Since the measure vibration levels were acceptable during EPU implementation on Dresden Unit 2 (December 2001), the monitoring points were reduced in number on the substantially similar Unit 3 piping to a representative sample. The EPU vibration monitoring program results in all cases support the operation of the Dresden units at EPU power levels.

3.9.2.2 Seismic Qualification of Safety-Related Mechanical Equipment

Safety-related mechanical equipment is qualified by either dynamic or static analysis methods.

Where a dynamic analysis was not performed, the horizontal seismic coefficients for rigid equipment in the reactor-turbine building were considered to be equal to or greater than the building acceleration at the installed elevation. The vertical seismic coefficient was considered as two-thirds of ground acceleration, i.e., 0.067 g. The input motion to the equipment was assumed to be the absolute acceleration of the structure at the points of support of the equipment.

A reassessment of the seismic adequacy of mechanical and electrical equipment at Dresden Unit 2 was performed under the systematic evaluation program (SEP), Topic III-6, titled, "Seismic Design Considerations." In addition, Generic Letter (GL) 87-02, "Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue (USI) A-46," requires verification of seismic adequacy. Section 3.10.1 addresses the methods which will be used to respond to Generic Letter 87-02.

3.9.2.3 Dynamic Response Analysis of Reactor Internals Under Operational Flow Transients and Steady-State Conditions

The vibration of components inside the reactor vessel during normal operation could result in a control instability or result in a structural failure that might affect the coolant flow, the operational efficiency, or general operability. Turbulence and hydrodynamic instabilities in the coolant flow in past designs have been the main sources of the vibratory excitations that have produced significant vibratory amplitudes. The excitation forces from the discontinuities in two-phase flow have produced some vibrations at the steam separators; however, these amplitudes have been found to be negligibly small.

The coolant flow paths are designed to avoid the known sources of flow-induced vibratory excitation forces. Coolant flow is directed along paths that minimize equipment loading and vibrations, i.e., wherever possible, parallel and laminar flow is used. Where flow normal to a surface or a cylindrical body is required, the structural resonant frequencies are not allowed to coincide with the known excitation frequencies, such as the vortex shedding frequency, inherent in that flow configuration. Where the flow path can be modulated by the motion of the components, such as seals and vanes, the system hydrodynamic stability was checked analytically.

The overall reactor internals system was analyzed as a multidegree of freedom system of lumped components. This analysis determined the system's natural frequencies, the resultant vibratory mode shapes, and the relationship between the vibration amplitudes and the critical stresses in the system.

With this information and the knowledge from past experience regarding the expected sources and forces of vibratory excitation, the vibration integrity of the system was checked. If the system integrity was at all questionable, then correctional design changes were incorporated and the system was reanalyzed until all doubts had been eliminated.

3.9.2.4 Preoperational Flow-Induced Vibration Testing of Reactor Internals

Every effort was made in the design stage to determine sources of vibratory excitation. However, since it was impossible to predict all sources of vibratory excitation, a vibration measurement and testing program was included in the preoperational tests. These vibration measurements were designed to detect the presence of any gross structural dynamic instabilities as well as the steady-state vibration response (at significant levels) of the reactor internal components.

The primary purpose of the vibration measurements made on the reactor and internals was to demonstrate the mechanical integrity of the system when subjected to vibratory motions. These measurements were also designed to check the validity and accuracy of the analytical procedures used to calculate the seismic vibration characteristics.

The direction and approximate location of the points at which the vibration measurements were made are indicated in Table 3.9-2. All points were measured during cold coolant-flow tests for flowrates up to approximately 85% of rated mass-flow. Only those points whose motion would be affected by two-phase (water and steam) flow were measured during power operation.

The vibratory motion of the incore guide tubes could be measured without the fuel in place. The instrumented control rod guide tubes required, as a minimum, the fuel to be in place on these tubes for them to give correct vibration response readings. In both cases, however, the orifice plates had to be in place so that flow patterns would be close to normal.

The sensors used were accelerometers, strain gauges, and linear differential transducers, and were of such design as to withstand the environment to which they were subjected.

In all vibration tests, the coolant was maintained at ±20°F of starting temperature as much as practical. To maintain this temperature with the recirculation pumps operating, shutdown cooling was needed.

APED-5453[3] and GEAP-22274[4] provide a generalized discussion of the vibration analysis and testing of the reactor internals.

Extensive vibration monitoring was done on Dresden Unit 2. The results of the vibration monitoring conducted on Dresden Unit 2 show all values to be within the design specified limit. Since the design and construction of Dresden Unit 3 is identical to Dresden Unit 2, vibration monitoring is not necessary on Dresden Unit 3.

No study is being made to monitor for the presence of loose parts within the primary coolant system. Loose parts within the primary coolant system can be detected by anomalies in the flow instrumentation available on the reactor coolant system.

As part of the Dresden Unit 2 and 3 steam dryer replacement program, the Quad Cities Unit 2 dryer and four main steam lines (MSL) were instrumented for the purpose of measuring the pressure loads acting on the dryer. Structural analyses were performed to demonstrate the adequacy of the Dresden Unit 3 replacement steam dryer design using predicted loads based on the four MSL strain gage measurements obtained during startup with the Dresden Unit 3 replacement dryer. Dresden Unit 2 replacement dryer installation compared MSL "B" strain gage data obtained with old dryer in D2R19 with Dresden Unit 3 MSL "B" strain gage data obtained with the replacement dryers (Reference 44). The results determined that the replacement dryers satisfy both the fatigue limit and ASME Code limits for normal, upset and faulted events at EPU conditions.

3.9.3 ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

This subsection provides a description of the loads and acceptance criteria applicable to mechanical systems and components.

3.9.3.1 Load Combinations, Design Transients, and Stress Limits

The following subsections provide a discussion of the design of the reactor vessel and vessel supports (Section 3.9.3.1.1), mechanical equipment (Section 3.9.3.1.2) and piping (Section 3.9.3.1.3). Fatigue evaluation of the reactor vessel was discussed previously in Section 3.9.1.1.1.

As defined in Section 3.2, mechanical systems and components which are designated as safety Class I are either components vital to safe plant shutdown and removal of decay heat or are systems or components whose failure could cause significant release of radioactivity. Throughout this section use of the term "Class I" refers to this classification basis and not to ASME Code classifications. See Section 3.2 for definition of all safety classifications.

3.9.3.1.1 Reactor Pressure Vessel and Supports

The reactor vessel is described in Section 5.3. The reactor vessel is supported by a steel skirt. The top of the skirt is welded to the bottom of the vessel. The base of the skirt is continuously supported by a ring girder fastened to a concrete foundation, which carries the load through the drywell to the reactor building foundation slab.

Stabilizer brackets, located below the vessel flange, are connected to tension bars with flexible couplings (see Figure 3.9-1). The bars are then connected through the drywell to the concrete structure outside the drywell to limit horizontal vibration and to resist seismic and jet reaction forces. The bars are designed to permit axial expansion.

3.9.3.1.1.1 Acceptance Criteria

The Dresden reactor pressure vessels were designed according to ASME Section III, 1963 Edition, including the Summer 1964 Addenda, plus code case interpretations pertaining to primary nuclear reactor vessels applicable on February 8, 1965 (see Reference 43). Applicable code cases and exceptions are described in Section 3.2.

Design of the primary reactor vessel supports was governed by the ASME Code, the American Institute for Steel Construction (AISC) Structural Steel Code, the American Concrete Institute (ACI) Code.

A re-evaluation of the RPV and RPV support system was performed in 1995 as part of the core shroud modification project. All affected components were demonstrated to satisfy the applicable code requirements.

3.9.3.1.1.2 Design Loadings

Information regarding the design transients and fatigue evaluation of the reactor pressure vessel is presented in Section 3.9.1.1.

This subsection describes the loads and load combinations applicable to the design of the reactor pressure vessel internals and supports.

The applicable loads for the reactor vessel internals and supports, and for the emergency core cooling system equipment and piping covered in Section 3.9.3.1.2.2, are defined as follows:

D = Dead load of structure and equipment plus any other permanent loads contributing stress, such as soil or hydrostatic loads or operating pressures, and live loads expected to be present when the plant is operating.

P = Pressure due to loss-of-coolant accident (LOCA).

R = Jet force or pressure on structure due to rupture of any one pipe.

H = Force on structure due to thermal expansion of pipes under operating conditions.

E = Operating basis earthquake (OBE) load, ground horizontal g = 0.10. vertical g = 0.067.

T = Thermal loads on containment due to LOCA.

E′ = Design basis earthquake (DBE) load, ground horizontal g = 0.20, vertical g = 0.13.

Following are the load combinations used for the reactor vessel and vessel supports.

Reactor Primary Internals

D + E Stresses which occur as a result of the maximum possible combination of loadings encountered in operational conditions; they are within the stress criteria of ASME Section III, Class A Vessel.

D + E′ The primary stresses and primary plus secondary stresses are examined on a rational basis taking into account elastic and plastic strains. These strains are limited to preclude failure by deformation which would compromise any of the engineered safeguards or prevent safe shutdown of the reactor.

P + D Primary stresses are within the stress criteria of ASME Section III, Class A. The primary stresses and primary plus secondary stresses are examined on a rational basis taking into account elastic and plastic strains. These strains are limited to preclude failure by deformation which would compromise any of the engineered safeguards or prevent safe shutdown of the reactor.

Reactor Primary Vessel Supports

D + H + E Stresses remain within Code allowables without the usual increase for earthquake loadings (AISC for structural steel; ACI for reinforced concrete).

D + H + R + E Stresses do not exceed:

- 150% of AISC allowables for structural steel
- 90% of yield stress for reinforcing bars
- 85% of ultimate stress for concrete

D + H + E′ No functional failure - usually stresses do not exceed the yield point of the material for steel or the ultimate strength of the concrete. If these limits are exceeded energy absorption capacity is determined and compared to the energy input from the earthquake. The design is such that energy absorption capacity exceeds energy input.

3.9.3.1.1.3 Design Evaluation

When Dresden was originally designed, two independent calculations were made for the seismic analysis of the reactor and reactor building. The first analysis was performed by J.A. Blume and Associates Engineers in January of 1967 (Appendix 5.A, Exhibit H). Seismic loads on the RPV support skirt and shroud support were extracted from the Blume report and used in the stress analysis performed by Babcock & Wilcox (B&W). Later, in December 1968, General Electric Company (GE) performed a more detailed seismic analysis of the RPV and internals (Reference 16). General Electric Company's calculations showed that the seismic loads on the vessel skirt and shroud support were less than the seismic loads calculated by J.A. Blume and Associates and, therefore, the stress analysis by B&W is conservative.

As part of the core shroud repair project, new horizontal seismic analysis of Dresden Units 2 and 3 were performed. The seismic model includes the RPV, internals and supporting structures. Prior to use for the core shroud repair, the model was benchmarked against the design basis model used for the original seismic analysis. The revised seismic model included updates to incorporate the current fuel as well as the core shroud repair hardware in the form of three horizontal and one rotational springs. The horizontal springs are located at the elevation of the core support plate, the jet pump riser braces and the top guide. The rotational spring represents the tie rod assemblies.

The seismic analyses were performed using the time history method. The input motions included the north-south component of the May 18, 1940 El Centro earthquake record and a synthetic time history record enveloping the Housner spectra. Both time histories were normalized to a peak ground motion of 0.20 g for the Design Basis Earthquake (DBE). Bounding DBE and Operation Basis Earthquake (OBE) loads were obtained. All relevant modes of vibration of the coupled system were considered. The damping factors utilized are defined in Table 3.7-1. The resulting shears, moments, displacements and accelerations were determined and used to reevaluate the adequacy of the reactor pressure vessel support system.

3.9.3.1.1.3.1 Original Evaluation

The RPV was analyzed as a single-mode system fixed at the base of the concrete pedestal, with lateral support at the stabilizer elevation connecting the RPV to the sacrificial shield and a horizontal pipe truss system connecting the sacrificial shield to the building.

The RPV seismic analysis is affected by seismic motions at the support points only. Therefore, the RPV can be decoupled from the rest of the building and analyzed separately taking into consideration the effects of input motion at all supports and conservatively combining the individual effects.

Although two different methods of analysis were used in the design of the reactor vessel and the reactor building (response-spectrum method and time-history method, respectively), the inputs to the separate analyses were consistent. Only the damping coefficients for the two analyses differed.

A critical damping factor of 2% was used in the analysis of the RPV and the concrete supports. This resulted in conservative interaction forces between the reactor and supports.

3.9.3.1.1.3.2 1995 Evaluation

As part of the core shroud repair project, new evaluations and stress reports were prepared to determine the effect of the revised loads and displacements on the core shroud, RPV and RPV internals. Evaluations of the core shroud, core shroud repair hardware, RPV, RPV stabilizer system, core plate wedges and RPV support system were performed, demonstrating that the code requirements were satisfied.

1997 Evaluation

Evaluations of the ATRIUM-9B fuel assemblies including fuel channel, fuel rod, water channel, upper and low tie plates, and spacers were performed and they met their design basis (References 34, 35, and 36).

3.9.3.1.1.3.3 Support System Evaluation

Vertical loads from the RPV are transmitted to the foundation through the RPV skirt, RPV support girder, and RPV support pedestal. Lateral loads are transmitted to the building through RPV stabilizers. The RPV stabilizers are attached near the top third of the RPV and are connected to the top of the concrete and steel shield wall. The shield wall in turn is anchored at the base to the top of the RPV pedestal and restrained at the top by a horizontal tubular truss system. The lateral loads are transmitted through the truss system to the drywell shear lug mechanism. This shear lug mechanism permits vertical movement of the drywell but restricts rotational movement. However, lateral loads are transmitted through the shear lug mechanism to the heavy concrete envelope around the drywell which is part of the reactor building. A portion of the lateral loads are transmitted from the reactor pressure vessel to the RPV pedestal and then on to the foundation.

Following is a discussion of the critical stresses in all parts of the supporting structure (reactor skirt, steel ring girder, bolts, concrete pedestal, including thermal, seismic, and jet stresses) which may arise during an earthquake. Figures 3.9-1, 3.9-2, and 3.9-3 are sketches of the supporting structure and the vessel stabilizers.

For the RPV to ring girder, the horizontal shear at the bottom of the RPV skirt flange is transferred to the top flange of the ring girder through the high-strength, bolt-friction-type connection. Friction between the connecting parts exists under all load conditions because the bolt-proof load is greater than the maximum tension any bolt will be subject to. Hence, the skirt ring and the girder will always be in contact and the friction is of such magnitude to withstand the shear forces due to earthquake.

The horizontal reactor shears are carried through the anchor bolts into the reactor support pedestal concrete. The 120 anchor bolts are not considered to have oversized hole connections; the clearances have normal construction tolerances which must be available in order to assemble the structure. The resulting low stresses, coupled with the fact that bolt friction and sole plate bearing allowance for the transfer of shear loads was neglected, are standard practice and result in a conservative design.

The reactor vessel skirt analyses resulted in primary membrane and primary shear stresses that are within the code allowables. Shear is carried through the skirt that acts as a large membrane or shear ring.

The allowable stresses for the various steel RPV support components and the RPV concrete pedestal are within the code allowables. See references 17, 24, 25, 26, 27 and 28 for the details of the re-evaluation of these components that was performed as part of the core shroud modification project.

The concrete structure inside the drywell that supports the reactor pedestal is prevented from sliding by a 6-in. x 1-in. continuous shear plate that stands vertically, is welded to the inside of the drywell, and is located on the same diameter as the drywell skirt. The shear in the two 3/8-inch fillet welds attaching this shear plate to the drywell under the OBE is 3550 psi. The bearing on the concrete for this condition is 315 psi. These values are within the allowable of 15,800 psi for the weld metal and 935 psi for the concrete. For the SSE the stress in the weld is 7000 psi and in bearing on the concrete is 615 psi. The weld stress and the concrete allowable bearing stress for the SSE are still below the working stress allowable.

There is free movement of air between the inside and outside of the reactor pedestal; therefore, no temperature gradient through the concrete wall need be considered.

The vessel does not rock under earthquake conditions because the tension force due to the earthquake does not exceed the clamping force of the bolts. Thus the plates cannot separate and there can be no stretch of the bolts or increase in the tensile load.

The anchor bolts connecting the ring girder to the concrete pedestal will stretch under seismic loading. The effect of this stretch in the anchor bolts is to cause a rotation of the RPV about its base, resulting in a horizontal deflection of the RPV above its base. Evaluations of the effect of this phenomena have concluded that due to the very small relative elongation of the anchor bolts, the corresponding increase in the lateral displacement of the RPV is also small, and therefore its influence on rocking response is considered negligible.

The thermal expansion effects between the ring girder and concrete pedestal on the anchor bolts were conservatively evaluated by assuming that the operating temperature at the bottom flange of the ring girder (maximum 150°F) is attained while the concrete pedestal is at 70°F. This results in a differential temperature of 80°F, and the ring girder will try to expand radially.

To restrain this expansion requires a force of 4280 lb/in. inward on the circumference of the ring girder. It was assumed that this force is shared by the ring girder anchor bolts and the shield wall anchor bolts in proportion to their cross sectional areas. Under these conditions the resulting shear stress in both the ring girder anchor bolts and the shield wall anchor bolts is 2400 psi. To develop this shear in the ring girder anchor bolts, the space in the oversize holes of the ring girder bottom flange was filled. To transfer the force from the ring girder to the shield wall an additional depth of concrete of 4 inches was placed between the ring girder and the shield wall. The allowable shearing stress is 10,000 psi.

3.9.3.1.2 Mechanical Equipment

The following section describes the acceptance criteria, loading conditions, and evaluations applicable to mechanical equipment at Dresden Station.

3.9.3.1.2.1 Acceptance Criteria

Applicable codes, addenda and code cases for the design of Class I pumps and valves of the reactor coolant pressure boundary are described in Section 3.2.

Class II defines all equipment which are not designated Class I. Class II pumps and valves have been designed in accordance with normal practices for design of power plants in the State of Illinois, including local building codes.

3.9.3.1.2.2 Loading Conditions

Using the nomenclature defined in Section 3.9.3.1.1.2, following are the load combinations applicable to the emergency core cooling system equipment and piping for original design.

D + T + H + E Stresses remain within code allowable limits. ASME Section III, Class C [pumps]; ASME Section III, Class C and TEMA C, shell side; ASME Section VIII; TEMA R [LPCI heat exchanger].

D + T + H + E′ Primary stresses are within the stress criteria of ASME Section III, Class A. The secondary stresses, and primary plus secondary stresses are examined on a rational basis taking into account elastic and plastic strains. These strains are limited to preclude failure by deformation which would compromise any of the engineered safeguards or prevent safe shutdown of the reactor.

3.9.3.1.2.3 Evaluation

For the evaluation of equipment loads for original design, six components of loading (Fx, Fy, Fz, Mx, My, Mz) were derived in the same manner as the piping restraint loads. These loads were then given to the equipment designer for review to ensure that the loads could be safely sustained.

For equipment subject to Mark I hydrodynamic loads (see Section 6.2 for a discussion of the nature and origin of these loads), equipment nozzle connections were modeled as anchors for computer analysis of the attached piping. Large stiffness values were specified in the models at the anchor locations. Stresses at equipment nozzles were computed using the governing load combinations for the attached piping.

In general, the equipment nozzles and equipment casings were considered acceptable if the attached piping at the nozzles met the acceptance criteria for the piping. Additionally, the loads on equipment nozzles were evaluated as follows:

1. The pipe stress due to loads defined in NUREG-0661[5] for load combinations described in Table 3.9-5 at the equipment nozzle meets:

   A. The 10% rule of Section 6.2.b of the “Mark I Containment Program Structural Acceptance Criteria Plant Unique Analysis Applications Guide” (PUAAG)[6], or

   B. The pipe stress at the equipment nozzle obtained from the original design.

2. For equipment where a SQUG type evaluation was performed, the equipment anchorage was evaluated considering the piping reaction loads on the nozzles.

Check valves and manual valves, which are subject to Mark I hydrodynamic loads, were modeled in the piping analysis as piping elements, with increased stiffness and masses to represent the properties of the valve body. Lumped mass models were included in the piping analysis to represent valves with actuators, with the valve mass lumped at the center of gravity. For these valves, the stiffness and mass of the valve body and stem were considered, along with the eccentricity of the valve operator. Stresses were computed at the weakest sections of the yoke for each dynamic loading given in Table 3.9-5. The stresses in the valve body and the actuator components do not exceed yield stress.

3.9.3.1.3 Piping

The following subsections describe the acceptance criteria, loading conditions, and evaluations applicable to piping systems at Dresden Station. This information has been organized under five distinct phases of the design process:

A. Original design,

B. 79-14 program,

C. Mark I program, and

D. Replacement projects.

E. Extended Power Uprate program

3.9.3.1.3.1 Original Design

This subsection describes the criteria and methods applied in the original design of piping. It should be noted that subsequent evaluations, performed under the 79-14 Program, Mark I Program, replacement projects, or for Extended Power Uprate Program have superseded the original evaluations for many of the safety Class I piping systems. For piping not reanalyzed under one of these programs or projects, the original design basis remains applicable.

3.9.3.1.3.1.1 Acceptance Criteria

Piping at Dresden was originally designed in accordance with the USAS B31.1 Code, 1967 Edition. In addition, all piping from the reactor vessel up to the first isolation or stop valve was under the jurisdiction of ASME Section I, 1965 Edition, Winter 1966 Addenda. Nuclear code cases N-1, N-2, N-3, N-4, N-7, N-8, N-9, N-10, and N-11 were also used. Design criteria are discussed further in the following subsections.

3.9.3.1.3.1.2 Evaluation

There were three methods used in the original seismic analysis of piping systems (including hangers, snubbers, etc.):

1. Dynamic analysis by mode superposition using floor response spectra;

2. Static analysis using conservative static coefficients; and

3. Static analysis using lateral deflection and force evaluation curves.

A brief description of how each method has been applied and piping systems on which they were used follows.

Method 1, mode superposition, was used on the main steam and feedwater lines inside the drywell and the recirculation system lines.

Method 2, static coefficients, was used in the analysis of those Class I piping systems 10 inches in diameter or larger.

Method 3, lateral deflection, was used primarily for Class I instrumentation piping and for making evaluation studies for installed piping systems to determine the capability of meeting Class I requirements.

Table 3.9-6 indicates the analysis techniques used for several systems. Detailed descriptions of these methods are provided in Section 3.7.

The results of the original stress analysis of various systems that were statically analyzed are provided in Table 3.9-7. All stress analyses of piping and components included appropriate thermal loads.

That portion of the main steam line within the drywell and out to the first anchor beyond the second isolation valve was dynamically analyzed by mode superposition using a floor response spectrum. The remaining portion of the line was designed statically.

3.9.3.1.3.2 79-14 Program

An extensive program was implemented to fulfill the requirements of NRC IE Bulletin 79-14.[7] The 79-14 program involved documenting a verification that the design documents (e.g., design specifications and drawings) used as input to the seismic analysis of safety-related piping systems reflected the final as-built condition. This was deemed necessary because changes in the condition or configuration of the plant piping systems may have occurred during construction or as a result of repairs or other modifications. The bulletin required the following actions:

A. Walkdown safety-related piping which is 2½ inches or larger;

B. Evaluate and reconcile nonconformances;

C. Evaluate operability;

D. Reconcile as-built versus as-designed; and

E. Restore piping system to Final Safety Analysis Report (FSAR) limits.

Under these requirements all affected piping systems were reanalyzed using original design criteria.[8] The piping analysis considered the effects of pressure, gravity, thermal expansion, seismic OBE, and seismic SSE loadings. Piping stresses were evaluated in accordance with ANSI B31.1 1967 and requirements specified in the FSAR.

The final 79-14 piping evaluation was performed using two methods of analysis. In the first, large bore systems (10 inches or larger), which were originally analyzed by computer, were reanalyzed by computer. In the second, smaller diameter piping systems (2½ inches - 8 inches) were analyzed by a combination of "cookbook" (Blume criteria) methods and computer analysis. For these systems, the original design was based upon the cookbook method. The Blume criteria constitute an accepted industry method of qualifying smaller diameter piping systems.

For the first method, documentation existed and a detailed evaluation of as-built to as-designed loads was made by comparing computer-generated loads. Necessary modifications were then made.

For the second method, only the criteria existed as documentation. The as-built condition was evaluated against the criteria used to develop the design. Where pipe support span violations occurred, a more detailed evaluation was performed. This involved using a computer analysis to evaluate the supports bounding the span. Support loads were compared to the typical Blume criteria loads. If the computer loads were less than the Blume loads, the support was considered qualified and no further action was taken. If the loads exceeded the Blume cookbook loads, modifications or new supports were added to satisfy FSAR criteria.

After reviewing the design results from the 79-14 program, all supports in the Blume-qualified category were evaluated against computer-calculated loads and modified, where necessary, to provide a uniform level of documentation for all supports originally in the 79-14 program.

Following the initial operability evaluations for 79-14, all systems and restraints were restored to original design limits.

3.9.3.1.3.3 Mark I Program

The primary containments for the Dresden Station Units 2 and 3 were designed, erected, pressure-tested, and N-stamped in accordance with ASME Section III, 1965 Edition with addenda up to and including Winter 1965. Subsequently, while performing large-scale testing for the Mark III containment system and in-plant testing for Mark I primary containment systems, new suppression chamber hydrodynamic loads were identified. The new loads are related to the postulated LOCA and safety relief valve (SRV) operation.

The new loads were identified by the NRC as a generic open item for utilities with Mark I containments. The Mark I Owners Group established a two-part program consisting of the Short-Term Program (STP) and the Long-Term Program (LTP). The STP, which was completed in 1976, demonstrated safety-to-failure margins of at least 2.0 to justify continued operation. The LTP included a submittal of the "Mark I Containment Program Load Definition Report" (LDR),[9] PUAAG,[6] and supporting reports on experimental and analytical tasks. The NRC reviewed the LTP generic documents and issued acceptance criteria to be used during the implementation of the Mark I plant unique analyses. The NRC acceptance criteria are described in Appendix A of NUREG-0661.[5]

The objectives of the LTP were to establish the final design loads and load combinations, and to verify that existing or modified containment systems are capable of withstanding these loads with acceptable design margins. To meet the objectives of the LTP, CECo implemented a containment study program that provided analysis, design, and modification, if required.

The primary containments and the nuclear steam supply systems (NSSSs) are identical for Dresden Units 2 and 3. Differences between Dresden Units 2 and 3 exist primarily in the torus attached piping (TAP) systems and their corresponding branch connections.

Section 6.2 provides a detailed description of the origin and nature of the various Mark I hydrodynamic loadings.

3.9.3.1.3.3.1 Acceptance Criteria

Section 4.0 of NUREG-0661[5] presents the NRC evaluation of the generic structural and mechanical acceptance criteria and of the general analysis techniques proposed by the Mark I Owners Group for use in the plant unique analyses.

The structures affected by Mark I loads were categorized according to their functions to assign the appropriate service limits. The general components of a Mark I suppression chamber have been classified in accordance with Section III of the ASME Code as specified in NUREG-0661.[5]

The piping systems affected by the Mark I loads were the torus attached piping and the safety relief valve discharge lines (SRVDLs), including main steam (MS) piping.

The criteria used in the plant unique analyses to evaluate the acceptability of the existing Mark I containment designs or to provide the basis for any plant modifications follow Section III of the ASME Code through the Summer 1977 Addenda, except for the MS and SRVDL, which follow the rules contained in USAS B31.1-1967.

The service limits are defined in terms of the Winter 1976 Addenda of the ASME Code, which introduced levels A, B, C, and D. The selection of specific service limits for each load combination was dependent on the functional requirements of the component analyzed and the nature of the applied load.

Safety Relief Valve Discharge Lines Inside Wetwell

Acceptance criteria for the stress analysis of the wetwell SRVDL piping are based on the PUAAG.[6] Stress allowables are based on the applicable ASME Code subsections.

The wetwell SRVDL piping and the T-quencher discharge device are classified as ASME Code Class 3 for analysis purposes. Acceptance criteria are therefore based on the requirements of ASME Section III, Subsection ND, and are summarized in Table 3.9-8.

The SRVDL piping within the limits of reinforcement normal to the vent line penetration (both above and below the vent shell) is classified as a Class MC component for analysis purposes and is addressed in Section 3.8.

Main Steam and Safety Relief Valve Discharge Lines Inside Drywell

The acceptance criteria for the MS and SRVDL piping follows the rules contained in the USAS B31.1-1967. It should be noted that the B31.1 Code made no provision for emergency condition loadings. The combination of pressure, dead weight, and SSE inertia loads was used to calculate primary stresses in Equation 9 of the B31.7 Code. The calculated primary stresses were compared to 1.8 times the hot allowable stress. This represents a 50% increase over the B31.1 allowed primary stresses of 1.2 times the hot allowable stress for loads of short duration. The 50% increase is consistent with Code Case 70 of B31.7 for emergency conditions.

Torus Attached Piping Systems

The acceptance criteria defined in the NUREG-0661[5] upon which the Dresden TAP analysis is based follow the rules for Class 2 piping contained in ASME Section III, Division 1 up to and including the 1977 Summer Addenda. The corresponding service level limits and allowable stresses are also consistent with the requirements of the ASME Code and NUREG-0661.[5] The TAP is analyzed in accordance with the requirements for Class 2 piping systems contained in Subsection NC of the ASME Code. Table 3.9-5 lists the applicable ASME Code equations and stress limits for each of the governing piping load combinations.

3.9.3.1.3.3.2 Loading Conditions

Safety Relief Valve Discharge Lines Inside Wetwell

The wetwell SRVDL piping is analyzed as ASME Code Class 3, except for the portion of piping within the limits of reinforcement at the vent-shell penetration. This small portion of pipe is classified as Class MC and is discussed in Section 3.8. For the Class 3 SRVDL piping (including the T-quencher) the governing load combinations are shown in Table 3.9-10. The appropriate service level and applicable ASME Code equation are also identified for each combination.

Main Steam and Safety Relief Valve Discharge Lines Inside Drywell

The loads for which the MS and SRVDL piping and supports inside the drywell and the vent line are designed are consistent with the original loads, except that the SRVDL piping has been upgraded seismically as recommended by the Mark I Owners Group.

The loads acting on the MS and SRVDL piping inside the drywell are categorized as follows:

A. Pressure (P_o = maximum operating pressure; P = design pressure),

B. Dead weight (DW),

C. Seismic:

   1. Operating basis earthquake inertia (OBEI)

   2. Operating basis earthquake displacement (OBED)

   3. Safe shutdown earthquake inertia (SSEI)

   4. Safe shutdown earthquake displacement (SSED)

D. Temperature (TE = thermal expansion and anchor motion),

E. Safety relief valve discharge (SRVD), and

F. Safety valve discharge (SVD).

Loads in Categories A - F were considered in the original design of the MS lines, but the analytical methods and modeling techniques were much simpler reflecting the state-of-the-art during the original design. Seismic loads were not considered in the original design of the SRVDL piping since the piping is not safety-related. The latest analysis, however, does consider seismic loads for the SRVDL piping.

The evaluation of MS and SRVDL piping inside the drywell considered the following load combinations and allowable stress values (using nomenclature described earlier):

Primary stresses

$P + DW \leq 1.0\ S_h$

$P_o + DW + [OBEI^2 + SRVD^2 + SVD^2]^{\frac{1}{2}} \leq 1.2\ S_h$

$P_o + DW + [SSEI^2 + SRVD^2 + SVD^2]^{\frac{1}{2}} \leq 1.8\ S_h$

Secondary stresses plus pressure and dead weight

$TE + OBED \leq S_A$

$TE + SSED \leq S_A$

$TE + OBED + P + DW \leq S_A + S_h$

$TE + SSED + P + DW \leq S_A + S_h$

where

S_h = Basic material allowable stress at maximum (hot) temperature from the Allowable Stress Tables of B31.1-1967, Appendix A.

S_A = $f\,(1.25\ S_c + 0.25\ S_h)$

S_c = Basic material allowable stress at minimum (cold) temperature from B31.1 Allowable Stress Tables

f = Stress range reduction factor for cyclic conditions for total number N of full temperature cycles over total number of years during which system is expected to be in operation ($f = 1.0$ for $N < 7000$).

Torus Attached Piping Systems

The loads acting on the TAP are categorized as follows:

A. Dead weight loads,

B. Seismic loads,

C. Pressure and temperature loads,

D. Operating load,

E. Static torus displacement loads,

F. Safety relief valve discharge loads,

G. Vent-clearing loads,

H. Pool swell loads,

I. Condensation oscillation loads,

J. Chugging loads, and

K. Torus motion loads.

The governing load combinations for the TAP are presented in Table 3.9-5.

The appropriate ASME Code equations for the TAP are also provided in the governing load combination table.

3.9.3.1.3.3.3 Evaluations

The general structural analysis techniques proposed by the Mark I Owners Group were utilized with sufficient detail to account for all significant structural response modes. These techniques are consistent with the methods used to develop the loading functions defined in the LDR.[9] For those loads considered in the original design but not redefined by the LDR, either the results of the original analysis were used or a new analysis was performed. New analyses were based on the methods employed in the original plant design.

The damping values used in the analysis of dynamic loading events are those specified in Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," which is in accordance with NUREG-0661.[5]

Safety Relief Valve Discharge Lines Inside Wetwell

The evaluation of the SRVDL piping inside the wetwell includes the performance of a structural analysis of T-quenchers and their supports for the SRV discharge-related loads and LOCA-related loads to verify that their design is adequate. Rigorous analytical techniques were used in this evaluation by means of detailed models and refined methods to compute the dynamic response of the T-quencher. The loads were input as static, quasi-static, or dynamic loads, and the interaction between the torus and SRVDL supports due to the loads was considered.

Table 3.9-11 summarizes the maximum Class 3 wetwell SRVDL piping and T-quencher stresses. The maximum calculated stress and Code allowable stresses are given for each applicable code equation for each service level.

Main Steam and Safety Relief Valve Discharge Lines Inside Drywell

For the evaluation of MS and SRVDL inside the drywell, several rigorous analytical techniques have been used to determine the structural response of the piping.

Dynamic analysis techniques were used to predict system response under seismic inertia and SRV loads. The dynamic analysis techniques consisted of either response-spectra or time-history methods, depending upon the input loading. Static analysis techniques have been utilized for seismic anchor motion loadings and the remaining load cases defined in Section 3.9.3.1.3.2.2.

For the drywell SRVDL piping, the B31.1-1967 Code stress allowables have been satisfied for all applicable stress equations.

Torus Attached Piping Systems

The TAP evaluation included performing a structural analysis of the TAP systems and suppression chamber penetrations for the effects of LOCA-related and SRV discharge-related loads to verify that the design of the torus attached piping and suppression chamber penetrations is adequate. Rigorous analytical techniques were used in this evaluation, utilizing detailed analytical models and refined methods for computing the dynamic response of the torus attached piping and penetrations. This analysis included consideration of the interaction effects of each piping system and the suppression chamber.

The results of the structural analysis for each load were used to evaluate load combinations for the piping, piping supports, equipment, and penetrations in accordance with NUREG-0661[5] and the PUAAG.[6] The analysis results were compared with the acceptance limits specified by the PUAAG and the applicable sections of the ASME Code for Class 2 piping.

Section 4.3.3.2 of NUREG-0661[5] requires that a fatigue evaluation of the SRV and TAP be performed for all loading conditions except pool swell.

The Mark I Owners Group prepared and submitted a generic fatigue evaluation report[10] to the NRC in late 1982. The report addressed fatigue on a generic basis using actual piping analysis results from essentially all Mark I plants. The resulting cumulative usage factors are below 0.5, demonstrating that further plant unique fatigue evaluations are not warranted. Therefore, the Dresden Unit 2 and 3 TAP is qualified based on this generic evaluation.

The maximum piping stresses resulting from governing load combinations for highly stressed locations on each large bore TAP line and for small diameter torus internal lines are presented in Tables 3.9-12 and 3.9-13, for Units 2 and 3, respectively. The maximum stresses for each service level are listed along with the associated Code equations and allowable stress values. The secondary stresses reported in this table do not include the effects of the higher long term post-LOCA temperature associated with the Extended Power Uprate (see Section 3.9.3.1.3.5).

In summary, the results show that the design of the large bore TAP systems are adequate for the loads, load combinations, and acceptance criteria limits specified in NUREG-0661[5] and the PUAAG.[6]

The Piping Configuration Verification Program (PCVP) was initiated to identify and document inconsistencies between the as-built configuration and the analysis documentation for Mark I piping systems. Any inconsistencies determined to be discrepancies were resolved by either changing the field condition to match the analysis documentation or reanalyzing the piping and/or supports to demonstrate acceptability. Where a new seismic analysis was performed, the response spectra or static coefficient method was used. This seismic response spectra analysis employed 1/2% damping. All of the discrepancies were resolved and FSAR compliance was demonstrated.

3.9.3.1.3.4 Replacement Projects

This subsection describes the criteria and methods applied to the evaluation of piping which has replaced existing piping at the Dresden Station.

Dresden Unit 3 Recirculation Pipe Replacement

The stress analysis for Class I piping covered by the scope of the recirculation pipe replacement project was performed in accordance with ASME Section III, Subsection NB, 1980 Edition, including the Summer 1982 Addenda. To reconcile the Section III analysis with the original design code, the ASME stresses calculated were compared to the original licensing allowables, with recalculation using original rules employed only when necessary. A comparison of the maximum stresses showed that all piping met the original licensing criteria.

Safety evaluations were performed for each modification package associated with the recirculation pipe replacement project. A systematic evaluation of each design change was performed in accordance with the CECo Safety Evaluation Guidelines. No case was identified which increased the probability of occurrence or the consequences of a previously evaluated event, created an event of a different type than previously evaluated, or reduced the margin of safety as defined in the basis for any technical specification identified.

3.9.3.1.3.5 Extended Power Uprate (EPU)

Operation at the Extended Power Uprate (EPU) conditions increased piping stresses caused by higher operating temperatures, pressures and flow rates. Additionally, piping components (i.e., pipe supports, equipment nozzles, etc.) are subjected to increased loadings due to the EPU. The following subsections discuss the Torus Attached Piping (TAP) and Main Steam (MS) Piping which required reanalysis as a result of EPU. The EPU analysis approach is described in Reference 39 and accepted by the NRC in Reference 40.

3.9.3.1.3.5.1 Torus Attached Piping Systems

The piping system evaluations for power uprate were performed by determining “change factors” for the changes in thermal, pressure, flow rate, and total design load conditions.

This method is based on determining a “change factor” by conservatively comparing the ratio of power uprate temperature, pressure and flow conditions to the corresponding pre-uprate conditions.

Where the “change factor” is less than 1.0, the pre-EPU (existing) conditions and no further review is performed.

For minor changes resulting in a “change factor” between 1.0 and 1.05 (or 5%), the increased was considered acceptable since the small increase is offset by conservatism inherent in the analytical methods used to calculate the existing stresses and loads. The conservatism include, but not limited to, the industry practice of enveloping multiple operating conditions and modeling pipe supports without consideration of gaps between piping and multiple operating conditions and modeling pipe supports without consideration of gaps between piping and supports. Pressure effects are considered in conjunction with other loading conditions which are unchanged by the supports. Pressure effects are considered in conjunction with other loading conditions which are unchanged by the EPU (e.g., weight, seismic) thus the overall effect of the pressure change factor is reduced. Therefore for "change factors" between 1.0 and 1.05, the existing stress and load values were considered to be unchanged and remain within allowable limits.

For "change factors" greater than 1.05, simple and conservative evaluations were performed to address the specific increase in stress and load values. Where the simple evaluation yielded a resultant stress ratio (i.e., calculated/allowable) that was less than 1.0, the resultant stress remains acceptable. For those conditions where the "change factor" and resultant stress ratio is greater than 1.05, the calculations were revised and/or piping support modifications were performed to bring the stress at EPU conditions within allowable limits.

The thermal "change factor" was based on the ratio of the thermal power uprate to pre-thermal power uprate operating temperature. That is, the thermal change factor is (T_{uprate} 70^{o}F)/($T_{pre\text{-}uprate}$ 70^{o}F). Using this method for the thermal change factor, evaluations resulted in a bounding evaluation of the thermal, impact on piping stresses and loads.

Similarly, the pressure "change factor" was determined by the $P_{uprate}/P_{pre\text{-}uprate}$ ratio and the flow rate "change factor" was determined by the $Flow_{uprate}/Flow_{pre\text{-}uprate}$ ratio. The total design load change factor is the total combined load associated with EPU conditions divided by the allowable design load, and was determined by the following formula:

$[DW + Pressure_{uprate} + Thermal_{uprate} + TransientLoad_{uprate} + Seismic] / Design\ Load_{analyzed.}$

Thermal changes were found to be the most significant, primarily for systems using the torus as a water suction source during long term post-LOCA conditions. No changes to the suppression pool loads (pool swell, condensation oscillation, chugging and SRV discharge) will result from the EPU (previous load definitions were determined to be bounding). Pressure changes were typically found to be negligible and were unchanged for most systems. There is a slight increase in predicted DBA pressures inside the torus, however most torus attached piping systems and components were previously analyzed for the maximum IBA pressures which bound even the new DBA pressures.

All piping systems subject to changes in temperature, pressure or flow were screened to determine the impact on the piping and piping components (i.e. supports, penetrations, equipment nozzles, etc.). Piping systems subjected to minor operating condition increases due to EPU were excluded from a detailed evaluation, as follows:

1. Thermal load increases of up to 5% (change factors between 1.00 and 1.05), were considered acceptable since these increases are offset by conservatism in analytical methods used to calculate the existing stresses and loads. Conservatisms include the enveloping of multiple thermal operating conditions and not considering pipe support gaps in the thermal analyses.

2. Pressure load increases up to 5% were considered acceptable due to margins in piping wall thickness.

3. Transient load increases up to 5% resulting from EPU related fluid flow rate changes were considered acceptable due to conservatism in load combinations (transient loads are combined with other conservative loads such as thermal and seismic).

4. Total design load increase of 5% were considered minor and acceptable by engineering judgment due to inherent conservatism in piping analysis methodology, as previously described.

The total design load criteria was not used for drywell steel, corner room steel, and/or flued head anchors without reviewing their qualification documentation to ensure that similar reasoning to this criteria had not been previously invoked for other load increases.

If the increases described above exceeded 5%, the analyzed margin between design load and the allowable load prior to uprate was used to justify the increases for uprate conditions (e.g., if the load component was considered acceptable).

If the load increase on a piping component was greater than the calculated available margin, then a detailed evaluation of the component was performed to evaluate the adequacy of the component for EPU conditions. If the detailed evaluation could not justify the increased EPU loads in accordance with the previously defined acceptance criteria, modification was designed for that component such that the modified component would meet that acceptance criteria.

All piping systems and piping components with changes in temperature, pressure or flow rate were screened for impact by EPU. If the change ratios for the piping systems were less than 1.05, the whole system, including the piping components (i.e., supports, penetrations, equipment nozzles, etc.), was considered acceptable. If any of the change ratios exceeded 5%, each piping component was reviewed independently.

The evaluation methodology used to assess impact of the long term post-LOCA temperature increase on torus water piping system components (piping components in systems pumping or exposed to torus water) is provided in more detail below. Pipe supports are discussed separately in Section 3.9.3.3.4.

<u>Pipe Stress</u>

The basic approach for the pipe stress evaluation was to factor up the existing Level A (ASME Eq. 10) pipe stresses by the thermal change ratio. The revised stress was then compared to the allowable pipe stress associated with the post-LOCA thermal condition. The application of ASME and B31.1 for the EPU pipe stress evaluations is consistent with the existing design and licensing basis.

The allowable pipe stress for post-LOCA conditions was based on the code of record for each piping system for one time secondary loads (e.g., single non-repeated anchor movement). For ASME piping, the allowable stress was taken as 3 S_h (equal to 45,000 psi for A-106 Gr. B piping). For B31.1 piping, the allowable was taken as 1.8 S_h (equal to 27,000 psi for A-106 Gr. B piping). For B31.1 piping, as an alternate, an allowable of 3 S_h minus the actual DW and Pressure stresses is allowed by Section 102.3.2d of B31.1.

<u>Displacements at Interferences</u>

Some piping models have displacement checks at certain locations where there may be interferences with nearby structures (ie., slab or wall penetrations, nearby plant equipment, etc.). The locations that were impacted were evaluated to make sure the revised thermal displacements did not result in damaging contact with these interferences.

Flanges

Some of the piping models have in-line flanges that have been evaluated for piping moments. These moments in the piping system are affected by the increase in temperature for these lines. For the affected flanges, revised thermal moments were calculated for the flanged joints and compared to the previously calculated allowables.

Valves

The stresses in valve bodies were already enveloped by the stresses reported for the piping, so these valves were covered in the piping stress evaluation. For valves with extended operators (i.e., MOVs), the stresses are a function of the valve acceleration and are not affected by increased thermal loads.

Containment Penetrations

Some of the piping systems penetrate the primary containment boundary (i.e., the torus or the drywell). At these penetrations, the containment shell is evaluated for the local stresses in the vicinity of the penetration due to the reactions at the penetration. The total stress in the containment shell is a combination of the local stresses due to the reaction loads from the piping, combined with the global shell stresses due to conditions inside containment. The revised post-LOCA forces and moments were calculated for all six degrees of freedom and compared to the previously qualified loads. In some cases, revised combined stresses in the containment were calculated and compared to the allowable stresses.

Equipment Nozzles

The existing design basis for piping loads on equipment is that the nozzles and casings are considered acceptable if the attached piping stress at the nozzles meets the Code requirements for the piping. For certain equipment at SQUG type evaluation had previously been performed, where the equipment anchorage was evaluated considering the piping reaction loads. This approach was extended to cover non-SQUG equipment such as the CS pumps. The affected equipment included the LPCI and CS pumps and the LPCI heat Exchangers (the RHR and CS pumps and the RHR Heat Exchangers at Quad Cities). If the loads on this equipment increased by more than 5%, the equipment anchorage was re-evaluated. In some cases, it was concluded that certain equipment is bounded by other similar equipment (i.e., identical equipment with higher nozzle loads).

Reactor Nozzles

Some of the piping systems tie directly into Reactor Nozzles. At these nozzles, an evaluation was performed to determine the impact of the nozzle reaction loads on the Reactor Pressure Vessel. The revised stresses in the RPV nozzles were calculated for EPU conditions and compared to the previously calculated allowable stresses. The nozzles were also previously evaluated for fatigue considerations. Sine the EPU post-LOCA thermal condition is a one-time event, its impact on the fatigue analysis of the nozzle was determined to be negligible.

All large bore (> 4" NPS) torus water piping systems were evaluated for the effected of increased operating temperatures and pressures. The scope of the small bore torus water piping systems that were evaluated for EPU conditions included small bore piping directly attached to the torus and small bore piping connected to large bore (greater than 4 inch) piping that is directly attached to the torus. Also, small bore lines attached to large bore lines that are not torus attached but transmit torus water during the long term post-LOCA mode were evaluated.

3.9.3.1.3.5.2 Main Steam Piping System

The EPU does not affect design basis loads for the MS System. However, the MS System flow increased by approximately 20% for EPU. A review of the increase in flow related loads associated with EPU indicated that piping loads due to the dynamic effects of the Turbine Stop Valve (TSV) fast closure (which is not included in the design basis loads) results in significant loads for the MS piping and supports.

10CFR50, Appendix A, General Design Criteria #15 requires that the Reactor Coolant System (RCS) and connected piping be "designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences." Dresden being a pre-General Design Criteria Plant (GDC) plant was designed to USAS B3.1-1967 that required consideration of the most severe condition of coincident pressure, temperature and loading. The plant transient dynamic load for safety valve opening was included in the Dresden and Quad Cities design requirements for the B31.1 piping. The Standard Review Plan (SRP), issued in April 1984, Revision 3 to Section 10.3, "Main Steam Supply System," stated that main steam systems must be designed to withstand the effects of rapid valve closure. However Section V, Implementation," of Revision 3 of SRP 10.3 states that currently licensed plants (i.e., prior to 1984) do not need to adhere to this requirement. Thus, neither the GDC nr SRP requirements relative to consideration of transient dynamic loads due to TSV closure apply to Dresden. Furthermore, a review of the Dresden license basis yielded no specific licensing commitments or statements for the design of the main steam piping relative to the turbine stop valve (TSV) closure. Therefore, the current licensing basis clearly does not require analysis of the loads due to turbine stop valve closure.

Even though consideration of TSV loads was determined to be beyond the current licensing basis, it is prudent to address these loads. The EPU evaluation approach for the TSV loads is based on an acceptance criteria for TSV loads which are less restrictive than the current application of the ASME and AISC codes, but which ensure that no permanent deformation of the piping, piping supports or supporting structural steel will occur as a result of the event.

Under EPU conditions the TSV closure loads were analyzed and modifications implemented to ensure that the TSV closure does not result in main steam system piping system failure. Since, thee is no current licensing basis for the acceptance criteria for the TSV loads, load combinations and acceptance criteria for the TSV loads were developed for the EPU evaluations. The main steam piping, pipe supports and supporting structures were evaluated for the TSV fluid transient loads in combination with pressure, deadweight, and thermal loads, as appropriate. Since a seismic event may cause a unit trip and a TSV closure, the TSV transient loads were also considered concurrent with applicable seismic loads. Since the TSV closure event is considered beyond the current licensing basis and the purpose is to demonstrate pressure boundary integrity of the piping, a TSV event was considered to occur concurrently with the SSE event only. The evaluation method is to demonstrate pressure boundary integrity of the piping and associated member/component evaluated to ensure that no gross deformation or integrity failure occurs. Also, due to the tie relationships between the significant loads resulting from TSV, SRV discharge and pipe break events (i.e, LOCA), no combination of these loads is required.

To demonstrate piping pressure boundary integrity subsequent to a TSC closure event, the piping was evaluated for the following additional loading combinations (LC):

Piping

LC1 Dead Load + Pressure + TSV Loads
LC2 Dead Load + Pressure + $[(\text{TSV Loads})^2 + (\text{SSE Loads})^2]^{1/2}$

The TSV fluid transient loads were generated utilizing the representative and bounding effective closing time for the TSV. For dynamic load combinations, oscillator (piping system) damping were considered to be 2% when considering TSV alone (LC1) and 3% when combined with seismic (LC 2) in accordance with guidance contained in Reg. Guide 1.61. Seismic damping values are based on the values stipulated in the UFSAR.

The following acceptance criteria were used for the load combinations listed above.

PIPING	
DW + P = TR*	ASME Level C
DW + P + $(SSE^2 + TR^{2)})^{1/2}$	ASME Level D

*TR = TSV loads

The main steam piping meets the above acceptance criteria. Also, all other current design and license basis criteria are met for the EPU conditions.

Main steam piping supports and support steel are discussed in section 3.9.3.3.4.

3.9.3.1.3.6 Systems Containing Cast Iron Components

In the early 1980’s, Dresden Unit 2 was part of the NRC's Systematic Evaluation Program (SEP). Under SEP Topic III-1, "Quality Group Classification of Components and Systems”, the NRC reviewed the classification of structures, systems, and components of plants designed and constructed from the late 1950's to late 1960's to current appropriate classifications, codes, and standards for seismic and quality groups. Dresden Unit 2's safety-related systems, which were designed to the USA Standard (USAS) Code for Pressure Piping B31.1-1967 "Power Piping", were evaluated against the fracture toughness requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III, 1977 Edition as supplemented by the Summer 1978 Addenda. Of particular interest are the significant changes in fracture toughness requirements that occurred in 1972.

During the SEP, Dresden failed to identify that the DGCW and CCSW systems contained cast iron valves and that the CCSW system contained cast iron pump casings. The NRC was notified of this error in a letter to the USNRC dated March 31, 2000 (Letter ID PSLTR #00-0066). Because the use of cast iron in safety-related systems was not evaluated at the time of the SEP, cast iron was not addressed in the NRC Safety Evaluations regarding SEP Topic III-1. Cast iron has lower ductility and fracture toughness than other materials typically used in safety-related piping systems. Although it is an acceptable material in the USAS B31.1-1967 code, there are no material specifications for cast iron that are acceptable in the 1977 ASME Section III Code, which formed the basis of the evaluation criteria of SEP Topic III-1. To accommodate the lower ductility and fracture toughness, cast iron valve bodies and pump casings in the DGCW and CCSW systems meet the following acceptance criteria.

3.9.3.1.3.6.1 Acceptance Criteria

General

1. The CCSW and the DGCW system piping are designated to USAS B31.1, 1967 Edition.
2. All cast iron valves and the CCSW pumps meet the manufacturers sped pressure and temperature service ratings.
3. The design temperature is not higher than 400°F, and not lower than 32°F.
4. The material of the cast iron components meets ASTM Specification A 126 or A 48.

5. All cast iron valves are manually operated, and meet the ANSI B16.10 Standard.
6. The cast iron components are not used with flammable, combustible, or toxic fluids.
7. The cast iron components are not subject to water hammer or rapid thermal or pressure transients. Mechanical impact such as hammering to disassemble flanged joints is not permitted.
8. There are no pipe supports at the cast iron valves. Displacements of the cast iron components are limited such that they do not contact other components in a seismic event.
9. All cast iron components are connected to piping of wass thickness 5/8" or smaller.
10. Welding to cast iron components is not permitted.

Valve Nozzle Stresses

The valve nozzle stresses caused by the connecting piping end moments meet the following stress limits (see Notes, below):

Sustained Loads:
Pressure Stress + Dead Load Stress $\leq$ 1.0 S_{CI}

Level B Loads:
Pressure Stress + Dead Load Stress + OBE Stress $\leq$ 1.2 S_{CI}

Level D Loads:
Pressure Stress + Dead Load Stress + DBE Stress $\leq$ 2.4 S_{CI}

Thermal Expansion plus Sustained Loads:
Pressure Stress + Dead Load Stress + Thermal Expansion Stress < 2.5 S_{CI}

Notes:
Definition of the stress terms is per ASME B31.1, 1989 Edition.
S_{CI} = 6.0 ksi per USAS B31.1-1967, Appendix A.
The Level D allowable is per Code Case 1606-1.
The thermal expansion plus sustained load allowable assumes fewer than 7000 thermal transient cycles. For cast iron, $S_{CI} = S_h$.

Pump Nozzle Stresses

The pump nozzle stresses caused by the connecting piping forces and moments meet the following stress limits (see Notes, below):

Level B Loads:
General Membrane Stress $\leq$ 1.1 S_{CI}
General or Local Membrane Stress + Bending Stress 1.65 S_{CI}

Level D Loads:
General Membrane Stress $\leq$ 2.0 S_{CI}
General or Local Membrane Stress + Bending Stress $\leq$ 2.4 S_{CI}

Notes:
Definition of stress terms is per ASME Section III, ND-3416, 1989 Edition.
Level B general membrane stress is due to pressure, and axial forces from dead load, OBE, and thermal expansion.
Level B bending stress is due to dead load, OBE, and thermal expansion. No stress intensification factors are applied.

Level D general membrane stress is due to pressure, and axial forces from dead load and DBE.
Level D bending stress is due to dead load and DBE. No stress intensification factors are applied.
S_{CI} = 6.0 ksi.

3.9.3.2 Pressure Relief Devices

Discussion of reactor coolant pressure boundary pressure relief devices may be found in Section 5.2.2.

3.9.3.3 Component Supports

The following subsections discuss component pipe support design practices for the original Dresden design (Section 3.9.3.3.1), the 79-14 program (Section 3.9.3.3.2), and the Mark I program (Section 3.9.3.3.3), and the Extended Power Uprate (Section 3.9.3.3.4).

3.9.3.3.1 Original Design

Piping and supports were originally designed in accordance with the USAS B31.1 Code, 1967 Edition.

The materials used in the fabrication of hangers, anchors, and supports met the requirements of USAS B31.1 Code for Pressure Piping Section 6, and the Manufacturer's Standardization Society Standard Practice MSS-SP-58 for normal operating and seismic design conditions.

Stresses in the materials used in the fabrication of anchors, guides, and restraints designed to resist dynamic loads due to pipe rupture and design seismic loading were within 90% of the material yield stress for the pipe rupture design conditions.

All hot-wound helical spring coils used in spring devices furnished as a component part of all spring hangers and supports met the requirements of ASTM-A-125-65.

The anchor block which restrains the reactor feedwater and main steam piping was designed for the design seismic forces on the pipe plus various pipe rupture cases. The anchor was assumed to be a rigid structure and the piping was analyzed for static loads, thermal forces, seismic forces, and pipe rupture loads. After determining the magnitudes of the various combinations of loads, the anchor was then detail designed to withstand the maximum derived forces and moments. The seismic load input to the anchor as determined by the dynamic analysis of the piping within the drywell provided a relatively small portion of the total loads due to rupture cases. Engineering evaluations of the portion of the lines outside the drywell that were statically designed led to the conclusion that the inertia forces generated by this section of piping could not result in a significantly different magnitude of seismic inputs to the anchor than the piping within the drywell that was dynamically analyzed. Since the rupture forces are much greater than the inertia force and no component in the anchor is stressed beyond allowable, the anchor meets the requirements of rigidity and safety.

The following pipe rupture cases, which are the major design load contributors both from an axial and moment standpoint, were considered:

A. Axial load;

B. Moment due to pipe split;

C. Moment and axial load due to pipe split past a 90^{o} elbow; and

D. Moment, axial load, and torsion due to a pipe split past two 90^{o} elbows.

Primary recirculation system support structures were designed to withstand the flow-induced vibration forces that may occur under abnormal operating conditions.

Pipe support baseplates for Dresden Units 2 and 3 were designed based upon rigid plate theory. The rigid plate analysis procedure was compared with a flexible plate analysis procedure which resulted in the conclusion that adequate margin existed in the original design.

3.9.3.3.2 79-14 Program

The pipe support consists of a set of load-carrying structural elements and component standard hardware spanning the piping system and the building structure. The supports are evaluated for loadings exerted by the piping system for various operating conditions. They are qualified, modified, or designed to comply with the FSAR, applicable codes, and the support component manufacturer's data.

The qualification of existing pipe supports was performed in accordance with the original design criteria discussed in Section 3.9.3.3.1, the AISC Manual of Steel Construction (Sixth Edition) and MSS-SP58 (1967). Additional design bases, such as vendor data and other limitations not covered in the FSAR or the above-mentioned codes, were reconstructed based on the existing design drawings and the understanding of the industry standards at the time. New supports were designed in accordance with MSS-SP58 (1975), ANSI B31.1 (1977 Addenda through Summer 1979), and AISC Manual of Steel Construction (Seventh Edition). Standard support components were selected to conform with MSS-SP69 (1976). Existing elements of modified supports comply with the criteria of existing supports. New elements of modified supports comply with criteria of new supports. Attachments to supports utilizing integral pipe attachments are of material compatible with the pipe and conform to the piping jurisdictional code.

Visual weld inspection was in accordance with guidelines prepared by the Nuclear Construction Issues Group, NCIG-01, Revision 2, titled "Visual Weld Acceptance Criteria for Structural Welding at Nuclear Power Plants."

3.9.3.3.3 Mark I Program

Safety Relief Valve Discharge Lines Supports Inside Wetwell

Acceptance criteria for the stress analysis of the wetwell SRVDL piping are based on the PUAAG.[6] Stress allowables are based on ASME Section III, Subsection NF, for the wetwell SRVDL supports, which are classified as ASME Code Class 3 for analysis purposes.

The acceptance criteria for SRVDL wetwell supports are in accordance with the structural design specification and are consistent with the AISC "Specification for the Design, Fabrication and Erection of Structural Steel Buildings." All stresses due to normal and severe environmental loading conditions are within normal AISC allowable limits. All stresses due to extreme environmental and emergency loading conditions are within 1.6 times the AISC allowable, with no stress exceeding 0.95 times the ASTM minimum specified yield strength of the material. These criteria are more conservative than Section III of the ASME Code, which is required by the Mark I program structural acceptance criteria.

For the Class 3 SRVDL supports, the governing load combinations are shown in Table 3.9-14. The appropriate service level is identified for each combination.

Maximum stress values and the corresponding appropriate code allowable stresses of the critical components of the intermediate support, and T-quencher supports are listed in Table 3.9-15. All critical stress values are within the AISC Code requirements, and therefore meet the requirements of the ASME Section III, Subsection NF.

Safety Relief Valve Discharge Lines Supports Inside Drywell

The acceptance criteria for the SRVDL vent line supports are consistent with the AISC "Specification for the Design, Fabrication and Erection of Structural Steel Buildings." These criteria are more conservative than Section III, Subsection NF, Division 1 of the ASME Code, which is required by the Mark I Program Structural Acceptance Criteria.

The design of the auxiliary steel and floor support structure was based on the allowable stresses as given in the AISC. All stresses due to normal and severe environmental loading conditions were within the normal AISC allowable limits. All stresses due to extreme environmental and emergency loading conditions were within 1.6 times the AISC allowable limits, with no stress greater than 0.95 times the ASTM minimum yield stress of the material.

The MS and SRVDL supports have been designed for the following load combinations (subscripts indicate load type, as defined in Section 3.9.3.1.3.3.2):

Upset conditions

$F_{DW} + F_{TE} + [F_{OBEI}^2 + F_{OBED}^2 + F_{SRVD}^2 + F_{SVD}^2]^{1/2}$

Emergency conditions

$F_{DW} + F_{TE} + [F_{SSEI}^2 + F_{SSED}^2 + F_{SRVD}^2 + F_{SVD}^2]^{1/2}$

where:

F = force or moment due to a particular load

The maximum snubber reaction loads for the load combinations for the MS and SRVDL inside the drywell are within the appropriate allowables. The maximum resultant loads in the rigid struts are within the appropriate strut allowables. The auxiliary steel and floor support structure are within the allowable limits. The SRVDL guides and attachments in the vent are also within the allowable limits as shown in Table 3.9-16.

Torus Attached Piping Supports

Pipe supports for torus attached piping were evaluated using standard linear elastic structural analysis methods, hand calculations, or standard structural analysis computer programs. Table 3.9-17 presents the governing load combinations applicable to TAP supports. The resultant component forces and stresses were compared to their respective allowable values.

Standard component allowables for Levels B, C, and D service limits were supplied by the manufacturer. Allowables for structural members, base plates, and welds are defined in ASME Section III, Subsection NE or NF, up to and including the 1977 Summer Addenda and in NUREG-0661.[5]

Anchor bolt allowables are based on manufacturer's test data in accordance with IEB-79-02 requirements and ACI-349-80. Base plate flexibility and shear-tension interaction were considered in the anchor bolt evaluation.

Integral attachments were evaluated by adding the local stresses in the pipe from each support load combination to the corresponding pipe stress load combination listed in Table 3.9-5. Allowable stresses are given in Table 3.9-9. Local stresses are generally calculated using methods described in Welding Research Council Bulletin WRC-107 and in ASME Code Case N-318.

In summary, the design of the TAP supports is adequate for the loads, load combinations, and acceptance criteria limits specified in NUREG-0661[5] and substantiates the piping analysis results.

3.9.3.3.4 Extended Power Uprate Evaluation

As discussed in Section 3.9.3.1.3.5, operation under extended power uprate (EPU) conditions results in increased piping stresses. The effects on piping supports are discussed in the following subsections for Torus Attached Piping (TAP) supports and Main Steam (MS) piping supports.

3.9.3.3.4.1 Torus Attached Piping Supports

Rigid Pipe Supports

Rigid supports were categorized as those supports that rigidly support both static and dynamic loads and include rod hangers where applicable), struts, guides, and piping anchors, etc. The basic approach was to calculate a revised post-LOCA load combination of Dead Weight (DW) plus EPU Thermal (T) (Thermal Expansion plus Thermal Anchor Movement) plus Safe Shutdown Earthquake (SSE) plus EPU Torus Displacement (TD). This load combination was classified as a level D or Faulted load combination. Therefore, a revised Interaction Coefficient (I.C.) (actual stress divided by allowable stress) was calculated by multiplying the maximum I.C. in the existing calculation by the total design load change factor defined as the new post-LOCA load combination (DW+T+SSE+TD) divided by the largest (peak) qualified load. In addition, for supports subjected to frictions loads (i.e., guide supports), or supports with integral welded attachments, additional evaluations were performed.

Snubbers

Since snubbers do not resist thermal loads, the new EPU thermal conditions will not affect the snubber loads. The thermal displacement will increase however so there is a potential for a top out or bottom out condition associated with the increased thermal displacements from EPU. In the late 1980s, allowable cold setting ranges were determined for each snubber to ensure that sufficient travel was available such that the snubbers would not bottom- or top-out on their range during thermal expansion. Included in this range calculation was a minimum of ½ inch “cushion” provided on each end of the range. Therefore a minimum of ½ inch of travel is available to handle additional thermal expansion above and beyond the current design displacements. A generic evaluation was performed, which concluded that the increase in thermal displacements due to the EPU would not exceed the ½ inch available travel.

In addition, the increased displacement will cause an increase in the swing angle for snubbers and other pinned supports. A generic evaluation was performed, which concluded that the increased in swing angles due to EPU conditions is minor and will not impair the functionality of the pinned type supports.

Spring Hanger Supports

For each affected spring hanger, the increased vertical thermal displacement was compared to the available displacement to top-bottom-out conditions., If the additional displacement exceeded the available displacement by more than 5%, then a modification was issued to reset or replace the existing spring can. The increase/decrease in the spring hanger load due to movement change is considered negligible.

3.9.3.3.4.2 Main Steam Piping Supports

As discussed in Section 3.9.3.1.3.5, operation under EPU conditions results in significant loads for the Main Steam piping supports due to the dynamic effects of the Turbine Stop Valve (TSV) fast closure. Although TSV loads were not part of the original licensing basis, it is prudent to address these loads for EPU.

To demonstrate piping pressure boundary integrity subsequent to a TSV closure event, pipe supports and supporting structures were evaluated for the following additional loading combinations (LC):

Pipe Supports and Pipe Support Structures:

LC 3 Dead Load + Operating Thermal Loads + TSV Loads
LC 4 Dead Load + Operating Thermal Loads + $[(\text{TSV Loads})^2 + (\text{SSE Loads})^2]^{1/2}$

The TSV fluid transient loads were generated utilizing the representative and bounding effective closing time for the TSV. For dynamic load combinations, oscillator (piping system) damping were considered to be 2% when considering TSV alone and 3% when combined with seismic in accordance with guidance contained in Reg. Guide 1.61. Seismic damping values are based on the values stipulated in the UFSAR.

For evaluation of the supporting drywell steel, where supports from different main steam lines are attached to the same drywell steel, the TSC loads were combined by the SRSS method. This is due to the variation in actuation time, which results in the pressure wave for different MS lines being out-of-phase with the peak loads occurring at different times.

Design Criteria for Structural Steel and Pipe Support Evaluations

LC 3–Dead Load+Operating Thermal Loads+TSV Loads

The allowable stresses shall be limited to 1.33* Normal AISC Allowable stresses.

LC 4–Dead Load+Operating Thermal Loads+SSE Loads+TSV Loads

The following table summarizes the acceptance criteria for the load combinations listed above.

APPLICABLE TSV LOAD COMBINATIONS	ACCEPTANCE CRITERIA
STRUCTURAL & AUXILIARY STEEL	
DW+TH+TR*	NORMAL 1.33 x AISC Allowable
DW+TH+(SSE^{2}+TR^{2}) ½	FAULTED 1.60 x AISC Allowable <0.95*Fy**
EXPANSION ANCHOR BOLTS	
DW+TH+TR	SAFETY FACTOR = 4
DW+TH+(SSE^{2}+TR^{2}) ½	SAFETY FACTOR = 2
PIPE SUPPORT COMPONENTS	
DW+TH+TR	ASME LEVEL C
DW+TH+(SSE^{2}+TR^{2}) ½	ASME LEVEL D
PIPING	
DW+P+TR	ASME LEVEL C
DW+P+(SSE^{2}+TR^{2}) ½	ASME LEVEL D

*TR = TSC loads
**Plastic section modulus can be used to determine the section stresses but must meet ductility criteria.

Structural Steel Members

Stress	Design Limit
Bending	1.6* AISC allowable based on plastic section modulus with stresses not to exceed 0.95 *Fy. For this to be used, the section should satisfy the compact section criteria and lateral bracing requirements of the AISC Code. AISC LRFD Specification may be consulted to obtain further clarifications.
Axial	1.6* AISC allowable not <0.95*Fy
Shear	0.95*Fy/(3)$^{1/2}$=0.548 *Fy

Plate Materials

Stress	Design Limit
Bending about Weak Axis	0.95 *Fy based on plastic section modulus
Bending about Strong Axis	0.95 *Fy based on plastic section modulus or 1.0* Fcr based on elastic section modulus, whichever is smaller.
Shear	0.95*Fy/(3)$^{1/2}$=0.548 *Fy

Bolts 1.60* AISC Allowables

Welds 1.60* AISC Allowables. The base metal shear for welds other than fillets shall not exceed 0.548* Fy of the base metal. Base metal stress shall not govern for fillet welds.

Where the MS pipe supports combined loads, per combinations LC3 & LC4 do not exceed the original design basis loads, LC3 vs OBE loads and LC4 vs SSE loads, the supporting structure was not reevaluated for the beyond design basis combinations.

With the modifications, all the MS piping, pipe supports, and supporting drywell steel meet the above acceptance criteria. Also, all other current design and license basis criteria are met for the EPU conditions.

3.9.4 Control Rod Drive Systems

The design of the CRD system is discussed in Section 4.6. Control rod drive materials are addressed in Section 4.5.

3.9.5 Reactor Pressure Vessel Internals

The following sections provide descriptions of the physical layout of the reactor pressure vessel internals (Section 3.9.5.1), of loading conditions applicable to their structural and functional integrity (Section 3.9.5.2), and of their design evaluation (Section 3.9.5.3). Design of the control rods is described in Section 4.6. Information on the reactor internals materials is provided in Section 4.5.2.

3.9.5.1 Design Arrangements

In addition to the fuel and control rods, reactor vessel internals include the following components:

A. Shroud, including the tie rods with spring stabilizers,

B. Baffle plate (shroud support plate),

C. Baffle plate supports,

D. Fuel support piece,

E. Control rod guide tubes,

F. Core top grid,

G. Core bottom grid,

H. Jet pumps,

I. Feedwater sparger,

J. Core spray spargers,

K. Standby liquid control system sparger,

L. Steam separator assembly,

M. Steam dryer assembly, and

N. Incore nuclear instrumentation tubes.

The shroud is a stainless steel cylinder which surrounds the reactor core and provides a barrier to separate the upward flow of the coolant through the reactor core from the downward recirculation flow. In-vessel inspections found linear indications in the horizonal core shroud welds. Metallurgical evaluation determined intergranular stress corrosion cracking to be the root cause of the linear indications. A core shroud repair designed to structurally replace the core shroud's horizontal welds H1 though H7 (also accounts for potential flaws on horizontal weld H8) and provide vertical clamping forces on the shroud, was installed on Unit 2 during the refueling outage in 1995. The core shroud repair design includes low tension tie rods with spring stabilizers connected between the separator head support ring and the jet pump support plate. Four tie rods were evenly distributed in the annulus region of the reactor pressure vessel. Spring stabilizers were mounted at top guide support ring and the core plate support ring in the annulus area between the core shroud and the reactor pressure vessel wall. A middle spring stabilizer is mounted on the tie rod at the same elevation as the jet pump riser braces. The shroud repair upper and lower springs transmit seismic loads from the nuclear core directly to the RPV via the core plate support ring and the top guide support ring. Four symmetrically spaced core plate wedges were installed to provide a redundant load transfer path between the core plate and the core shroud. The function of the shroud repair is to ensure the intent of the original design is maintained (i.e. to ensure core geometry and a refloodable volume are maintained).

Bolted on top of the shroud is the steam separator assembly which forms the top of the core discharge plenum. This provides a mixing chamber before the steam-water mixture enters the steam separator. Refer to figure 3.9-4 for the reactor vessel cut away isometric for illustration of the component arrangement.

The recirculation outlet and inlet plenum are separated by the baffle plate joining the bottom of the shroud to the vessel wall. The jet pump diffuser sits on and is welded to the baffle plate, making the jet pump diffuser section an integral part of the baffle plate.

The baffle plate and inner rim are made of Inconel to allow for welding to the ferritic base metal of the reactor vessel. The bottom of the shroud is welded on top of the rim, which provides for the differential expansion between the ferritic, Inconel, and stainless steel components. Inconel legs welded at intervals around the baffle plate support it from the vessel bottom head.

The baffle plate supports carry all the vertical weight of the shroud, steam separator and dryer assembly, top and bottom core grids, peripheral fuel assemblies, core plugs not carried on guide tubes, and jet pump components carried on the shroud. In addition, the supports must withstand the differential pressures of normal operations and blowdown accidents (either upward or downward), and for the vertical and horizontal thrust of the seismic design.

The reactor fuel supports are the 4-lobed, Type 304 stainless steel fuel support pieces mounted on top of the control rod guide tubes. Each support piece holds four fuel assemblies and is designed to hold the orifice plates used for core flow distribution. There are two types of orifices, one for peripheral assemblies and one for nonperipheral assemblies. The control rods pass through slots in the center of the support piece. Each fuel support piece is removed to take out the control rod with attached velocity limiter.

The control rod guide tubes extend up from the control rod drive housing through holes in the core bottom grid. Each tube is designed as a lateral guide for the control rod and as the vertical support for the fuel support piece which holds the four fuel assemblies surrounding the control rod. The guide tubes are fabricated from stainless steel with 0.165-inch nominal and 0.134-inch minimum wall thickness which results in a safety factor of 4 during the maximum applied loading. This maximum loading occurs at the end of control rod insertion so that even if the tube were to buckle the control rod would remain inserted. The bottom of the guide tube is inserted and locked into a sleeve in the control rod drive housing.

The core top grid appears as a series of beams at right angles forming square openings, each for four fuel assemblies. The grid provides lateral support and guidance for the assemblies. Holes in the beams are provided to receive the top hooks of the temporary control curtains, which are then prevented from unhooking by the adjacent fuel assemblies. The top grid is attached to the reactor core shroud.

The core bottom grid consists of a perforated stainless steel plate supported on a grid beam structure, which is in turn supported on the reactor core shroud. The fuel support pieces are held laterally in the grid openings. Sixteen fuel assemblies or core plugs at the periphery of the core, which are not adjacent to control rods, are directly supported by the bottom grid. Proper orificing for coolant flow is provided in the grid for these 16 assemblies. Smaller perforations in the core plate provide guidance for the incore neutron monitor guide tubes, between fuel assembly locations.

The 20 jet pumps are of stainless steel construction and consist of a driving nozzle, suction inlet, throat or mixing section, and diffuser. The jet pumps are arranged in two symmetric groups around the reactor core shroud in the downcomer annulus. Each of the 10 supply lines from the recirculation pumps supply high-pressure water to a pair of jet pumps. Each supply line is welded to a nozzle on the outside of the reactor vessel. On the inside of the vessel a stainless steel riser pipe terminates at the pair of jets. The riser is held in position by support arms welded to the vessel wall.

The jet nozzle, contoured inlet, and throat are joined together as a removable unit, clamped to the top piece of the riser by nut-locking system. The joint between the throat and the diffuser is a slip fit. The throat section is supported by a clamp ring attached to the riser.

The jet pump diffuser is a gradual conical section changing to a straight cylindrical section and flange at the lower end. The diffuser is inserted up through the hole in the baffle plate, and is supported by brackets from the vessel wall. A water seal is formed by a preloaded Belleville washer between the diffuser flange and the baffle plate and by a bellows seal welded at the top to the diffuser and down to the baffle plate.

The hydraulic and operational effects of the jet pump design are discussed in Section 5.4.1.

Feedwater sparger integrity is discussed in Section II.3.2 of Amendment No. 5 for the Dresden Unit 3 Plant Design and Analysis Report,[11] which also includes a discussion of the core spray sparger integrity. The following paragraphs, however, cover some of the features unique to the feedwater sparger.

Four feedwater spargers are utilized in the reactor. Each sparger is approximately 70 inches in length and mounted to the inside reactor vessel surface. The thermal sleeve is attached to the sparger midpoint; however, the sleeve is not welded to the vessel nozzle. Therefore the feedwater sparger is removable. The spargers are mounted in the vessel at one elevation to distribute the feedwater in a symmetric pattern about the vessel axis. Vibration consideration for feedwater spargers is the same as that discussed on page II.3.3-10 Amendment No. 5 for the Dresden Unit 3 Plant Design and Analysis Report.[11] Each sparger is supported by the thermal sleeve and a bracket mounted to each end of the sparger.

The feedwater nozzle inner bore, the thermal sleeves, and the feedwater spargers were modified (January 1981 on Unit 2 and January 1982 on Unit 3). This modification consisted of removing (by machining) clad from the feedwater nozzle, boring the inside diameter of the safe-end to accommodate the new feedwater sparger seal surfaces and finally installing the new design sparger. The modified thermal sleeve is a double seal/triple thermal sleeve which gives a dual seal interference fit (piston ring type). The four new stainless steel feedwater spargers now have nozzles instead of the conventional drilled round holes which were subject to cracking.

The new sparger/thermal sleeve design will extend the service life of the feedwater nozzles by limiting the amount of feedwater leakage past the thermal sleeve which will prevent thermal fatigue cracking in the nozzle and safe-end bores. Inspection intervals and methods were originally established in Table 2 of NUREG 0619. This equipment was inspected during D2R15 and D3R15 using an automated ultrasonic technique that met the intent of ASME Section XI Appendix VIII. The BWR Owners Group submitted a proposed alternate inspection to NUREG 0619 to the NRC (General Electric report GE-NE-523-A71-0594) dated October 30, 1995. The NRC accepted this proposal and issued a Safety Evaluations (TAC M94090) dated June 5, 1998. The SER permitted ultrasonic inspection meeting Appendix VIII of ASME Section XI to be used as an alternate to liquid penetrant inspections originally stipulated in Table 2 of NUREG 0619. Dresden will utilize the inspection methods and inspection frequencies stipulated in Table 2 of the SER.

The reactor has two 100%-capacity core spray spargers. Each sparger is in two halves to allow for thermal expansion and is supported by slip-fit brackets welded just below the top of the core shroud. Each half receives spray water from one of a pair of supply lines routed in the reactor vessel to accommodate differential movement between the shroud and the vessel. The supply line pair for each system terminates at a common vessel nozzle. The sparger distribution nozzles are pointed radially inward and downward at a slight angle to achieve specified distribution pattern.

During D3R14 and D2R15, automated ultrasonic "baseline" examinations were performed on the Core Spray down-comer piping welds internal to the reactor vessel between the reactor vessel nozzles and the shroud penetrations. Both units displayed evidence of IGSCC on these susceptible welds. A Flaw Evaluation was performed for each unit (S&L Reports: SL-5197 (D2) and SL-5130 (D3) which, considering flaw growth, evaluated structural design margin remaining and projected resulting leakage after two cycles of operation. The evaluations concluded sufficient structural margin would remain after two cycles and that the welds would be below code allowable stresses for all deisgn load combinations. The evaluations also concluded that the resulting projected leakage after two cycles of operation would be bounded by the leakage assumptions of the fuel vendors LOCA analysis. Monitoring of the condition of these welds will be as required by the applicable ASME Section XI flaw evaluation and as recommended by BWRVIP document “BWR Core Spray Materials Inspection and Flaw Evaluation Guidelines (BWRVIP-18)”. During D3R18 and D2R21, a sectional pipe replacement was performed on the lower portion of all four Core Spray downcomers on Unit 3 and on Unit 2.

The standby liquid control system sparger is a perforated pipe attached inside the bottom end of the core shroud. It discharges the sodium pentaborate solution into the cooling water which then rises upward through the reactor fuel.

The steam separator assembly consists of the core top plenum head into which are welded an array of standpipes, with a steam separator attached to the top of each standpipe. The assembly is bolted on top of the core shroud by long bolts which permit removal for refueling operations. The assembly is guided into place by vertical guide tracks on the inside of the reactor vessel and by locating pins on top of the shroud.

The fixed centrifugal-type steam separators have no moving parts. In each separator, the steam-water mixture rising through the standpipe passes vanes which impart a spin to establish a vortex which separates the steam from the water. The steam exits from the top of the separator and rises up to the dryers. The separated water exits from under the separator cap and returns to the trays among the standpipes, which drain into the downcomer annulus.

The steam dryer assembly is bolted on brackets on the inside of the reactor vessel wall below the steam outlet nozzle. A skirt extends down from the dryer assembly into the water to form a seal between the wet steam plenum below the dryers and the dry steam flowing out the top and down to the steam nozzles. Moisture is removed by impinging on the dryer vanes and flows down through collecting troughs and tubes to the water trays above the downcomer annulus. The vertical tracks inside the reactor vessel are also used to guide the dryer assembly into position.

There are 53 incore nuclear instrumentation guide tubes extending up through the bottom of the reactor vessel to the core top grid.

The guide tubes are inserted into the reactor through housings that are attached to the bottom head of the reactor vessel and extend down to the same level as the drive housing flanges.

Twelve of the tubes are closed at the top end and are designed for the same pressure as the reactor vessel to prevent leakage of reactor water. Four of these 12 tubes are for the source range monitor detectors and 8 for the intermediate range monitor detectors.

The other 41 each contain 4 local power range monitor (LPRM) incore detector strings and a guide tube for the traversing incore probe. Each of the 41 stainless steel tubes is approximately 1 inch in diameter, is open at the top for water cooling, and has a pressure seal at the bottom where the coaxial cables leave the reactor.

3.9.5.2 Loading Conditions

The reactor internals are designed mechanically to:

1. Provide an adequate distribution of coolant flow within the reactor, and

2. Maintain structural integrity during normal operations, seismic disturbances, and design basis accident conditions.

The specific design requirements for each internal component may vary due to differences in material, and location. Each component must be able to withstand the combined loadings from differential pressures and temperature, dead weight, fluid movement, control rod motion, seismic acceleration, and vibration. Allowable stresses as defined by the ASME Code will not be exceeded. Allowances must be made for thermal expansion, corrosion, and crud buildup.

The reactor core structural components are designed to accommodate the loadings applied during normal operation and maneuvering transients. Deflections are limited so that the normal functioning of the components under these conditions will not be impaired. Where deflections are not the limiting factor, ASME Section III is used as a guide to determine limiting stress intensities and cyclic loadings for the core internal structure.

The loading conditions which occur during excursions or loss-of-coolant accidents have been examined. The reactor core shroud, shroud support, and jet pump body, which comprise the inner vessel around the core within the reactor vessel, are designed to maintain a reflooding capability following a design basis loss-of-coolant accident. Reflooding the reactor core to the top of the jet pump inlets will provide adequate cooling of the fuel.

The design of the jet pump parts takes into account the pressure loading both in normal and accident conditions and the reactions at the supporting brackets due to differential thermal expansion of the pump and reactor primary vessel.

The reactor internals are designed to preclude failure which would result in any part being discharged through the main steam line, in the event of a steam line break, which might block a main steam line isolation valve.

The structural components which guide the control rods are analyzed to determine the loadings which would occur in a design basis loss-of-coolant accident. The reactor core structural components are designed so that deformations produced by accident loading will not prevent insertion of the control rods.

Pressure differentials, jet reactions, and earthquake loadings have been considered in the analysis of the feedwater sparger. These stresses were all within the requirements of ASME Section III for Class A Vessels. The sparger was analyzed assuming the thermal sleeve is welded into the nozzle. The resultant bracket loads were then given to the vessel manufacturer so that properly sized vessels brackets could be sized to meet the Section III criteria.

3.9.5.3 Design Bases

This section presents the details of key evaluations performed for the reactor vessel internals. A discussion of the jet pump assembly and its relationship to the vessel and to the other reactor internal components during steady-state and transient operation is included in the jet pump topical report APED-5460.[13] Section 4.3.2.2 of the report describes the stress analysis that was performed to demonstrate the adequacy of the structural design of the jet pump assembly and the core shroud during operation of the emergency core cooling system, which is the condition of maximum stress for the jet pump core shroud assemblies.

The reactor internals which must maintain their functional integrity to assure safe shutdown following the various postulated accidents are the following:

A. Reactivity Control Systems

   1. Control Rod and Control Rod Drive Systems

      a. Fuel channel-core support complex.

      b. Control rod control assemblies.

   2. Stand-by Liquid Control

B. Emergency Core Cooling Systems

   1. LPCI System

      a. Core shroud and baffle, relative to the ability to maintain water level in the core.

      b. Jet pump structure relative to the ability to introduce and maintain a water level in the core.

   2. HPCI System; with components corresponding to those above.

   3. Core Spray System

      a. Core spray piping and sparger in the reactor pressure vessel.

      b. Arrangement of the core support complex, relative to its ability to accept water from the core spray.

Based on analyses of the reactor internals during both normal and accident conditions, it was determined that stresses in the individual components are limiting, except in the following areas where deformation is the controlling parameter:

A. Scram capability coincident with a seismic event has been evaluated to ensure that control rods can be fully inserted under the most limiting channel distortion conditions (i.e. safe shutdown earthquake (SSE)). A fuel channel experiencing a channel distortion symptom (i.e. bow, bulge, etc.) may interact with a control rod blade and create undesired friction between the two components. The friction can be further increased during a seismic event as the affected channel will experience lateral movement concurrent with vertical and horizontal displacement of the control rod blade.

   Deflection of the fuel channels under accident pressure conditions is limited to an amount substantially less than that which would prevent control rod drive insertion. The maximum friction force exerted on the fuel channel by the control rod (as a result of interference), during a design basis accident, is less than 400 lb_f. The 400 lb_f is based on the worst case steady-state friction present from the control rod blade (~300 lb_f) and the worst case additional transient friction (~100 lb_f). The normal steady-state channel-control blade friction level is below 50 lb_f, so the typical total friction force is 150 lb_f. Under these insert conditions, the fuel bundles which weigh about 700 lb_f will not be lifted due to the resisting insertion forces of <400 lb_f friction from the deflection of its members. The minimum force exerted by the control rod drive on a control blade is 2500 lb_f. Therefore, control blades can be fully inserted against the forces of fuel channel deflection under the most severe accident conditions.

   Minimum reactor pressure for which a scram can still be assured to fully insert during a seismic event is established based on the best available industry data and site-specific analyses. Typically, 600 psig reactor pressure in conjunction with 940 psig accumulator pressure will ensure that susceptible channel distortion affected cells will incur complete scram insertion. Procedures require pre-startup testing of the susceptible population, with the control rod in any cell indicating high friction left at position 00 until after nominal reactor pressure (920 psig) is achieved. For a shutdown, the susceptible population is either tested prior to insertion or inserted to position 00 prior to reduction below the limiting reactor pressure.

B. Horizontal deflection of the control rod drive housings is limited to a value which through test has been demonstrated not to impede control rod insertion.

C. Deflection of the core plate and lower grid assembly is limited under normal operation to preclude taking up vertical clearance between the core plate and control rod guide tubes so that the core bypass leakage flow can be predicted. This results in stresses that are below yield even during accident conditions. The maximum deflection of the core plate under accident conditions is limited to 0.125 inches, which represents a considerable factor of safety below the deflection at which the core plate and guide tube could come into contact.

The maximum value of primary stress in reactor internal components generally results from the large pressure difference created when either the recirculation line or the steam line are completely severed. A discussion of these two accidents is given in Sections 3.9.5.3.1.2 and 3.9.5.3.1.3, respectively.

The sensitive point within the reactor pressure vessel which is most affected by operation of the emergency core cooling systems (HPCI and LPCI) is in the area of the jet pump to baffle plate joint. The stress and fatigue evaluation of this location is discussed in Section 3.9.5.3.2.

3.9.5.3.1 Pressure Loadings

Values of calculated pressure difference versus design pressure capability for major reactor internal components are included in Tables 3.9-18 and 3.9-19 to show the margin of safety that exists below ASME Section III limits. The margin of safety for these components which actually exist, based upon the GE Atomic Power Equipment Department (APED) design criteria for reactor internals, is equal to or greater than the margin specified in the tables. The loading combinations, and stress and deformation limits for reactor internal components are also discussed in these criteria.

A "best estimate" thermal-hydraulic analysis of a main steam line break was performed in 1994 as part of the core shroud flaw evaluations for Dresden Unit 3 and Quad Cities Unit 1. The results of this analysis provide differential pressures for use in performing structural flaw evaluations of the core shroud and internals. A detailed thermal-hydraulic analysis of a recirculation suction line break was performed in 1994 as part of the core shroud flaw evaluations for Dresden Unit 3 and Quad Cities Unit 1. The results of this analysis provide a detailed definition of the asymmetric blowdown loads that are applied to the core shroud under the bounding conditions of a recirculation suction line break.

As part of the Extended Power Uprate (EPU), structural integrity assessment of the key reactor internal components was performed, (Ref. 37). The thermal hydraulic analysis data, Reactor Internal Pressure Differences, and the acoustic and flow induced loads due to postulated Recirculation line break (LOCA) were used as input to the EPU evaluation.

3.9.5.3.1.1 Thermal-Hydraulic Model

In this section, the internal pressure forces which would be imposed across the internal reactor components during rapid depressurizations associated with pipe breaks are discussed in detail.

Internal reactor pressure forces are calculated for two postulated break conditions, a steam line rupture and recirculation line rupture. The steam line break is assumed to be a guillotine line severance which is located upstream of the flow limiter. This break gives the maximum break steam flow and maximum pressure forces. The conclusion of the event is complete blowdown to the drywell. The recirculation line break is assumed to be a guillotine line severance at the pressure vessel outlet. This places the break in the downcomer and the conclusion of this event is again to have a complete blowdown to the drywell. In both cases, reflooding of the reactor is accomplished by the emergency core cooling system. The break is assumed, in each case, to occur while the plant is operating at 2527 MWt with 98 x 10^6 lb/hr core recirculation flow.

When calculating internal pressure loading due to a blowdown accident, an analytical model is employed in which the pressure vessel is divided into major chambers or nodes.

The original design basis thermal-hydraulic model was prepared to calculate the various design basis input parameters required to support the design of the RPV and RPV internals. In the original design basis thermal-hydraulic models each node was connected to the adjoining nodes by a flow resistance as shown in Figure 3.9-5. The five nodes modeled are:

1. Sub-cooled lower plenum,

2. Saturated core,

3. Saturated upper plenum,

4. Saturated mixing plenum, and

5. Saturated steam dome.

The lower plenum to core resistance includes the inlet orifice, acceleration, local, and flow losses to the core midplane. The core to upper plenum resistance consists of the remaining core local losses and flow losses. The separator resistance is between the upper plenum and mixing plenum and steam dome. In the recirculation line break, one additional resistance is included - the resistance between the downcomer and the lower plenum through the open jet pumps of the broken line. Jet pumps are described in Section 5.4.1.

Referring to Figure 3.9-5, the pressure forces acting on major components are as shown in Table 3.9-20. The two design basis breaks will be discussed individually.

In 1994 additional thermal-hydraulic models were developed as part of the core shroud repair program. Separate thermal hydraulic models and analyses were performed for the steam line and recirculation line break conditions. The following sections provide a description of the analyses performed and the results obtained.

As part of the Extended Power Uprate (EPU), structural integrity assessment of the key reactor internal components was performed, (Ref. 37). The thermal hydraulic analysis data, (including GE 14 fuel), Reactor Internal Pressure Differences, and the acoustic and flow induced loads as a result of the postulated Recirculation line break (LOCA) were used as input to the EPU evaluation.

3.9.5.3.1.2 Recirculation Line Rupture

The instantaneous recirculation line rupture (double-ended) causes high flowrate from the downcomer and plenum regions. Initially, supercritical flow (high-single-phase flow) exists in the blowdown lines prior to flashing of the water. After bubbles form in the lines, two-phase critical flow is established and the blowdown rate is reduced from the supercritical flow value. No credit is taken for friction losses in the broken line.

Although the flowrate from the downcomer is high, the pressure change rate in the mixing plenum is only about 20 psi/s assuming no turbine control valve action to maintain pressure. Because large amounts of saturated water are present in the mixing plenum, the depressurization rate is low due to the accompanying flashing.

Large pressure forces due to depressurization of the subcooled lower plenum do not develop in the BWR plant. The principle reason in this case is that, in the event of a line break, the subcooled lower plenum does not discharge directly to the atmosphere. Instead, it discharges to the downcomer region through the inoperative jet pump diffusers, and the downcomer pressure is maintained by compression of the steam above the mixing plenum.

Thus, large pressure forces cannot develop across the diffusers and shroud support because the inoperative jet pump diffusers are open between the downcomer and lower plenum. Even though the lower plenum is subcooled, its depressurization rate is limited by the downcomer and mixing plenum depressurization rate. The fact that some water flows through the jet pump nozzles to the atmosphere is not serious since the flow will be critical or "choked" in the nozzles, and the total nozzle area is only 15% of a 28-inch OD recirculation line.

Results of the recirculation line break are given in Table 3.9-18 and compared to component capabilities. The upper shroud, lower shroud, and shroud support are of interest with respect to the emergency core cooling system. The guide tubes and core plate are necessary for scram capability.

The calculated maximum pressure differential across the core for the recirculation line break does not increase above that at rated conditions, (i.e., 18 psi upward); well below the 45-psi pressure differential required for fuel bundle lifting.

The calculated maximum pressure differential across the fuel channel would be approximately 9 psi outward (initial value) for the recirculation line break. Core inlet flow decreases to about 40% of rated flow resulting in a decrease in channel box pressure level. Since the channel deflection is no more than that occurring during normal operation, control rod interference cannot occur.

If the mechanism by which the fluid is actually accelerated to its maximum flowrate is specifically to be considered, then the effect of the actual break opening time must be included since this is a significant factor in the acceleration phenomenon. Following a sudden recirculation line break, about 7 - 75 milliseconds is required to accelerate the fluid to its maximum flow depending on the actual pipe length from the vessel to the break. Since the actual break opening time is expected to be 100 milliseconds or longer, a relatively gradual fluid acceleration will occur and the resulting asymmetric loads are low. Therefore, the loads discussed above are the maximum loads to be expected following a sudden complete pipe line break.

Pipe rupture studies such as those performed at GE and Battelle Memorial Institute investigating fracture mechanics provide some insight into the mechanism of break enlargement. Although no specific data is available which would quantitatively define break opening times to be expected for large systems, it is clear from these studies that large amounts of energy are required to cause sudden enlargement of an existing flow into a through-wall crack and subsequently into a large break which would allow unobstructed blowdown flow.

Since a finite time is required for this energy to be supplied by the fluid system to the crack location, the postulated large break cannot occur instantaneously. Furthermore, the studies have shown that an existing part-through wall flaw which is as much as 2 feet long would propagate through the wall and cause a detectable leak without propagating into the postulated large break. Therefore, it is expected that at least 100 milliseconds would be realistically required for a crack to propagate into a large break.

As discussed above, large asymmetric loads are not expected for a realistic break opening time. However, a hypothetical case has been analyzed in which the break opening was conservatively assumed to be instantaneous. It is assumed that the fluid pressure at the break drops instantaneously from rated pressure to saturation pressure and generates a step change in pressure which propagates toward the vessel. The analysis was performed for a break just outside the pressure vessel nozzle.

A detailed thermal-hydraulic analyses of a double-ended guillotine break of the reactor recirculation suction line was performed to obtain a more accurate definition of the asymmetrical lateral blowdown loads that are applied to the core shroud. The TRACG computer code was used to calculate a detailed pressure distribution in the downcomer annulus during the blowdown period as a function of time. The resulting pressure distribution was then used to compute the resultant lateral forces applied to the core shroud.

The TRACG model included detailed nodalization that was developed as a result of a sensitivity study regarding the effect of azimuthal and axial nodalization on the resulting blowdown load. Proportional simulation of the jet pumps, feedwater flow and frictional effects were included in the model. Other RPV components such as the steam separators, guidetubes and external recirculation loops were also modeled. Additional sensitivity studies were performed to determine the effect of nodalization, time step size, friction loss coefficient, and break flow area on the resulting blowdown load. The bounding case was determined to include a 120% safe end break area and a 100% friction loss coefficient. The lowest feed water temperature was used in the analysis to account for the subcooling in the reactor downcommer annulus, resulting in the bounding blowdown load.

The blowdown force and corresponding core shroud moment were calculated in a plane parallel to the break (0° - 180°) and orthogonal to the plane of the break (90° - 270°). The evaluation of the orthogonal plane was performed to account for the nonsymmetrical operation of the jet pumps during the transient. The results of the two orthogonal components were combined to provide a bounding estimate of the total applied lateral load. For the determination of the maximum force and moment, the critical time period is within the first five seconds when subcooled blowdown occurs and the highest load is applied to the core shroud. Once two-phase blowdown begins the loads decrease significantly. This analysis was performed to determine the bounding blowdown loads and thus does not include the acoustic wave response within the initial 50 milliseconds.

The acoustic phenomena associated with an instantaneous break of a reactor recirculation suction line results in a short duration asymmetric lateral load that is applied to the core shroud. This asymmetric load is generated as a result of the finite time that the shock wave takes to travel from the broken reactor recirculation suction line to the other side of the annulus. A reassessment of the acoustic lateral load was performed as part of the core shroud evaluations and repairs. Based on the available industry information (Reference 21), an equivalent static load of 60 kips was established. This bounding lateral load was calculated using the envelope of several different load time histories and the applicable dynamic load factors (Reference 23).

A finite element analysis of the core shroud was prepared to verify that the stresses are within the ASME section lll limits. This analysis and corresponding stress evaluation (References 29, 30, 31, 32 and 33) included all of the applied loading cases including recirculation line breaks and seismic.

The shroud is reevaluated for Extended Power Uprate (EPU) recirculation line break loads as described in Reference 37. The increased loads due to EPU result in stresses within the allowable stresses for the faulted condition. Therefore the shroud remains qualified for the EPU.

3.9.5.3.1.3 Steam Line Rupture

Following the instantaneous, double-ended, steam line rupture, critical flow is established in each broken line. Since the break is postulated to be upstream of the flow restrictor, the break area is the sum of one open steam line area plus one steam flow restrictor area. As shown in Figure 3.9-7, this break causes the system to depressurize at about 35 psi/s, during initial steam blowdown. About 4 seconds later the depressurization rate is reduced to about zero when the two-phase mixture at about 7 % quality enters the steam line. In comparison, for a steam line break downstream of a flow limiter the initial depressurization rate is 25 psi/s as reported in Section 15.6.4.6.

The design break is assumed to have a constant break area of 2.33 square feet. Actually the effective break area will diminish with time since the isolation valves are closing in one end of the break. When the isolation valves have been closed, the effective break area is reduced to only one steam line.

Rapid decompression of the subcooled lower plenum does not occur because the decompression rate is limited by the saturated upper core regions.

The initial pressure differential increase across the separators and shroud support is caused by the momentum effects associated with the accelerating flow into the depressurizing mixing plenum. The increased loadings at approximately 2 seconds are the result of saturating the previously subcooled lower plenum inventory. The high exit mass flowrate is associated with this phenomenon.

Flashing will decrease as the inventory becomes depleted. As this occurs the loadings across the various internal components will be reduced. Subsequently, no means exist for sustaining large differentials between any of the vessel regions and all pressure differentials drop to low values. For this reason the curves have not been extended beyond 10 seconds.

The shroud loads discussed above are the maximum loads that will occur following a main steam line break. The asymmetric load due to steam line break is small due to the compressible effects of steam and the large expansion as the wave enters the pressure vessel, and does not alter the design basis loads. Because steam is highly compressible, it is not possible to transmit a rarefaction shock similar to the one that can be transmitted in water even for an instantaneous break. In the event of a sudden complete steam line break, the linear gradient is such that the sonic velocity at the back of the wave (low-pressure) is much less than at the front of the wave (high-pressure). Therefore, even if the break is hypothetically assumed to be instantaneous, the compressible effects of the steam prevent the transmission of a shock wave.

Compressibility effects will also limit the amplitude of the linear rarefaction wave that would be transmitted into the pressure vessel. This is because steam is highly compressible and as the ramped rarefaction wave begins to expand into the pressure vessel, a relatively small decrease in pressure would result in a rapid increase in particle velocity which would quickly establish steady flow at the break. This has the effect of limiting the amplitude of the rarefaction wave that can be transmitted into the vessel.

Based on one dimensional plane wave theory, the amplitude of this ramped rarefaction wave would be further decreased by expanding to the cross-sectional area of the vessel. Since this low-amplitude plane wave would be propagating axially down the vessel, the asymmetric load on the shroud would be small and does not alter the design basis loads for the shroud.

The maximum vessel internal loading has been evaluated without any consideration of the rise in coolant level that would occur after a steam line break. This level rise would in fact cause two-phase blowdown from the vessel and thus reduce the depressurization rate and the time when the maximum loadings would occur. It is also assumed that the recirculation pumps remain at full speed through the transient. Since they help to sustain interregion pressure differentials this is a pessimistic assumption. Similarly the assumption of continued injection of full feedwater flow is conservative since it would contribute to the depressurization rate and thus maximize the internals loadings.

Besides the internal forces, there are two other concerns related to the postulated steam line break accident. The first is the possibility of lifting fuel bundles due to the transient pressure differentials imposed across the core. The second is the degree of interference that might exist between the control rods and the channel walls because the channel walls deflect outward under the pressure differentials existing at the time the blades are being inserted. Both of these concerns are alleviated because of the following conditions.

The calculated maximum pressure differential across the core would be considerably less than the 45-psi value required to lift fuel bundles. In fact, as shown by Curve 1 of Figure 3.9-8 it is only slightly over rated differential pressure. These calculations were based on the assumptions of continued feedwater flow, zero steam line resistance and constant break area which all tend to increase the depressurization rate and, therefore, cause the lower plenum to flash prematurely. Even if any bundles did lift, the bundle would only lift an inch or two before relief action would occur at the nose piece and the pressure drop across the core would be rapidly reduced. The results presented here bound the increased hold down margin for heavier GE 14 fuel bundles (Reference 38).

The maximum pressure differential tending to bulge the channel outward was calculated to be approximately 12 psi. Test data from a similar type fuel channel indicated that the deflections followed the elastic equation at room temperatures for stresses greater than 2 times the yield stress.

Therefore, based on this experimental factor and the corresponding yield stresses at operating temperatures the best estimate would be that a pressure differential of approximately 25 psi could be applied to the channel without causing the sides to deflect sufficiently to bind the control rods.

Even if it were possible for the channel walls to make contact with the control rods, the deflection is not sufficient to cause permanent distortion, and the channel springs back when the transient pressure decreases. Furthermore, the blades could be inserted even if the channel did pinch the blade. Calculations were performed assuming that a 20-psi transient peak pressure difference existed as a steady-state force on the entire channel. The net normal force acting on each of the control blade rollers was then calculated. (Note that GE control blades, which this calculation is based on, have rollers. ABB control blades have pads. This analysis is still applicable.) Assuming only sliding could take place and using a coefficient of friction of unity the total upward force required to force the walls apart was only 440 pounds per blade.

The control rod drive mechanism is characterized by high forces when scrammed. At zero reactor pressure a drive develops a force of 6000 pounds tending to insert the rod, using the energy stored in the accumulator. The effect of the accumulator decreases as reactor pressure increases, but is approximately 3000 pounds at a reactor pressure of 1000 psi at the beginning of the scram stroke, well in excess of the 440 pounds calculated above. The drive is also scrammed by reactor pressure alone, the force exerted from this energy source being approximately 1100 pounds. Thus, there is no question that the drives are capable of inserting the blades.

Another study was based on a statistical evaluation of the manufacturing tolerances considering three-point contact. The results of this study indicate that even with outward pressure differences of 25-psi adequate clearance for the control rod movement would remain. Furthermore the signal to insert the control rods would occur within approximately 1 second after the accident and the rods would be inserted before the peak pressure difference across the channel could occur.

Therefore, it is concluded that the pressure difference across the core is not sufficiently high to lift the bundles; that the control rods will be fully inserted before the maximum pressure differences across the channels occur; and that the calculated maximum pressure difference across the channel would not be sufficient to pinch the control blade.

3.9.5.3.2 Thermal Shock Effects on Core Internals

High-stress or strain points have been analyzed on the internals structure during the LPCI thermal shock transient. Three specific locations are summarized and shown on Figure 3.9-9:

1. Baffle plate ligament strains,

2. Shroud-to-baffle discontinuity strains, and

3. Inside-shroud highest irradiation zone.

The baffle plate peak ligament strain analysis results in a peak strain range of 6.5%. This strain range, while higher than the 5.0% strain range permitted in ASME Section III for 10 cycles of loading, corresponds to about 6 allowable cycles of an extended type ASME Section III curve which would apply to fewer loading cycles than 10. Figure 3.9-10 illustrates both the ASME Section III curve and the basic material curves from which it was established (with the safety factor of 2 on strain or 20 on cycles whichever is more conservative). It is seen that extension of the ASME Section III curve represents a similar criteria to that used in ASME Section III but applied to fewer cycles of loading than 10. For this 304 stainless steel material, a 10% peak strain range would correspond to 1 allowable cycle of loading. It is emphasized that even 10% strain level for single cycle loading represents a very conservative suggested limit because this has a large safety margin below the point at which even minor cracking would be expected to begin. Since the conditions which lead to the calculated peak strain range of 6.5% are not expected to occur even once during the entire reactor lifetime, the strain is considered quite tolerable.

The result of the baffle to shroud analysis for strain is as follows:

Amplitude of alternating stress	180,000 psi
Allowable ASME Section III cycles	220
Maximum strain range	1.34%

The shroud receives the maximum irradiation at the inside surface opposite the midpoint of the core where the total integrated neutron flux at end of life is 2.7×10^{20} nvt (greater than 1 MeV). The maximum thermal shock stress in this region is 155,700 psi or 0.57% strain. All reactor internal structural members located in high-flux regions, including the shroud, are constructed of 304 stainless steel which does not suffer from irradiation embrittlement. It does experience hardening and an apparent loss in uniform elongation but its reduction in area is not changed. Since the reduction in area is the property which relates to tolerable local strain, it can be concluded that irradiation can generally be ignored. However, even on the basis of changes in the total elongation, one would conclude that this material at 2.7×10^{20} nvt integrated flux would be capable of about 15 - 20% elongation.

The strain range of 0.57% was calculated at the midpoint of the shroud which is the zone of highest neutron irradiation. The value of 0.57% strain range was determined by dividing the calculated stress range of 155,700 psi (peak surface stress) by the modulus of elasticity for Type 304 stainless steel which was assumed to be 27.5×10^{6} psi. The calculated strain range of 0.57% represents a considerable margin of safety below measured values of percent reduction in area (which is the property that relates to tolerable local strain) for annealed Type 304 stainless steel irradiated to 1×10^{21} nvt (greater than 1 MeV). The value of percent reduction in area for Type 304 stainless steel reported in Reference 14 is a minimum of approximately 38% for a temperature of 550°F and neutron flux of 1×10^{21} nvt (greater than 1 MeV) and in Reference 15 a reduction in area of 52.5% is reported for a temperature of 750°F and neutron flux of 6.9×10^{21} nvt (greater than 1 MeV). At lower values of temperature or neutron flux, the percent reduction in area is generally even higher. Therefore, thermal shock effects on the shroud at the point of highest irradiation level will not jeopardize the proper functioning of the shroud following the design basis accident (DBA).

A "best estimate" thermal-hydraulic analysis of a main steam line break was performed in 1994 to support the core shroud safety assessments and flaw evaluations (Reference 19). Since this analysis was performed utilizing "best estimate" techniques it is not a design basis main steam line break analysis. The differential pressures calculated in this analysis are applicable only for use in structural flaw assessments and safety consequences evaluations.

As part of the extended Power Uprate, the reactor internals including the core shroud are reanalyzed as discussed in Reference 37.

3.9.5.3.3 Thermal Shock Effects on Reactor Vessel Components

Several high stress points on the reactor vessel have been analyzed approximately and conservatively to determine the effects of LPCI cold water injection. The points examined are as follows:

1. Recirculation inlet nozzle,

2. Midcore inside of vessel, and

3. Control rod drive penetration.

The results on the recirculation nozzle are as follows:

	Sleeve	Nozzle
Amplitude of alternating stress	595,000 psi	215,000 psi
Allowable ASME Section III cycles	12	130
Maximum strain range	4.5%	1.6%

The results at midcore inside of vessel are 67,500 psi peak stress. More than 1000 such cycles would be imposed under ASME Section III fatigue criteria. The total maximum vessel irradiation (greater than 1 MeV) at this point has been found to be 2.4×10^{17} nvt which is below the threshold level of any nil ductility temperature (NDT) shift for the vessel material. Therefore, irradiation effects can be ignored at all locations on the vessel.

The results on the control rod drive penetration are:

Amplitude of alternating stress	560,000 psi
Allowable ASME Section III cycles	14
Maximum strain range	3.7%

3.9.5.3.4 Seismic Loading

An initial dynamic analysis was performed which determined the seismic responses of the Dresden Unit 2 reactor internals (See sections 3.9.3.1.1.3.1). The methods, approximations, and computer programs used in this analysis are detailed in a report by GE Atomic Power Equipment Department.[16]

The nuclear steam supply system of Dresden Unit 2 was modeled with lumped mass configurations. The internals model included the following components: shroud, CRD housing, top guide and core plate, fuel, guide tubes, separators, dryer, and vessel head in addition to the flanges, vessel skirt, standpipes, pedestal, shield wall, building, foundation, and the vessel itself. Not included in the mathematical model were light components such as jet pumps, incore guide tube and housing, spargers and their supply headers. Representative damping values used were reinforced concrete structure, 5%; reinforced or prestressed concrete primary containment structure, 2%; vessel and skirt, 1%; shroud, 1%; fuel, 7%; guide tubes, 1%; and control rod drive, 1%. The maximum seismic shears and moments of reactor internals due to their respective SSEs were determined. These were used to determine the adequacy of the original component design.

As part of the core shroud repair project new horizontal seismic analyses of the RPV and RPV internals was performed. A description of the modeling and analyses performed is provided in section 3.9.3.1.1.3. The core shroud repair was designed to structurally replace the core shroud's horizontal welds H1 through H7 and provide vertical clamping forces on the shroud. The repair hardware was installed on Unit 2 during the D2R14 refueling outage. The shroud repair upper and lower springs transmit seismic loads from the nuclear core directly to the RPV via the core plate support ring and the top guide support ring. A new rebaselined seismic model including the RPV and the RPV internals was generated with the core shroud repair hardware installed. In these analyses all relevant modes of vibration of the coupled system were considered. The damping factors utilized are defined in Table 3.7-1.

The maximum siesmic shears and moments for the fuel assemblies were developed as a part of core shroud repair project.

For Extended Power Uprate (EPU), the effect of GE 14 Fuel assembly properties on the seismic loads were assessed and found to result in no significant change in loads. (Reference 37)

The effects of Westinghouse SVEA-96 Optima 2 fuel on the combined seismic/LOCA response of the reactor vessel internals and core were evaluated in a dynamic analysis of the fuel (Reference 41). That analysis showed that introduction of Optima 2 fuel results in no significant change in loads on reactor internals components and that all design criteria are met for the response of the Optima 2 fuel assemblies to the combined seismic and LOCA loading. The methods, computer programs, calculations, and results used in this analysis are documented in a Westinghouse calculation (Reference 41).

Starting from D3R24 refueling outage, Dresden is transitioning to reload application of the AREVA ATRIUM 10XM fuel. The ATRIUM 10XM fuel assembly design meets all the mechanical design criteria for BWR fuel designs. The structural dynamic evaluation of the fuel assembly and fuel channel due to the seismic loads is performed with inputs of component stiffness, natural frequencies, and damping values derived from the tests. The testing and analyses have shown the dynamic response of the ATRIUM 10XM fuel design to be very similar to other BWR fuel designs that the same basic channel configuration and weight. The testing and analyses conclude that the ATRIUM 10XM fuel assembly and channel meet the mechanical compatibility requirements for use in Dresden. The methods, testing and results of the analyses are documented in Reference 45.

To transition from a full core of Westinghouse fuel to AREVA fuel, the effect of a full core of AREVA ATRIUM 10XM and a mixed core with Westinghouse Optima 2 were evaluated by GE for the maximum seismic fuel midpoint deflection relative to the core plate and the maximum seismic member end loads on the reactor vessel internal components based on updated Dresden 2 & 3 primary structure models of record for Safe Shutdown Earthquake (SSE). The modeled core combinations are: (i) 1/3rd AREVA + 2/3rd Westinghouse, (ii) 2/3rd AREVA + 1/3rd Westinghouse, and a (iii) Full core of AREVA fuel. The evaluation concluded that seismic deflections for the three Westinghouse/AREVA fuel core configurations analyzed are less than those for the GE14 fuel design with the bounding deflection of fuel relative to the core plate.

Also, seismic loads due to AREVA/Westinghouse fuel combinations are qualified based on comparison to the seismic load analysis of record. The peak force for the reactor vessel with the introduction of AREVA and Westinghouse fuel typically decreases with one exception of an increase for the shroud shear at the support plate with a full AREVA core. The maximum stress of the limiting shroud support plate is relatively small and remains bounded with sufficient margin by the maximum value analyzed (Reference 46).

3.9.6 Inservice Testing of Pumps and Valves

Presently, inservice testing (IST) of pumps and valves is governed by the Fourth 10-Year Interval IST Program which will remain in effect through October 31, 2012. The IST program was developed in response to the requirements of 10 CFR 50.55a.

In accordance with 10 CFR 50.55a, IST programs are updated at 10-year intervals to comply with the requirements of the addition and addenda of the ASME Code. Specifically, the regulation requires that IST program revisions meet the requirements (to the extent practical) of the latest ASME Code edition and addenda incorporated by reference in Paragraph (b) of 10 CFR 50.55a 12 months prior to the start of the 10-year inspection interval. The current IST program is based upon the requirements of the ANSI/ASME Operations and Maintenance Standard OM Code (1998 Edition through 2000 Addenda), consistent with the requirements of 10 CFR 50.55a.

The construction permits for Dresden Units 2 and 3 were issued on January 10, 1966, and October 14, 1966, respectively. At that time the ASME Code covered only nuclear reactor vessels and associated piping up to and including the first isolation or check valve. Piping, pumps, and valves were built primarily to the Power Piping Code rules of USAS B31.1. Consequently, the Dresden IST program contains no ASME Code Class 1, 2, or 3 designed systems. The system classifications used as a basis for the IST program are based on the requirements given in 10 CFR 50.55a(g) and Regulatory Guide 1.26, and were developed for the sole purpose of assigning the appropriate IST requirements. Components within the reactor coolant pressure boundary (RCPB), as defined in 10 CFR 50.2, are designated Inservice Inspection (ISI) Class 1 as determined by 10CFR 50.55a, with the exceptions allowed by 10CFR 50.55a(c). Other safety-related components are designated as ISI Class 2 and 3 in accordance with the guidelines of Regulatory Guide 1.26. Pursuant to 10 CFR 50.55a(a)(1), IST requirements of Section XI of the ASME Code are then assigned to these components, within the constraints of existing plant design.

The extent of the Class 1, 2, and 3 designations for systems or portions of systems subject to the IST requirements are identified on the Dresden Piping and Instrumentation Diagrams (P&ID). In accordance with Regulatory Guide 1.26, the IST boundaries on the P&ID are limited to safety-related systems which contain water, steam, or radioactive materials.

Inservice inspection and testing of the reactor coolant pressure boundary is addressed in Section 5.2.4. Inservice inspection for Class 2 and 3 components is discussed in Section 6.6. Preservice inspection and testing of pumps and valves is discussed in Chapter 14.

3.9.6.1 Inservice Testing of Pumps

The inservice testing program for ISI Class 1, 2, and 3 pumps meets the requirements of ANSI/ASME Operations and Maintenance Standard OM Code ISTB. Where these requirements were determined to be impractical, specific requests for relief have been approved by the NRC.

The IST program establishes the requirements for inservice testing to assess the operational readiness of certain centrifugal and positive displacement pumps used in nuclear power plants. The pumps covered are those that are provided with an emergency power source, which are required in shutting down the reactor to the cold shutdown condition, maintaining the cold shutdown condition, or mitigating the consequences of an accident. In addition to ISI Class 1, 2, and 3 pumps, some safety-related pumps and some nonsafety-related pumps have been included in the IST program at the request of the NRC.

3.9.6.2 Inservice Testing of Valves

The IST program for ISI Class 1, 2 and 3 valves meets the requirements of ANSI/ASME Operations and Maintenance Standard OM Code ISTC. Where these requirements were determined to be impractical, specific requests for relief have been approved by the NRC.

The IST program establishes the requirements for IST to assess the operational readiness of certain valves and pressure relief devices (and their actuating and position indicating systems). The valves covered are those which are required to perform a specific function in shutting down the reactor to the cold shutdown condition, in maintaining the cold shutdown condition, or in mitigating the consequences of an accident. The pressure relief devices covered are those for protecting systems, or portions of systems, which are required to perform a specific function in maintaining the cold shutdown condition, or in mitigating the consequences of an accident. In addition to ISI Class 1, 2, and 3 valves, some safety-related valves and some nonsafety-related valves have been included in the IST program at the request of the NRC.

3.9.7 References

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Table 3.9-1

SUMMARY OF DESIGN BASIS, PREDICTED, AND ALLOWABLE THERMAL CYCLES FOR THE REACTOR PRESSURE VESSEL

Cycle Description	Units 2 and 3 Original Design Basis Allowable[1]	Unit 2 Cycle Prediction Year 40[2]	Unit 3 Cycle Prediction Year 40[2]	Units 2 and 3 New Design Basis Allowable[3]
Plant cooldown[4]	119	293	263	293
Plant heatup[4]	120	297	266	298
Safety relief valve blowdown[4]	1	3	0	5
Reduction of power for plant shutdown[4]	119	158	111	159
Turbine roll with feedwater injection[4]	120	154	107	160
Head spray injection[4]	119	15	23	119
Loss of feedwater heaters — full[4]	80	9	11	114
Loss of feedwater heaters □ partial[4]	80	16	41	80
Loss of feedwater flow[4]	80	10	12	80
Scram[4]	200	248	242	294
Turbine Trip[4]	40	78	73	NA[6]
Batch feedwater addition during hot standby or plant cooldown[4]	595	40	72	122
Reduced power operation, 75%–100%	NA	NA	NA	10,000[5]
Reduced power operation, 50%–75%	NA	NA	NA	2,000[5]
Vessel pressure test to 1250 psig	NA	NA	NA	130

SUMMARY OF DESIGN BASIS, PREDICTED, AND ALLOWABLE THERMAL CYCLES FOR THE REACTOR PRESSURE VESSEL

Cycle Description	Units 2 and 3 Original Design Basis Allowable(1)	Unit 2 Cycle Prediction Year 40(2)	Unit 3 Cycle Prediction Year 40(2)	Units 2 and 3 New Design Basis Allowable(3)
Improper start of shutdown recirculation loop	NA	NA	NA	10
Sudden start of recirculation loop	NA	NA	NA	10
Overpressure up to 1250 psig	NA	NA	NA	1
Overpressure up to 1375 psig	NA	NA	NA	1
Bolt-up	NA	NA	NA	123
Unbolt	NA	NA	NA	123

Notes:

1. Original cycles formed the original basis for Dresden design. These were the originally analyzed values.

2. Predicted cycles for each unit are based upon extrapolating actual counted cycles through October 1989 over the full 40 year plant life and are thus predictions of the actual cycles that each unit will experience.

3. New design basis allowables provide new basis for Dresden design and envelop the predicted cycles for both units.

4. Cycles reviewed by General Electric (GE).

5. There is no impact on vessel fatigue from reduced power operation cycles.

6. Turbine trip cycles are counted as either Loss of Feedwater Heater events or scram events, so an allowable is not applicable.

Table 3.9-2

REACTOR PRESSURE VESSEL INTERNALS
VIBRATION MEASUREMENTS LOCATION AND DIRECTION

		Quantity		
		Cold Flow Tests	Power Operation Tests	Type
1.	Four Control Rod Guide Tubes	4		SG
	Measure axial strain in center of span, 129" level, at 2 points 90° apart (±45°).			
2.	Four Incore Guide Tubes	4		SG
	Measure axial strain at approximately 167" level, for 2 points 90° apart (±45°).			
3.	Four Fuel Channels	8	4(1)	SG
	Measure axial strain at 2 levels, 288" and 327", in the center of the 2 faces adjacent to the control rod for each fuel channel.			
4.	Core Plate	3		A
	Measure acceleration in three directions vertical and horizontal at 0° and 90°. Mount triaxial array of sensors of temporary fuel nozzle plugs near the center of the core.			
5.	Shroud	4	4	D
	Measure the horizontal displacement of the shroud at the flange, 385" level, at the four locations 8°, 98°, 188° and 278°. (Tangential motion of shroud OD preferred.)			
6.	Separators	4		D
	Measure the horizontal displacement of the separator assembly at the 557" level by measuring the relative displacement of the outer ring with respect to the vessel wall at 8°, 188°, and 278°. (Tangential motion of the ring preferred.)			

Table 3.9-2 (Continued)

REACTOR PRESSURE VESSEL INTERNALS
VIBRATION MEASUREMENTS LOCATION AND DIRECTION

		Quantity: Cold Flow Tests	Quantity: Power Operation Tests	Type
7.	Recirculation Loops	6(2)		A
	Measure the horizontal motions of the following sections in the recirculation loops. The motion is to be measured in directions radial and tangential to the pressure vessel.			
a.	Suction line at -64" level	12(2)		M(2)
b.	Pump body at -252" level	12(2)		M(2)
c.	Top of pump motor at -84" level	12(2)		M(2)
d.	Inlet manifolds at 3 azimuth locations each and also including vertical direction.	12(2)		M(2)
8.	Jet Pumps			
	Measure the horizontal relative displacement at the following locations:			
a.	Top of riser pipe to pressure vessel at +-45° directions in pipe (referenced to vessel radial)	2		D or SG
b.	Top of jet pump throat to pressure vessel wall. Make measurements on both pumps in the pair attached to riser in a. above	2		D
c.	Across slip joint between throat and diffuser at +-45° directions (referenced to vessel radial). Make measurements on both pumps as in the pair of b. above	4		D
d.	Top of diffuser to pressure vessel at +-45° directions in diffuser. On same diffuser as c. above	4		D
e.	One sensor on each of 5 (1 riser for 2 pumps) assemblies.	5	2(1)	D

REACTOR PRESSURE VESSEL INTERNALS
VIBRATION MEASUREMENTS LOCATION AND DIRECTION

	Quantity		
	Cold Flow Tests	Power Operation Tests	Type
9. Differential Pressures (Dynamic 0 — 100 Hz)			
Measure the differential pressures at the following locations:			
a. Across the jet pump mounting ring, 123" level	1	1	
b. Across the core plate by using a spare fuel nozzle plug, 207" level	1		
c. Across the shroud head, 416" level	1	1	

SG = strain gauge
A = accelerometer
D = linear differential transducers

Notes:

1. The measurement points used were selected after a study of the cold-flow test results. They were the most active gauges on the two channels.

2. The 12 points here are locations that were by covered by a manual survey using a portable vibration meter "M." The six points shown are selected locations from the manual surveys and were displayed on the chart recorder.

Table 3.9-3

Intentionally Deleted

Table 3.9-4

Intentionally Deleted

Table 3.9-5

GOVERNING MARK I LOAD COMBINATIONS - TORUS ATTACHED PIPING

NUREG-0661 Load Combination Number	Load Combinations (1),(5),(8)	ASME Code Equation (2)
A-1	$P + DW + OL$	8
A-2(9)	$TE + THAM + TD + QAB_{\{D\}}$	10(3)
A-3(9)	$TE_{\{1\}} + THAM_{\{1\}} + TD_{\{1\}}$ or $TD_{\{2\}}$ or $TD_{\{3\}} + QAB_{\{D\}} + SSE_{\{D\}}$	10(3)^(3
A-4(9)	$TE_{\{1\}} + THAM_{\{1\}} + TD_{\{1\}}$ or $TD_{\{2\}}$ or $TD_{\{3\}} + PCHUG_{\{D\}}\ QAB_{\{D\}} + SSE_{\{D\}}$	10(3)
A-5(9)	$TE_{\{1\}} + THAM_{\{1\}} + TD_{\{1\}}$ or $TD_{\{2\}}$ or $TD_{\{3\}} + CHUG_{\{D\}}\ QAB_{\{D\}} + SSE_{\{D\}}$	10(3)
A-6(9)	$TE_{\{1\}} + THAM_{\{1\}} + TD_{\{3\}} + PSO_{\{D\}}$	10(3)
A-7(9)	$TE_{\{1\}} + THAM_{\{1\}} + TD_{\{3\}} + PS_D + QAB_D + SSE_D$	10(3)
A-8(4),(9)	$TE_{\{1\}} + THAM_{\{1\}} + TD_{\{3\}} + CO_{\{D\}} + OBE_{\{D\}}$	10(3)
A-9(9)	$TE_{\{2\}} + THAM_{\{2\}} + TD_{\{4\}} + SSE_D$	10(3),(10)
B-1	$P_O + DW + OBE_I + OL$	9
B-2	$P_O + DW + QAB + QAB_I + OL$	9
C-1	$P_O + DW + QAB + QAB_I + SSE_I + OL$	9
C-2	$P_O + DW + PCHUG + PCHUG_I + QAB + QAB_I + OL$	9
C-3	$P_O + DW + CHUG + CHUG_I + QAB + QAB_I + OL$	9
D-1(6)	$P_O + DW + PCHUG + PCHUG_I + QAB + QAB_I + SSE_I + OL$	9
D-2(6)	$P_O + DW + CHUG + CHUG_I + QAB + QAB_I + SSE_I + OL$	9
D-3	$P_O + DW + PSO + PSO_I + VCLO$	9
D-4(6)	$P_O + DW + PS + PS_I + VCL + QAB + QAB_I + SSE_I + OL$	9
D-5(6)	$P_O + DW + CO + CO_I + OBE_I + OL$	9
T-1(7)	$1.25P + DW$	8

Table 3.9-5 (Continued)

GOVERNING MARK I LOAD COMBINATIONS - TORUS ATTACHED PIPING

Notes:

1. Nomenclature:

DW	=	Dead weight loading
OBE_I	=	OBE inertia loading
OBE_D		OBE displacement loading
SSE_I	=	SSE inertia loading
SSE_D	=	SSE displacement loading
P,Po	=	Design pressure and maximum operating pressure, respectively
TE	=	Thermal expansion loads under normal conditions
$TE_{\{1\}}$	=	Thermal expansion loads under accident conditions
$TE_{\{2\}}$	=	Thermal expansion loads under long term post-LOCA conditions
THAM	=	Thermal anchor movement under normal conditions
$THAM_1$	=	Thermal anchor movement under accident conditions
$THAM_{\{2\}}$	=	Thermal anchor movement under long term post-LOCA conditions
OL	=	Operating thrust loads
TD	=	Torus displacement - normal conditions
TD_1	=	Torus displacement - small break accident conditions
TD_2	=	Torus displacement - intermediate break accident conditions
TD_3	=	Torus displacement - design basis accident conditions
$TD_{\{4\}}$	=	Torus displacement - long term post-LOCA conditions
QAB	=	Safety relief valve discharge pressure loads
VCL, VCLO	=	Vent-clearing pressure loads, with and without drywell/wetwell pressure differential, respectively
PS, PSO	=	Pool swell pressure loads, with and without drywell/wetwell pressure differential, respectively
CO	=	Condensation oscillation loads
PCHUG	=	Prechug loads
CHUG	=	Postchug loads
QAB_I, QAB_D	=	Inertia and displacement loads, respectively, from torus due to SRV discharge

Table 3.9-5 (Continued)

GOVERNING MARK I LOAD COMBINATIONS - TORUS ATTACHED PIPING

Notes: (Continued)

CO_I, CO_D	=	Condensation oscillation inertia and displacement loads, respectively
PS_D, PSO_D	=	Pool swell displacement loads, with and without pressure differential, respectively
$PCHUG_I$, $PCHUG_D$	=	Prechug inertia and displacement loads, respectively, of torus
$CHUG_I$, $CHUG_D$	=	Postchug inertia and displacement loads, respectively, of torus
PS_I, PSO_I	=	Pool swell inertial loads, with and without pressure differential, respectively

2. Equations are defined in ASME Section III, Subsection NC-3650.

3. As an alternate, ASME Section III, Equation 11 may be met.

4. For the DBA condition, SRV discharge loads need not be combined with CO and chugging loads.

5. Only governing load combinations are considered here.

6. The larger of LOCA and SSE combined by the SRSS method or LOCA and OBE combined by the absolute sum method is used.

7. Hydrostatic test condition: DW for all lines shall be with lines full of water at 70°F.

8. Independent dynamic loads may be combined by SRSS.

9. Stresses are for stress range. When dynamic displacement loads (i.e., QAB_D, SSE_D, etc.) are included in the inertia portion of the loads due to coupling analysis method, they are not required to be included here.

10. As an alternate, ASME Section III, Equation 10a may be met.

Table 3.9-6

ANALYTICAL TECHNIQUES RELATED TO CLASS I PIPING SYSTEMS

Item	Type of Analysis	Number of Modes
Recirculation Loop Piping	Response Spectrum	6
Suppression Chamber Ring Header	Response Spectrum	6
Feedwater Lines	Response Spectrum	7
Main Steam Lines	Response Spectrum	7
16-inch LPCI Shutdown	Response Spectrum	6
16-inch Shutdown	Response Spectrum	8
LPCI Pump Suction	Static	NA
LPCI Pump Discharge to Containment Cooling Heat Exchanger	Static	NA
LPCI Pump Discharge from Containment Cooling Heat Exchanger to Drywell	Static	NA
HPCI Turbine Exhaust	Static	NA
HPCI Pump Suction	Static	NA
HPCI Pump Discharge	Static	NA
Core Spray Pump Suction	Static	NA
Core Spray Pump Discharge	Static	NA
Containment Cooling to Heat Exchanger	Static	NA
Containment Cooling from Heat Exchanger to 48-inch Standpipe	Static	NA
Isolation Condenser Supply	Static	NA
Isolation Condenser Return	Static	NA

Table 3.9-7

MAXIUMUM STRESSES FOR ORIGINAL CLASS I STATICALLY ANALYZED PIPING
SAFE SHUTDOWN EARTHQUAKE

Item	System	Point No.	Material	O.D. (inches)	Thickness (inches)	Intensification Factor	Horizontal Static Coefficient K	Weight Stress (psi)	Pressure Stress (psi)	Total DBE Seismic Stress (psi)	Combined Stress Sigma (psi)	Yield Stress (psi)
1	HPCI Turbine Exhaust	105 Bend	ASTM A106 GR.B	24.000	.375	4.27	1.4	74	1529	7,576	9,179	30,000
2	HPCI Pump Suction	55 TGNT	ASTM A106 GR.B	16.000	.375	3.17	1.4	3935	744	5,612	31,008	35,000
3	Core Spray Pump Suction	125 TGNTR	ASTM A106 GR.B	16.000	.375	4.97	1.4	1314	744	5,068	32,463	35,000
4	Core Spray Pump Discharge	30 Bend	ASTM A106 GR.B	10.750	.593	1.83	1.4	534	1329	24,086	25,949	35,000
5	LPCI Pump Suction	145 TGNT BP	ASTM A106 GR.B	14.000	.375	4.57	1.4	1335	601	7,324	40,173	35,000
6	LPCI Pump Discharge to Containment Cooling Heat Exchanger	70 TGNT BP	ASTM A106 GR.B	12.750	.375	4.51	1.4	263	2327	5,462	28,147	35,000
7	LPCI Pump Discharge from Containment Cooling Heat Exchanger to Drywell	330 TGNT BP	ASTM A106 GR.B	18.000	.438	3.38	1.4	416	2859	8,818	34,070	35,000
8	Containment Cooling to Heat Exchanger	50 TGNT	ASTM A106 GR.B	16.000	.375	2.30	1.4	7168	2977	5,634	32,422	35,000

Table 3.9-7 (Continued)

MAXIUMUM STRESSES FOR ORIGINAL CLASS I STATICALLY ANALYZED PIPED
DESIGN BASIS EARTHQUAKE

Item	System	Point No.	Material	O.D. (inches)	Thickness (inches)	Intensification Factor	Horizontal Static Coefficient K	Weight Stress (psi)	Pressure Stress (psi)	Total DBE Seismic Stress (psi)	Combined Stress Sigma (psi)	Yield Stress (psi)
9	Containment Cooling from Heat Exchanger to 48-inch Stand pipe	220 Bend	ASTM A106 GR.B	14.000	.375	2.94	1.4	706	601	33,422	34,729	35,000
10	Isolation Condenser Supply	60 Anchor	ASTM A358 TP 304	14.000	.638	1.00	2.50	2694	5935	3,211	11,840	17,000
11	Isolation Condenser Return	90 Anchor	ASTM A358 TP 304	13.170	.585	1.00	2.50	2272	6112	5,476	13,860	17,000
12	HPCI Pump Discharge	305 Bend	ASTM A106 GR.B	14.000	1.093	3.34	1.4	2402	3710	21,608	27,720	35,000

Notes:

1. For piping 2" and under, ASTM A335 Grade P11 or P22 may be substituted for ASTM A106 Grade B material for the same schedule. For fittings and valves 2" and under, ASTM A182 Grade F11 or F22 may be substituted for ASTM A105 for the same rating. Substitutions are allowed up to a maximum temperature of 450°F (operating or design) and apply to non-safety related piping and fittings only. No generic substitution of safety related piping/fittings is allowed.

Table 3.9-8

SAFETY RELIEF VALVE DISCHARGE LINE
CLASS 3 MARK I PIPING ACCEPTANCE CRITERIA

Code Equation [1]	Service Level	Stress Limit[1]	Allowable Stress (ksi)[2]		Loads Combinations[3]
			Carbon	Stainless	
8	A	1.0 S_h	15.0	16.32	1
10	A, B	1.0 S_a	22.5	27.58	2
11	A, B	$S_h + S_a$	37.5	43.90	1+2
9	B	1.2 S_h	18.0	19.58	3, 4, 5
9	C	1.8 S_h	27.0	29.38	6, 7, 8
9	D	2.4 S_h	36.0	39.16	9, 10, 11

Notes:

1. See ASME Section III, Subsection ND-3650.
2. Carbon steel components include the SRVDL, ramshead, and reducer. Stainless steel components include the T-Quencher arms.
3. See Table 3.9-10 for load combinations.

Table 3.9-9

APPLICABLE ASME CODE EQUATIONS AND
ALLOWABLE STRESSES FOR MARK I TORUS ATTACHED PIPING

Stress Type	ASME Code Equation Number	Service Level	Stress Limit[1]	Allowable Value (ksi) [2], [3]	Governing Load Combination Number[4]
Primary	8	A	1.0 S_h	15.0	A-1, T-1
Primary	9	B	1.2 S_h	18.0	B-1, B-2
Primary	9	C	1.8 S_h	27.0	C-1 Through C-3
Primary	9	D	2.4 S_h	36.0	D-1 Through D-5
Secondary	10	A	1.0 S_a	22.5	A-2 Through A-8
Secondary	10_a	A	3.0 S_h	45.0	A-9
Primary and Secondary	11	A	$S_h + S_a$	37.5	[1]

Notes:

1. See ASME Section III, Subsection NC-3650.

2. Increased allowables as defined in NUREG-0661 have been utilized for piping systems which have been classified as nonessential.

3. Allowable stress values are for ASTM A106, Grade B material since this material is used for most of the TAP systems.

4. Governing load combination numbers are listed in Table 3.9-5.

For piping 2" and under, ASTM A335 Grade P11 or P22 may be substituted for ASTM A106 Grade B material for the same schedule. For fittings and valves 2" and under, ASTM A182 Grade F11 or F22 may be substituted for ASTM A105 for the same rating. Substitutions are allowed up to a maximum temperature of 450°F (operating or design) and apply to non-safety related piping and fittings only. No generic substitution of safety related piping/fittings is allowed.

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Table 3.9-10

SAFETY RELIEF VALVE DISCHARGE LINE
GOVERNING MARK I LOAD COMBINATIONS – CLASS 3 PIPING

Combination Number	Load Combination	Code Equation (1)	Service Level
1	PDES + WGHT	8	A
2	TRN 1	10	A
3	PMAX + WGHT + OBEI	9	B
4	PMAX + WGHT + $[(A1P1)^2 + (TQWJ)^2 + (UWCP)^2 + (UWCA)^2 + (SRVD)^2 + (SR1I)^2]^{1/2}$	9	B
5	PMAX + WGHT + $[(C3P1)^2 + (TQWJ)^2 + (UWCP)^2 + (UWCA)^2 + (SRVD)^2 + (SR1I)^2]^{1/2}$	9	B
6	PMAX + WGHT + $[(A1P1)^2 + (TQWJ)^2 + (UWCP)^2 + (UWCA)^2 + (SRVD)^2 + (SR1I)^2 + (SSEI)^2]^{1/2}$	9	C
7	PMAX + WGHT + $[(C3P1)^2 + (TQWJ)^2 + (UWCP)^2 + (UWCA)^2 + (SRVD)^2 + (SR1I)^2 + (SSEI)^2]^{1/2}$	9	C
8	PMAX + WGHT + $[(C3P2)^2 + (TQWJ)^2 + (UWCP)^2 + (UWCA)^2 + (SRDV)^2 + (SR1I)^2 + (PCDG)^2 + (PC2I)^2]^{1/2}$	9	C
9	PMAX + WGHT + $[(C3P2)^2 + (TQWJ)^2 + (UWCP)^2 + (UWCA)^2 + (SRVD)^2 + (SR1I)^2 + (PCDG)^2 + (PC2I)^2 + (SSEI)^2]^{1/2}$	9	D
10	PMAX + WGHT + $[(C3P2)^2 + (TQWJ)^2 + (UWCP)^2 + (UWCA)^2 + (SRVD)^2 + (SR1I)^2 + (CODG)^2 + (CO2I)^2 + (SSEI)^2]^{1/2}$	9	D
11	PMAX + WGHT + $[(C3P2)^2 + (TQWJ)^2 + (UWCP)^2 + (UWCA)^2 + (SRVD)^2 + (SR1I)^2 + (PSDG)^2 + (PS2I)^2 + (SSEI)^2]^{1/2}$	9	D

Notes:

1. See ASME Section III, Subsection ND-3650.

Table 3.9-10 (continued)

SAFETY RELIEF VALVE DISCHARGE LINE
GOVERNING MARK I LOAD COMBINATIONS – CLASS 3 PIPING

Nomenclature

PDES	=	Design pressure loading
PMAX	=	Maximum operating pressure loading
WGHT	=	Dead weight loading
THL1	=	Maximum operating thermal loading
THL2	=	LOCA condition thermal loading
TRN1	=	Envelope of THL1 and THL2
OBEI	=	OBE inertia loading
SSEI	=	SSE inertia loading
A1P1	=	Thrust force from first SRV actuation □ normal conditions
C3P1	=	Thrust force from subsequent SRV actuations □ normal conditions
C3P2	=	Thrust force from all SRV actuation □ LOCA condition
TQWJ	=	Water clearing drag loading
UWCA	=	Axial water clearing T-quencher thrust
UWCP	=	Perpendicular water clearing T-quencher thrust
SRVD	=	SRV bubble drag
SR1I	=	Torus inertia interaction loading
PS2I	=	Pool swell torus inertia interaction loading
CODG	=	Downcomer condensation oscillation pressure loading
CO2I	=	Condensation oscillation inertial interaction loading
PCDG	=	Downcomer chugging pressure loading
PC2I	=	Chugging inertial interaction loading
PSDG	=	Pool swell impact and drag loading

Table 3.9-11

SAFETY RELIEF VALVE DISCHARGE LINE (SRVDL)
MARK I STRESS ANALYSIS RESULTS – CLASS 3 PIPING

Component	Code Equation	Service Level	Stress (ksi) Calculated	Stress (ksi) Allowable
SRVDL	8	A	8.01	15.0
	10	A, B	15.99	22.5
	9	B	15.94	18.0
	9	C	23.87	27.0
	9	D	30.40	36.0
T-Quencher	8	A	2.51	16.32
	10	A, B	0	27.58
	9[(1)]	B	12.26	19.58
	9[(1)]	C	12.34	29.38
	9[(1)]	D	12.40	39.16

Notes:

1. Calculated Equation 9 stresses for service levels B, C, and D are approximately equal. Since the predominant load is perpendicular uneven water clearing, this load is applicable to all three service levels.

Table 3.9-12

MARK I ANALYSIS RESULTS FOR TORUS ATTACHED PIPING STRESS – UNIT 2

Service Level	A	B	C	D	Secondary
ASME Code Equation	8	9	9	9	10
Allowable Stress (ksi)	15.00 17.50(1)	18.00 21.00(1)	27.00 31.50(1)	36.00 42.00(1)	22.50/37.50(2) 26.25(1)/43.75(1)(2)
System Description	Maximum Stress (ksi)				
ECCS Suction Header	9.08	16.36	21.62	21.78	39.78
Vacuum Relief	5.50	11.00	19.00	21.10	15.20
LPCI Test Line and Spray Header Discharge from Pumps 2A/2B	6.29	16.38	23.93	30.85	21.80
LPCI Test Line and Spray Header Discharge from Pump 2C/2D	8.20	15.10	22.10	27.70	31.50(2)
HPCI Condensate Drain Pot	0.44	6.44	19.79	21.46	0.0
HPCI Turbine Exhaust	8.10	13.20	22.50	30.50	13.70
Pressure Suppression	2.40	5.30	9.10	10.20	11.30
Core Spray 2A Discharge	4.30	15.00	20.00	32.10	27.20(2)
Core Spray 2B Discharge	5.50	12.26	16.98	33.47	27.16(2)
LPCI Pump 2A/2B Suction	5.52	14.04	17.35	17.42	32.16(2)
LPCI Pump 2C/2D Suction	5.95	11.73	14.07	14.16	32.43(2)
Core Spray 2A Suction	4.63	14.72	21.88	22.05	16.01
Core Spray 2B Suction	5.10	13.96	25.05	25.16	15.34
HPCI Pump Suction	3.38	6.36	7.24	7.36	21.90

Notes:

1. For ECCS suction header.
2. Equation 11 is used in place of Equation 10.

Table 3.9-13

MARK I ANALYSIS RESULTS FOR TORUS ATTACHED PIPING STRESS — UNIT 3

Service Level	A	B	C	D	Secondary
ASME Code Equation	8	9	9	9	10
Allowable Stress (ksi)	15.00 17.50(1)	18.00 21.00(1)	27.00 31.50(1)	36.00 42.00(1)	22.50/37.50(2) 26.25(1)/43.75(1)(2)
System Description	Maximum Stress (ksi)				
ECCS Suction Header	12.57	17.28	23.70	23.98	37.37(2)
Vacuum Relief	5.35	10.20	18.70	21.40	30.81(2)
LPCI Test Line and Spray Header Discharge from Pumps 3A/3B	4.90	12.91	19.33	25.32	33.09(2)
LPCI Test Line and Spray Header Discharge from Pump 3C/3D	5.72	17.37	23.83	30.36	33.53(2)
HPCI Condensate Drain Pot	0.44	6.44	19.79	21.46	0.0
HPCI Turbine Exhaust	7.86	10.72	14.40	27.85	30.18(2)
Pressure Suppression	2.40	5.72	9.60	10.70	10.10
Core Spray 3A Discharge	5.62	12.67	22.95	35.73	31.01(2)
Core Spray 3B Discharge	4.06	11.36	17.05	34.82	13.24
LPCI Pump 3A/3B Suction	5.07	9.53	11.90	12.14	35.14(2)
LPCI Pump 3C/3D Suction	5.41	9.99	10.77	10.90	31.54(2)
Core Spray 3A Suction	6.84	13.97	18.51	29.88	15.14
Core Spray 3B Suction	5.07	16.27	20.83	34.46	14.12
HPCI Pump Suction	3.66	6.01	8.11	8.28	15.88

Notes:

1. For ECCS suction header.
2. Equation 11 is used in place of Equation 10.

Table 3.9-14

SAFETY RELIEF VALVE DISCHARGE LINE GOVERNING MARK I LOAD COMBINATIONS – CLASS 3 PIPING SUPPORTS

Combination Number(1)	Load Combinations(2)	Service Level
1A	WGHT	A
1B	WGHT + THL1	A
2A	WGHT + OBEI	B
2B	WGHT + THL1 + OBEI	B
3A	WGHT + $[(A1P1)^2 + (TQWJ)^2 + (UWCP)^2 + (UWCA)^2 + (SRVD)^2 + (SR1I)^2]^{½}$	B
3B	WGHT + THL1 + $[(C3P1)^2 + (TQWJ)^2 + (UWCP)^2 + (UWCA)^2 +(SRVD)^2 + (SR1I)^2]^{½}$	B
4A	WGHT + $[(A1P1)^2 + (TQWJ)^2 + (UWCP)^2 + (UWCA)^2 + (SRVD)^2 + (SR1I)^2 + (SSEI)^2]^{½}$	C
4B	WGHT + THL1 + $[(C3P1)^2 + (TQWJ)^2 + (UWCP)^2 + (UWCA)^2 +(SRVD)^2 + (SR1I)^2 + (SSEI)^2]^{1/2}$	C
5A	WGHT + $[(C3P2)^2 + (TQWJ)^2 + (UWCP)^2 + (UWCA)^2 + (SRVD)^2 +(SR1I)^2 + (PCDG)^2 + (PC2I)^2]^{½}$	C
5B	WGHT + THL2 + $[(C3P2)^2 + (TQWJ)^2 + (UWCP)^2 + (UWCA)^2 +(SRVD)^2 + (SR1I)^2 + (PCDG)^2 + (PC2I)^2]^{½}$	C
6A	WGHT + $[(C3P2)^2 + (TQWJ)^2 + (UWCP)^2 + (UWCA)^2 + (SRVD)^2 +(SR1I)^2 + (PCDG)^2 + (PC2I)^2 + (SSEI)^2]^{½}$	D
6B	WGHT + THL2 + $[(C3P2)^2 + (TQWJ)^2 + (UWCP)^2 + (UWCA)^2 + (SRVD)^2 + (SR1I)^2 + (PCDG)^2+ (PC2I)^2 + (SSEI)^2]^{½}$	D

Table 3.9-14 (Continued)

SAFETY RELIEF VALVE DISCHARGE LINE GOVERNING MARK I LOAD COMBINATIONS – CLASS 3 PIPING SUPPORTS

Combination Number(1)	Load Combinations(2)	Service Level
7A	$WGHT + [(C3P2)^2 + (TQWJ)^2 + (UWCP)^2 + (UWCA)^2 + (SRVD)^2 + (SR1I)^2 + (CODG)^2 + (CO2I)^2 + (SSEI)^2]^{1/2}$	D
7B	$WGHT + THL2 + [(C3P2)^2 + (TQWJ)^2 + (UWCP)^2 + (UWCA)^2 + (SRVD)^2 + (SR1I)^2 + (CODG)^2 + (CO2I)^2 + (SSEI)^2]^{1/2}$	D
8A	$WGHT + [(C3P2)^2 + (TQWJ)^2 + (UWCP)^2 + (UWCA)^2 + (SRVD)^2 + (SR1I)^2 + (PSDG)^2 + (PS2I)^2 + (SSEI)^2]^{1/2}$	D
8B	$WGHT + THL2 + [(C3P2)^2 + (TQWJ)^2 + (UWCP)^2 + (UWCA)^2 + (SRVD)^2 + (SR1I)^2 + (PSDG)^2 + (PS2I)^2 + (SSEI)^2]^{1/2}$	D

Notes:

1. Combination "A" = without thermal expansion load
   Combination "B" = with thermal expansion load

2. See Table 3.9-10 nomenclature for definition of individual loads.

Table 3.9-15

MARK I SAFETY RELIEF VALVE DISCHARGE LINE SUPPORTS INSIDE WETWELL
MAXIMUM AND CODE ALLOWABLE STRESSES FOR CRITICAL COMPONENTS

Item	Material	Maximum Stress (ksi)	Allowable Stress (ksi)
T-quencher support			
Beam	ASTM A53	0.85[(1)]	1.0[(1)]
Beam end connection bolts	ASTM A325	7.1	16.9
Beam end header support plate	ASTM SA516 GR.70	14.9	22.8
Support plate guides	ASTM SA516 GR.70	23.3	28.5
Support plate bolts	ASTM A 564; F_u = 190 ksi	39.5	62.7
Support plate welds (full penetration)	ASTM SA516 GR.70	13.4	22.8
Ramshead lug retainer	ASTM SA516 GR.70	20.9	22.8
Intermediate Support			
Beam	ASTM A53	0.74[(1)]	1.0[(1)]
Beam end connection bolts	ASTM A325	5.8	15.8
Beam end connection plates	ASTM SA516 GR.70	9.2	19.4
Collar support strut	ASTM A53 GR.B	0.80[(1)]	1.0[(1)]
Collar bolts	ASTM A325	3.0	25.8

Notes:

1. These values are the results of an interaction equation.

Table 3.9-16

MARK I SAFETY RELIEF VALVE DISCHARGE LINE SUPPORTS INSIDE DRYWELL
MAXIMUM RESULTS AND CODE ALLOWABLES FOR
CRITICAL SUPPORT COMPONENTS

Item	Material	Actual Interaction Ratio	Allowable Interaction Ratio
SRV guides in vent:			
Guide plate	ASTM A36	0.73	1.0
Auxiliary beam	ASTM A36	0.46	1.0
Auxiliary beam connection	ASTM A36	0.30	1.0

Table 3.9-17

LOAD COMBINATIONS — MARK I TORUS ATTACHED PIPING SUPPORTS

Load Combination Number	Load Combinations (1),(3),(6)
S-1	$DW + OL + OBE_I$
S-2	$DW + OL + QAB + QAB_I$
S-3	DW(5)
S-4	$DW + OL + QAB + QAB_I + SSE_I$
S-5	$DW + OL + QAB + QAB_I + PCHUG + PCHUG_I$
S-6	$DW + OL + QAB + QAB_I + CHUG + CHUG_I$
S-7(2)	$DW + OL + QAB + QAB_I + SSE_I + PCHUG + PCHUG_I$
S-8(2)	$DW + OL + QAB + QAB_I + SSE_I + CHUG + CHUG_I$
S-9	$DW + OL + OBE_I + CO + CO_I$
S-10(2)	$DW + OL + QAB + QAB_I + SSE_I + PS + PS_I + VCL$
S-11	$DW + OL + PSO + PSO_I + VCLO$
S-12	$DW + OL + OBE_I + TE + THAM + TD + OBE_D$
S-13	$DW + OL + QAB + QAB_I + TE + THAM + TD + QAB_D$
S-14(2 (7))	$DW + OL + QAB + QAB_I + PCHUG + PCHUG_I + TE_1 + THAM_1 + TD_3$(4) $+ QAB_D + PCHUG_D$
S-15(2) (7)	$DW + OL + QAB + QAB_I + CHUG + CHUG_I + TE_I + THAM_1 + TD_3$(4) $+ QAB_D + CHUG_D$
S-16(2) (7)	$DW + OL + QAB + QAB_I + SSE_I + PCHUG + PCHUG_I + TE_1 + THAM_1 + TD_3$(4) $+ QAB_D + SSE_D + PCHUG_D$
S-17(2) (7)	$DW + OL + QAB + QAB_I + SSE_I + CHUG + CHUG_I + TE_1 + THAM_1 + TD_3$(4) $+ QAB_D + SSE_D + CHUG_D$
S-18 (7)	$DW + OL + OBE_I + CO + CO_I + TE_1 + THAM_1 + TD_3$(4) $+ OBE_D + CO_D$

Table 3.9-17 (Continued)

LOAD COMBINATIONS — MARK I TORUS ATTACHED PIPING SUPPORTS

Load Combination Number	Load Combinations (1),(3),(6)
S-19 (2), (8)	$DW + OL + QAB + QAB_I + SSE_I + PS + PS_I + VCL + TE_1 + THAM_1 + TD_3^{(4)} + QAB_D + SSE_D + PS_D$
S-20 (8)	$DW + OL + PSO + PSO_I + VCLO + TE_1 + THAM_1 + TD_3^{(4)} + PSO_D$
S-21 (8)	$DW + OL + QAB + QAB_I + SSE_I + TE + THAM + TD + QAB_D + SSE_D$
S-22	$DW + OL + SSE_I + TE_2 + THAM_2 + TD_4$

Notes:

1. See Table 3.9-5, Note 1 for definition of individual loads.
2. Larger of LOCA and SSE combined by the SRSS method or LOCA and OBE combined absolutely.
3. The most severe combination of static loads must be considered.
4. TD_1, TD_2, or TD_3 case; whichever is most severe.
5. Applicable to nonwater lines only (hydrotest load).
6. Dynamic loads combined by SRSS for selected supports.
7. When dynamic displacement loads (i.e., QAB_D, SSE_D, etc.) are included in the inertia portion of the loads due to coupling analysis method, they are not required to be included here.

Table 3.9-18

RPV INTERNALS PRESSURE DIFFERENTIAL
DUE TO RECIRCULATION LINE RUPTURE (HISTORICAL INFORMATION)

Major Component	Maximum dP (psi)(3)	Design Capability dP (psi)
Shroud Support(2)	25 (initial)	100
Guide Tube	17 (initial)	68
Lower Shroud(2)	25 (initial)	185
Upper Shroud(2)	7 (initial)	185
Core Plate	17 (initial)	50
Shroud Head Assembly	9	25

Notes:

1. This is the pressure differential consistent with ASME Code allowable stresses. For primary loading, considerably higher differentials can be sustained before failure.

2. Core cooling dependent.

3. For extended power uprate operation, these values were not recalculated since the faulted condition RPV internals pressure differential for a steam line break (Table 3.9-19) is the bounding condition.

Table 3.9-19

RPV INTERNALS PRESSURE DIFFERENTIAL DUE TO STEAM LINE BREAK

Major Component	Maximum Delta-P (psi)[2]	Design Capability Delta-P (psi)[1]
Shroud Support	43.0	100
Guide Tube	29.5	68
Lower Shroud	43.0	185
Upper Shroud	20.0	185
Core Plate	29.5	50
Shroud Head Assembly	21.0	25

Notes:

1. Capability within ASME Code allowable stresses. For primary loading, considerably higher differentials can be sustained before failure.

2. Replacement Dryer Safety Analysis (Reference 42) for faulted conditions.

Table 3.9-20

RPV INTERNALS PRESSURE FORCES

Major Component	Pressure Force	Initial Value (psi)	EPU (psi) *
Shroud Support	P_1-P_4	25	25
Guide Tube	P_1-P_3	17	20
Core Plate	P_1-P_3	17	20
Lower Shroud	P_1-P_4	25	25
Upper Shroud	P_3-P_4	7	6
Shroud Head Assembly	P_3-P_4	7	6
Jet Pump Diffuser	P_1-P_4	25	25

Notes:

1. Refer to Figure 3.9-5 (BWR Internal Configuration) for location of pressure nodes.

2. *Values for Extended Power Uprate operation per Reference 37.

Table 3.9-21

RESULTANT CORE SHROUD LATERAL LOADS
FOR A RECIRCULATION SUCTION LINE BREAK

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3.10 SEISMIC QUALIFICATION OF CLASS I INSTRUMENTATION AND ELECTRICAL EQUIPMENT

This section describes the seismic qualification of Class I instrumentation and electrical equipment and their supports for original plant equipment and for certain new and/or replacement equipment added since 1985. The new and/or replacement equipment covered in this section are those Regulatory Guide 1.97 and 10 CFR 50.49 equipment that require seismic qualification. Seismic qualification of other components and piping systems are described in Sections 3.7, 3.8, and 3.9.

3.10.1 Seismic Qualification Criteria

The original seismic design criteria for Dresden Units 2 and 3 were developed by John A. Blume and Associates based on the recommendation of seismologist Perry Byerly.

Dresden Station was originally designed for a design level earthquake, equivalent to the operating basis earthquake (OBE) with a peak ground acceleration of 0.1 g. The design was reviewed to assure that the plant would resist twice the response loads for the 0.1 g earthquake without hindering the ability of the plant to be safely shut down.

Seismic design requirements and procedures have evolved significantly since the time Dresden Station received its construction permit. Recognizing this evolution, the NRC found that it was necessary to make a reassessment of the seismic safety of older operating plants. The Dresden Unit 2 seismic reassessment was performed under the Systematic Evaluation Program (SEP), Topic III-6, titled "Seismic Design Considerations," June 30, 1982.

Generic letter (GL) 87-02, "Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue (USI) A-46," which was issued on February 19, 1987, also addresses seismic assessment of older plants. The generic letter was issued to implement the USI A-46 resolution which concluded that the seismic adequacy of certain equipment in older operating nuclear plants must be reviewed against seismic criteria not yet in use when these plants were licensed.

Supplement No. 1 to Generic Letter 87-02 was issued on May 22, 1992. It transmitted the NRC staff's Supplemental Safety Evaluation Report No. 2 (SSER-2) on the Seismic Qualification Utility Group's (SQUG) Generic Implementing Procedure, Revision 2 as corrected on February 14, 1992 (referred to as GIP-2). The GIP-2 methodology relies primarily on the use of existing earthquake and testing experience data to verify the seismic adequacy of generic classes of equipment in contrast to seismic qualification procedures, which rely on analysis or testing of each item of equipment.

CECo committed to use the following as its method for responding to Generic Letter 87-02:

- GIP-2 in its entirety (both SQUG commitments and implementation guidance);

- Clarifications, interpretations, and exceptions to GIP-2 identified in SSER-2;
- Letter of August 21, 1992 (N.P. Smith to J.G. Partlow), SQUG Response to Generic Letter 87-02; and
- Letter of October 2, 1992 (J.G. Partlow to N.P. Smith), NRC Response to Seismic Qualification Group.

The following two clarifications apply:

1. CECo will use previously performed anchorage evaluations to expedite and/or minimize the GIP verification efforts, provided that the anchorage evaluations previously performed meet the criteria and procedures approved by the staff in SSER-2.
2. CECo will use existing seismic qualification test reports to demonstrate operability for any equipment on its safe shutdown equipment list that was previously qualified to IEEE 344-1975.

For new and replacement equipment, the GIP-2 methodology is applied if consistent with the licensing basis for the equipment. In particular, each new or replacement item of equipment and parts is evaluated for any design changes that could reduce its seismic capacity from that reflected by the earthquake experience or generic testing equipment classes. This includes verification of the seismic adequacy of commercial grade equipment being dedicated for safety-related purposes.

For Regulatory Guide 1.97 new and replacement equipment requiring seismic qualification, the requirements of IEEE 344-1975, Regulatory Guide 1.100, Revision 1, and Dresden Station will be satisfied.

3.10.2 Methods and Procedures for Qualifying Electrical Equipment and Instrumentation

The Dresden Unit 2 and 3 Master Equipment List (MEL) is a compilation of structures, systems, and components. Among other attributes, it identifies the safety classification, electrical classification (1E or non-1E), and seismic qualification applicability for electrical equipment and instrumentation.

The electrical component section of the MEL has been transferred to an electronic work control system such as EWCS.

Electrical equipment and instrumentation are qualified using one or more of the following methods:

A. Qualification by analysis,

B. Qualification by test, or

C. Qualification by combination of analysis and test.

3.10.2.1 Qualification by Analysis

Equipment is qualified by analysis if the equipment is not too complex and can be represented in a mathematical model for performing static analysis and/or dynamic analysis.

3.10.2.1.1 Static Analysis

Static analysis is performed for equipment which is determined to be rigid. The seismic forces on each component of the equipment are obtained by concentrating the total mass at the equipment's center of gravity and multiplying the values of the mass and the appropriate floor acceleration from the seismic response spectra. The resulting stresses are added to the other equipment stresses, as per the design criteria, to determine if the equipment is adequate to withstand this load.

3.10.2.1.2 Dynamic Analysis

Dynamic analysis is performed for flexible equipment. The equipment is analyzed using the response spectrum or time-history methods. Both of these methods have been used to qualify equipment for Dresden.

3.10.2.2 Qualification by Test

If the equipment is flexible and too complex to be represented properly by an analytical model, then the equipment is qualified by test. Testing is also performed where the equipment is required to operate during or after a seismic event but this cannot be established analytically. Seismic tests are performed by subjecting the equipment to vibratory motion which conservatively simulates the motion postulated at the equipment mounting during an OBE, followed by the vibratory motion associated with a safe shutdown earthquake (SSE).

The components of the instrumentation and electrical systems of the reactor protection system and engineered safety features of Dresden are capable of performing their required safety functions while withstanding the forces generated by a 0.2 g earthquake. Test results of components that were qualified by test were reported to the NRC. A detailed review of the equipment in the test program showed that the results of these tests are applicable to Dresden.

3.10.2.3 Qualification by Combination of Test and Analysis

Some electrical equipment and instrumentation are qualified by a combination of test and analysis. This qualification can be achieved through various methods such as extrapolation for similar equipment and extrapolation for similar seismic conditions.

3.10.3 Methods and Procedures of Analysis or Testing of Supports of Electrical Equipment and Instrumentation

Supports of Class I electrical equipment and instrumentation have been qualified using the same methods as described in Section 3.10.2. Examples of these supports include: battery racks, control consoles, cabinets, instrument racks, panels, motor control centers, and switchgears.

In response to IEN 80-21 and the SEP Positive Anchorage Verification Program, anchorage for safety-related electrical equipment was verified through plant walkdowns, and modifications were performed on the anchorage of some equipment to provide adequate support to withstand all postulated seismic loading.

3.10.4 Qualification Results

The protective system instrumentation and supporting panels or cabinets located in the control room were analyzed, tested, or investigated to confirm that they can perform their safety related functions under all required seismic loads at their respective locations.

The seismic qualification of new and replacement components for Regulatory Guide 1.97 components (that require seismic qualification) is performed to the requirements of IEEE 344-1975.

3.11 ENVIRONMENTAL QUALIFICATION OF ELECTRICAL EQUIPMENT

The environmental qualification (EQ) of electrical equipment for Dresden Station is performed per the guidelines of IE Bulletin 79-01B and the requirements of 10 CFR 50.49. After the issuance of IE Bulletin 79-01B, CECo submitted a response to the NRC in which the compliance methodology was delineated and a list of components to be qualified was enclosed. In June of 1981, the NRC issued a safety evaluation report (SER) to all licensees. In response to this SER, CECo submitted additional information regarding the qualification of safety-related electrical equipment. The NRC issued another SER to CECo along with a Technical Evaluation Report (TER) in December 1982. After discussion with the NRC, the final resolution of the SER was submitted and approved by the NRC in February 1986. The EQ program, as it has evolved and is currently being implemented, is described in the following sections.

The final rule, Environmental Qualification of Electric Equipment for Nuclear Power Plants, was published in the Federal Register as Section 50.49 of 10 CFR 50 on January 21, 1983, and became effective on February 22, 1983. In accordance with 10 CFR 50.49(k), equipment for Dresden Units 2 and 3, except for replacement equipment, may be qualified to the criteria specified in either the Division of Operating Reactors (DOR) Guidelines or NUREG-0588. Replacement equipment installed subsequent to February 22, 1983, must be qualified in accordance with the provisions of 10 CFR 50.49, using the guidance of Regulatory Guide 1.89, unless there are sound reasons to the contrary.

3.11.1 Equipment Identification and Environmental Conditions

Per the requirements of 10 CFR 50.49, each licensee must establish a program to environmentally qualify electrical equipment. Paragraph (b) of 10 CFR 50.49 groups this equipment into the following three categories:

A. Safety-related electrical equipment as defined in 10 CFR 50.49(b)(1);

B. Nonsafety-related electrical equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions by the safety-related equipment as defined in subparagraph (i) through (iii) of Paragraph (b)(1) of 10 CFR 50.49; and

C. Certain post-accident monitoring equipment.

The following discussions present the methodology used to identify electrical equipment falling into the above three categories.

3.11.1.1 Identification of Safety-Related Electrical Equipment Requiring Environmental Qualification

The definition of safety-related equipment covered by 10 CFR 50.49(b)(1) includes equipment relied upon to remain functional during or following design basis events. Paragraph (c) of 10 CFR 50.49 clarifies the scope to exclude some safety-related equipment. Equipment which is not required to operate in a harsh environment or which operates before the environment becomes harsh, need not be environmentally qualified. Therefore, not all safety-related equipment must be environmentally qualified.

Furthermore, only the equipment's ability to perform its safety functions (that is, those functions delineated in subparagraphs [i] through [iii] of 10 CFR 50.49[b][1]) must be considered in environmental qualification. The equipment's capability to perform nonsafety functions need not be assured.

The methodology used to identify safety-related electrical equipment that is required to be environmentally qualified is as follows:

A. All design basis events such as loss-of-coolant accidents (LOCAs), main steam line breaks (MSLBs) inside containment, and high energy line breaks (HELBs) outside containment were reviewed.

B. A list of systems required to mitigate the consequences of LOCAs, MSLBs, and HELBs was developed from plant safety analyses, Technical Specifications, and emergency operating procedures. The following six functions were considered:

    1. Emergency reactor shutdown,

    2. Containment isolation,

    3. Reactor core cooling,

    4. Containment heat removal,

    5. Core residual heat removal, and

    6. Prevention of a significant release of radioactive material to the surrounding environment.

C. The equipment required to remain functional in these systems was identified by review of system descriptions and appropriate drawings (P&IDs, schematics, electrical single-line diagrams, and control logic diagrams). System/component failure analyses were performed to identify the electrical equipment which require environmental qualification. Wiring diagrams were reviewed as necessary to identify connection types, terminal blocks, etc., which support electrical component function and also require environmental qualification. Plant emergency operating procedures were used as a guide to identify devices and display instruments required by the operator. Not all equipment in a particular safety-related system requires environmental qualification and post-accident active or passive functional capability in order to accomplish accident mitigation. Depending on system design, certain motor-operated valves, solenoid-operated pneumatic valves, temperature switches, limit switches, and instrumentation may not be required to perform a safety function or to mitigate the consequences of an accident in order for the system to accomplish its design basis safety function. Several other systems require only that the containment isolation portion of the system remains functional.

D. Plant areas with environmental parameters (pressure, temperature, humidity, radiation level, submergence level, etc.) which increase significantly above normal ambient conditions as a result of a design basis event were defined as harsh post-accident areas. Containment spray and radiation dose from recirculating radioactive fluids were included in these considerations.

E. A review of the location of the equipment was performed. Equipment required to function but not located within a harsh post-accident area was judged outside the scope of 10 CFR 50.49. In addition, certain equipment items are not exposed to a harsh environment at the same time or prior to when they are required to perform a safety function; these items were also judged outside the scope of 10 CFR 50.49.

F. For 10 CFR 50.49 electrical equipment, the required post-design basis event operating time was determined. This is the time period following occurrence of the design basis event for which the equipment must remain functional in order to accomplish safety or display functions or must not fail in an adverse manner. Subsequent failure of the equipment would not be detrimental to plant safety.

Based on the above methodology, a safety-related systems listing and an EQ Equipment List (including display instruments) were developed. These lists, together with other plant listings, were inputs to the development of the Master Equipment List (MEL). The MEL identifies the set of electrical equipment requiring environmental qualification. It is revised and updated on a continuing basis to reflect plant design changes and new information.

The electrical component /EQ section of the MEL has been transferred to an electronic work control system such as EWCS.

The methodology used in the MEL to identify safety-related electrical equipment requiring environmental qualification is in full compliance with the requirements of IE Bulletin 79-01B, Supplements 1 and 2, and 10 CFR 50.49. Therefore, the MEL is judged to address all electrical equipment within the scope of 10 CFR 50.49(b)(1).

3.11.1.2 Identification of Nonsafety-Related Electrical Equipment Requiring Environmental Qualification

Paragraph (b)(2) of 10 CFR 50.49 includes in its scope nonsafety-related electrical equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions by the safety-related equipment. Environmental qualification is not required for nonsafety-related electrical equipment whose failure under postulated environmental conditions does not affect the accomplishment of safety functions. An evaluation of the possibility of failure of nonsafety-related equipment in a manner detrimental to safety-related equipment used a combination of methods which are summarized below:

A. A list of safety-related electric equipment, as defined in Paragraph (b)(1) of 10 CFR 50.49, that are required to remain functional during or following design basis events such as LOCAs, MSLBs inside containment, and HELBs outside containment was generated (described in Section 3.11.1.1).

B. A system failure analysis was performed on each safety-related system to identify the set of equipment requiring environmental qualification. The system failure analysis included a review of safety system operation, system interaction, and operation of equipment within each safety system. This failure analysis identified all auxiliary systems and equipment which are necessary for the required operation of the safety-related system or equipment. This effort included review of the plant safety analyses, Technical Specifications, emergency operating procedures, P&IDs, schematics, wiring diagrams, electrical one-line diagrams, and control logic diagrams. The entire instrument loop associated with each identified instrument was reviewed to identify any other components whose malfunction could adversely affect operation of the equipment required to remain functional.

Based on the above considerations, the current MEL and the review methodology is judged to adequately address electrical equipment within the scope of 10 CFR 50.49(b)(2). The electrical component section of the MEL has been transferred to an electronic work control system such as EWCS.

3.11.1.3 Identification of Post-Accident Monitoring Equipment Requiring Environmental Qualification

Paragraph (b)(3) of 10 CFR 50.49 includes in its scope "certain post-accident monitoring equipment." Specific guidance regarding the parameters to be monitored is provided in Regulatory Guide 1.97, Revision 2. Equipment classified per Regulatory Guide 1.97, Revision 2, as Category 1 or Category 2, located in harsh post-accident areas is judged to be within the scope of Paragraph (b)(3) of 10 CFR 50.49 and is included in the EQ program. These items are also designated as EQ in the MEL. A compliance list of Regulatory Guide 1.97 instruments can be found in the Regulatory Guide 1.97 submittal to the NRC, dated August 1, 1985. The EQ section of the MEL has been transferred to an electronic work control system such as EWCS.

Based on the above considerations, all electrical equipment within the scope of 10 CFR 50.49(b)(3) is adequately addressed and incorporated into the EQ program.

Section 7.5 provides additional information regarding Regulatory Guide 1.97 compliance.

3.11.1.4 Environmental Conditions

The zones and selection of environmental parameters for Dresden are described in the following sections. In each case the harsh and normal environmental conditions represent conservative bounding conditions for these zones. If location-specific environmental conditions are necessary for qualification of certain equipment, a more detailed analysis may be performed to establish the environmental conditions for the location. Hence, the qualification environments specified in the qualification records may not agree in all cases with the zone parameters identified in the zone tables. In such cases, unique calculations have been performed to justify these conditions and are a part of the environmental qualification records.

The plant is divided into 54 distinct zones. Each of these is assigned a unique zone number. Figures 3.11-1 through 3.11-5 depict the location and boundaries of these zones within the plant. Tables 3.11-1 and 3.11-2 list separately the temperature, pressure, and radiation parameters for each zone under normal conditions and conditions due to a HELB and a LOCA. Table 3.11-3 lists the relative humidity conditions for each zone under normal, spurious, and accident conditions.

3.11.1.4.1 Harsh Post-Accident Areas

By definition, a harsh environment meets one or more of the following conditions due to a design basis event:

A. Temperatures above 120°F;

B. Total radiation exposure greater than 5 x 10^4 rads; or

C. Pressure transient in excess of atmospheric pressure resulting from a LOCA or HELB inside the drywell, the pressure suppression pool, and the main steam tunnel.

Exception to above 120°F definition is taken for the HPCI Rooms where the peak temperature resulting from a LOCA is 127°F with room coolers and 134°F without room coolers and the environment is considered mild.

Qualification for humid environments is required only when the humidity is in conjunction with harsh temperatures. Humidity aging is not required.

The IE Bulletin 79-01B requirement restricts design basis events that must be considered to the LOCA/HELB inside containment and HELB outside containment. In this context, the HELB includes the MSLB.

10 CFR 50.49 is more generally worded, requiring consideration of all postulated design basis events; however, the difference is not significant. A nonpipe break design basis event (such as a refueling accident, dropped fuel assembly during refueling) or a rod drop accident (reactivity excursion following sudden uncontrolled control rod withdrawal) produces less severe service conditions than a LOCA or HELB. The postulated nonpipe break design basis events are described in detail in Chapter 15.

The LOCA is postulated only in the drywell. A full range of LOCAs was analyzed, from a small rupture, where the makeup flow is greater than the coolant rate, to the largest LOCA, a double-ended rupture of the 28-inch recirculation line. The analysis showed that the full double-ended recirculation line break results in maximum drywell pressure. Therefore, this postulated LOCA defines the post-LOCA environment for environmental qualification.

Although smaller breaks may produce somewhat more severe long-term environments and longer operability time requirements for equipment, the most severe environmental conditions were used to perform environmental qualification. Major assumptions used in the LOCA analysis are discussed in Sections 6.2 and 15.6.

The only non-LOCA pipe break in the drywell that can produce a harsh environment comparable to the LOCA is the MSLB. Therefore, no other HELBs in the drywell are analyzed. The analysis of postulated MSLBs in the drywell considers postulated break sizes ranging from 0.01 to 0.75 square feet. A steam break larger than 0.75 square feet will result in rapid reactor vessel depressurization. This rapid depressurization will cause flashing of saturated water in the vessel, two-phase level swell, and two-phase flow from the break. The drywell temperature resulting from two-phase flow will be saturation temperature at the drywell pressure, which is considerably less than the temperature reached in a steam break.

The postulated HELBs outside the drywell are determined in "Analysis of Effects of Pipe Breaks Outside Primary Containment, Dresden Station Units 2 and 3," Special Report 37, Revision 1.[1] The HELBs considered are in the following lines:

A. Main steam;

B. Reactor feedwater;

C. High pressure coolant injection (turbine-driven);

D. Reactor water cleanup; and

E. Isolation condenser.

The analysis considers full area circumferential breaks (pipe severed) and longitudinal breaks (lengthwise split) at various locations in rooms where the lines run.

3.11.1.4.2 Mild Post-Accident Areas

A mild environment, by definition, meets all of the following criteria:

A. Temperature equal to or lower than 120°F;

B. Total radiation equal to or lower than 5×10^4 rads; and

C. Pressure no higher than that of all plant locations other than the drywell, the pressure suppression pool, and the main steam tunnel. A LOCA or HELB results only in minor changes in pressure in the mild environment areas.

Areas where the temperature does not exceed 120°F, due to a DBA, are considered mild temperature areas (except for HPCI Rooms where this temperature is 127°F with room coolers and 134°F without room coolers) and as such do not expose equipment required to perform safety related functions in response to a DBA to immediate or prolonged high-stress conditions during a DBA. The maximum service temperature represents no significant change from the normal temperature for equipment located in these areas. For all equipment located in these areas, the temperature is the result of normal plant operation; the loss of the heating, ventilation, and air conditioning (HVAC) system; or operation of equipment required for post-accident plant recovery. It is not the result of direct exposure to a LOCA or HELB environment. In all cases, the increase in temperature from the normal temperature to the maximum 120°F is gradual. The resulting applied stresses on the equipment are relatively low and well within the maximum stress level capability of the equipment, which is conservatively designed, fabricated, installed, and maintained. In some cases, the temperature during normal plant operation may slightly exceed 120°F. Operability of similar equipment in such mild temperature environments has been demonstrated by many years of experience in the utility industry. This operating experience does not indicate that a common-mode failure of safety-related equipment resulting from mild temperature environments is a problem. Furthermore, 10 CFR 50.49 does not require qualification to mild environments. Hence, no additional evaluations or documentation are necessary to ensure that this equipment performs its safety function.

3.11.2 Qualification Tests and Methodology

The approach to achieve environmental qualification of pertinent electrical equipment is summarized as follows:

A. Equipment located in mild temperature/mild radiation environments was not included within the scope of the NRC SER in accordance with 10 CFR 50.49. No action was required.

B. Qualification analysis or qualification testing (or a combination of both) was performed to ensure that equipment located in harsh temperature/mild radiation environments was fully qualified for the harsh temperature environment.

C. Equipment located in mild temperature and harsh radiation environments was qualified for a harsh radiation environment by either a combination of analysis and testing, or qualification testing, or was replaced with a fully qualified component.

D. Equipment located in harsh temperature/harsh radiation environments was qualified by testing or was replaced with a qualified component.

The DOR Guidelines and 10 CFR 50.49 require consideration of equipment aging due to material degradation occurring during normal plant life as a result of temperature, radiation, and wearing effects. Electrical equipment having materials susceptible to significant age-related degradation have been identified. A qualified (designated) life for each equipment type has been established with requisite replacement or component refurbishment schedules. Various methods, including the following, were employed in establishing the qualified life for equipment:

A. Use of available qualification test data on similar or identical components or equipment to support a conservative equivalent life extrapolation of the enveloping temperature test profile (using Arrhenius techniques);

B. Contact with vendors to obtain bills of material, material information, and technical data to identify age-sensitive materials;

C. Review and engineering evaluation of industry data and technical literature to determine material radiation threshold and the capability to withstand high temperature without degradation; and

D. Use of engineering analyses to establish a reasonable qualified life and justified replacement schedule.

Calculations, assumptions, technical data, and references are incorporated into the qualification records. The results of these evaluations and analyses are incorporated into the plant maintenance and surveillance program to ensure that equipment qualification is maintained.

Due to limitations in the state-of-the-art, synergistic effects were not addressed unless known synergisms were identified and were considered to have significant effect on equipment's safety function.

3.11.3 Qualification Test Results

All equipment that is determined to be within the scope of 10 CFR 50.49 is listed in the MEL issued by the Engineering Department. For all environmentally qualified equipment groups, an EQ binder exists which includes the qualification report; qualification data evaluation and maintenance, surveillance, and installation requirements. The set of EQ binders is a controlled document that is maintained by the Engineering Department. EQ binders provide documentation of evaluations, analyses, and test results to show that pertinent electrical equipment is environmentally qualified to perform intended functions for its qualified life plus post-design basis event exposure. The above satisfies the requirements of 10 CFR 50.49(d).

The existing maintenance and surveillance programs are used to specifically address the maintenance and surveillance requirements of EQ (e.g., required maintenance resulting from use of components and parts with limited qualified life). These current programs are as follows:

A. Like-for-like parts are used to maintain presently installed, qualified components whether these components are qualified to the DOR Guidelines or to NUREG-0588, Category I or Category II. When identical parts are not available, an engineering analysis and part evaluation are performed to ensure the replacement part is qualified for the intended function and environment. This ensures the continued qualification of installed components.

B. When presently installed components (qualified to the DOR Guidelines or to NUREG-0588, Category II) must be replaced, every effort is made to replace them with components qualified to NUREG-0588, Category I. Sound reasons to the contrary, as defined in Regulatory Guide 1.89, may preclude this upgrading practice when deemed necessary on a case-by-case basis.

3.11.4 Loss of Ventilation

Appropriate HVAC systems are provided throughout the plant, to maintain zone temperatures below qualification level and to protect equipment from extreme environmental conditions. This includes systems provided for the following areas:

A. Control room, battery rooms, computer room, and electrical equipment room (for further discussion, see Sections 6.4 and 9.4); and

B. Standby diesel generator rooms (see Section 9.4 for further discussion).

A room cooler consisting of an air-to-water heat exchanger and fan is also provided for each high pressure coolant injection (HPCI) and the low pressure coolant injection (LPCI)/core spray(CS) pump room (see Section 9.4).

The HPCI room cooler is required to exempt equipment in the HPCI room from the environmental qualification requirements of 10CFR50.49. However, analysis was performed to demonstrate that HPCI safety-related electrical equipment will remain operable even if the room cooler heat exchanger does not function, provided the fan can operate to mix air in the room.

The LPCI/CS room coolers are not required to support the operation of the LPCI/CS systems. All safety-related electrical equipment located in the LPCI/CS rooms that is subject to 10CFR50.49 has been environmentally qualified for the elevated temperatures resulting from a LOCA concurrent with a loss of the room coolers (LORC). The elevated temperatures from a LOCA concurrent with a LORC are reflected in Table 3.11-2.

In determining the normal temperature parameters for environmental zones (shown in Table 3.11-1), the evaluation includes the effects of normal plant operation, loss of HVAC, or operation of equipment required for post-design basis event plant recovery. Where comparatively high values for normal temperature appear, these result from conditions other than direct exposure to a LOCA or HELB.

3.11.5 Estimated Chemical and Radiation Environment

No special chemical environments that warrant investigation for their effects on safety-related equipment are present at Dresden. Demineralized water is used for containment spray, and its effect on safety-related equipment has been evaluated.

A radiation study was performed to establish integrated doses to equipment following a postulated LOCA. The core fission product inventory originally used to establish the post accident radiation environments was based on the GE document, "Radiation Source Information for NUREG-0578 Implementation Computer Run."[2]

The introduction of SPC fuel (ATRIUM-9B and 9x9-2) did not invalidate the results of the study (References 3 and 4) because the reactor core inventory and the potential radioactive releases from the core were not changed significantly from that obtained with GE fuel.

The core fission product inventory has since been revised to address a core uprate to 2957MWt and the use of other fuel types (GE14, Westinghouse OPTIMA2 and AREVA ATRIUM 10XM) with a 24 month fuel cycle. [5] [6]. The radiation environments for normal service and post accident conditions listed in Tables 3.11-1 and 3.11-2, respectively, reflect the uprated core. The methodology discussed below to develop the environmental dose conditions was utilized for original plant licensing and remains valid for uprate.

The fission products were diluted into the appropriate fluid media as follows:

Fluid	Noble Gases (%)	Halogens (%)	Other (%)
Suppression pool liquid	---	50	1
Reactor coolant liquid	100	50	1
Containment atmosphere	100	25	---
Reactor steam	100	25	---

Dilution of the fission products was considered using the fluid volume as the dilution media.

For components located inside the drywell, only gamma doses were considered if the component was enclosed in an inorganic material (e.g., valve motor actuators in metal enclosures). The gamma dose was established based on immersion of the component in the gaseous drywell atmosphere for the time that the component must remain functional. For components enclosed in organic material (e.g., cable), beta radiation doses were also calculated. Where components enclosed in organic materials are installed in metal enclosures (e.g., cable in conduit or flex-conduit), beta radiation is neglected. The effects of the beta dose have been considered by analysis. The analysis demonstrates that the combined gamma and beta dose is
less than the specified total radiation requirement for the components enclosed in organic materials.

For components located outside the drywell, source terms were established for piping systems containing reactor steam, reactor coolant liquid, suppression pool liquid, and containment atmosphere. Because the piping wall thickness is sufficient to shield against beta radiation, only gamma radiation was considered. Distance from each safety-related component to the piping systems containing post-LOCA radioactive fluids was established. The integrated dose was calculated based on the piping source term, distance from pipe to component, and component operating time. Where a component could receive doses from more than one piping system, the doses were added to derive a total dose.

3.11.6 References

1. Analysis of Effects of Piping Break Outside Primary Containment, Dresden Station Units 2 and 3, Special Report 37, Revision 1, February 1975.

2. General Electric Document, Radiation Source Information for NUREG-0578 Implementation Computer Run, SNUMB 7007S, November 1979.

3. "Response to IE Bulletin 79-01B, Post-LOCA/HELB Radiation Exposure Levels Received by ESF System Components for Dresden Nuclear Power Station, Units 2 and 3, Commonwealth Edison Company, Docket Numbers 50-237 and 50-249" (Bechtel Radiation Strudy), Specificaiton 13524-068-N101, Rev. 0, 1985.

4. "BWR Owners' Group NUREG-0578 Implmentation: Analyses and Positions for Plant-Unique Submittals", NEDO-24782, August 1980.

5. General Electric Document GE-NE-A22-00103-64-01, "Dresden and Quad Cities Asset Enhancement Program, Task T0802: Radiation Sources and Fission Products, August 2000.

6. DRE00-0073, Rev. 001, Dose and Dose Rate Scaling Factors to Evaluate Impact of EPU on Radiological Equipment Qualification and Vital Access.

Table 3.11-1

ENVIRONMENTAL ZONE PARAMETERS FOR NORMAL SERVICE CONDITIONS

Zone	Temperature (°F)	Pressure (psia)	40-Year Dose (rads)
1	150	16.2	1.1×10^7
2	104	14.7	4.2×10^4
3	104	14.7	$<1.0 \times 10^4$
4	104(1)	14.7	$<1.0 \times 10^4$
5	104(1)	14.7	$<1.0 \times 10^4$
6	104(1)	14.7	$<1.0 \times 10^4$
7	125	14.7	2.2×10^6
7a	120	14.7	$<1.0 \times 10^4$
7b	120(6)	14.7	1.2×10^7
8	150	16.2	1.1×10^7
9	151	14.7	2.2×10^6
9a	125	14.7	2.2×10^6
10	104	14.7	$<1.0 \times 10^4$
11	104	14.7	$<1.0 \times 10^4$
12	104	14.7	$<1.0 \times 10^4$
13	104	14.7	$<1.0 \times 10^4$
14	104	14.7	$<1.0 \times 10^4$
15(2)	104	14.7	$<1.0 \times 10^4$
16	104	14.7	$<1.0 \times 10^4$
17	104	14.7	$<1.0 \times 10^4$
18	120	14.7	$<1.0 \times 10^4$
19	120	14.7	$<1.0 \times 10^4$
19a	80	14.7	$<1.0 \times 10^4$
20	120	14.7	$<1.0 \times 10^4$
20a	124	14.7	2.2×10^6
21	150	16.2	2.2×10^7
22(4)	104	14.7	6.5×10^6
23	104	14.7	2.2×10^4
24	104	14.7	$<1.0 \times 10^4$
25	150	14.7	2.2×10^6
26	104	14.7	$<1.0 \times 10^4$
27	104	14.7	$<1.0 \times 10^4$
28	104	14.7	$<1.0 \times 10^4$

Table 3.11-1 (Continued)

ENVIRONMENTAL ZONE PARAMETERS FOR NORMAL SERVICE CONDITIONS

Zone	Temperature (°F)	Pressure (psia)	40-Year Dose (rads)
29	124	14.7	2.2×10^6
30	120	14.7	$<1.0 \times 10^4$
30a	95	14.7	$<1.0 \times 10^4$
31	125	14.7	2.2×10^6
31a	120	14.7	$<1.0 \times 10^4$
32	150	16.2	2.2×10^7
33	150	14.7	2.2×10^6
34	104	14.7	4.3×10^6
35	104	14.7	$<1.0 \times 10^4$
36	104	14.7	$<1.0 \times 10^4$
37	104	14.7	$<1.0 \times 10^4$
38(5)	120	14.7	$<1.0 \times 10^4$
39(3)	120	14.7	$<1.0 \times 10^4$
40	104	14.7	$<1.0 \times 10^4$
41	150	16.2	1.0×10^7
42	104	14.7	$<1.0 \times 10^4$
43	104	14.7	$<1.0 \times 10^4$
44(7)	104	14.7	$<1.0 \times 10^4$
45	104	14.7	$<1.0 \times 10^4$
46	120	14.7	$<1.0 \times 10^4$
47	120	14.7	$<1.0 \times 10^4$

Notes:

1. When the ECCS pump in the room is operating, the normal service temperature can rise to 150°F.
2. Zone 15 is a 5-foot zone around 18-inch line 1603-18".
3. Zone 39 is a 5-foot zone around 18-inch line 7504-18".
4. Pump rooms are seperate from the HX bay and experience lower doses per NEDO-24218; however, the area is currently included as part of Zone 22 (2.2×10^4 rads).
5. The normal 40-year dose in the area around the steam jet air ejectors is 6.4×10^6 rads per NEDO-24218.
6. Section 9.4.3.1 reports design temperatures of 103°F in occupied areas, 120°F in cells and collector tank room, and 150°F in the concentrator and concentrator tank waste cells.
7. The portion of Zone 44 which is north of Column H is regarded as part of the Turbine Building and has the following normal service parameters: 120°F, 14.7 psia, $<1.0 \times 10^4$ rads.

Table 3.11-2

ENVIRONMENTAL ZONE PARAMETERS FOR POST-ACCIDENT CONDITIONS

	HELB		LOCA			
Zone	Temperature (°F)	Pressure (psia)	Temperature[12] (°F)	Pressure (psia)	1-Hour Dose (rads)	30-Day Dose (rads)
1	338	40.0	295	63.0	2.8×10^{7}	1.4×10^{8}
2	242	15.1	175	14.7	2.0×10^{6}	1.4×10^{7}
3	221	14.9	127/128	14.7	$<1.0 \times 10^{4}$	$<1.0 \times 10^{4}$
4	270[5]	14.9[5]	190[6]	14.7	2.9×10^{5}	5.7×10^{6}
5	270[7]	15.0[7]	190[6]	14.7	2.9×10^{5}	5.7×10^{6}
6	298[8]	16.6	127[10]	14.7	2.5×10^{5} [(2)]	4.9×10^{6} [(2)]
7	120	14.7	120	14.7	$<1.0 \times 10^{4}$	$<1.0 \times 10^{4}$
7a	120	14.7	120	14.7	$<1.0 \times 10^{4}$	$<1.0 \times 10^{4}$
7b	120	14.7	120[11]	14.7	$<1.0 \times 10^{4}$	$<1.0 \times 10^{4}$
8	338	40.0	295	63.0	2.8×10^{7}	1.4×10^{8}
9[13]	304	27.5	150	14.7	1.5×10^{5}	2.8×10^{6}
9a	304	27.5	125	14.7	1.5×10^{5}	2.8×10^{6}
10	201	14.8	135/128	14.7	3.6×10^{5}	2.0×10^{6}
11	201	14.8	135/128	14.7	1.5×10^{5}	2.8×10^{6}
12	201	14.8	135/128	14.7	$<1.0 \times 10^{4}$	1.5×10^{5}
13	201	14.8	135/128	14.7	1.5×10^{5}	2.8×10^{6}
14	201	14.8	135/128	14.7	9.1×10^{4}	3.9×10^{5}

Table 3.11-2 (Continued)

ENVIRONMENTAL ZONE PARAMETERS FOR POST-ACCIDENT CONDITIONS

	HELB		LOCA			
Zone	Temperature (°F)	Pressure (psia)	Temperature[12] (°F)	Pressure (psia)	1-Hour Dose (rads)	30-Day Dose (rads)
15[(3)]	201	14.8	135/128	14.7	3.6×10^5	2.0×10^6
16	201	14.8	135/128	14.7	2.2×10^5	8.1×10^5
17	201	14.8	135/128	14.7	2.1×10^5	8.1×10^5
18	304	27.5	142/169	14.7	1.7×10^5	2.8×10^6
19	120	14.7	120	14.7	$<1.0 \times 10^4$	$<1.0 \times 10^4$
19a	80	14.7	120	14.7	$<1.0 \times 10^4$	$<1.0 \times 10^4$
20	120	14.7	120	14.7	$<1.0 \times 10^4$	$<1.0 \times 10^4$
20a	200	16.5	120	14.7	$<1.0 \times 10^4$	$<1.0 \times 10^4$
21	338	40.0	295	63.0	2.8×10^7	1.4×10^8
22	275	45.0	150/156	14.7	2.1×10^5	8.1×10^5
23	212	14.8	135/133	14.7	$<1.0 \times 10^4$	$<1.0 \times 10^4$
24	212	14.8	135/133	14.7	1.4×10^5	2.8×10^6
25	299	15.1	150	14.7	6.2×10^6	4.9×10^7
26	212	14.8	135/133	14.7	1.5×10^5	2.8×10^6
27	212	14.8	135/133	14.7	3.6×10^5	1.8×10^6
28	256	32.7	130/131	14.7	1.3×10^5	2.7×10^6

Table 3.11-2 (Continued)

ENVIRONMENTAL ZONE PARAMETERS FOR POST-ACCIDENT CONDITIONS

	HELB		LOCA			
Zone	Temperature (°F)	Pressure (psia)	Temperature[12] (°F)	Pressure (psia)	1-Hour Dose (rads)	30-Day Dose (rads)
29	200	16.5	120	14.7	$<1.0 \times 10^4$	$<1.0 \times 10^4$
30	120	14.7	120	14.7	$<1.0 \times 10^4$	1.3×10^7
30a	95	14.7	95	14.7	$<1.0 \times 10^4$	$<1.0 \times 10^4$
31	200	16.5	125	14.7	1.5×10^5	2.8×10^6
31a	120	14.7	120	14.7	$<1.0 \times 10^4$ (9)	$<1.0 \times 10^4$ (9)
32	338	40.0	295	63.0	2.8×10^7	1.4×10^8
33	299	15.1	150	14.7	6.7×10^5	3.5×10^6
34	210	14.7	136/127	14.7	$<1.0 \times 10^4$	$<1.0 \times 10^4$
35	210	14.7	136/127	14.7	$<1.0 \times 10^4$	$<1.0 \times 10^4$
36	210	14.7	136/127	14.7	7.3×10^5	3.9×10^6
37	210	14.7	136/127	14.7	7.7×10^5	4.1×10^6
38	120	14.7	120	14.7	$<1.0 \times 10^4$	$<1.0 \times 10^4$
39(4)	120	14.7	120	14.7	3.6×10^5	2.0×10^6
40	104	14.7	136/127	14.7	$<1.0 \times 10^4$	$<1.0 \times 10^4$
41	338	40.0	295	63.0	2.8×10^7	1.4×10^8
42	292	14.8	130/117	14.7	6.4×10^6	4.9×10^7

Table 3.11-2 (Continued)

ENVIRONMENTAL ZONE PARAMETERS FOR POST-ACCIDENT CONDITIONS

	HELB		LOCA			
Zone	Temperature (°F)	Pressure (psia)	Temperature[12] (°F)	Pressure (psia)	1-Hour Dose (rads)	30-Day Dose (rads)
43	104	14.7	118/117	14.7	$<1.0 \times 10^4$	$<1.0 \times 10^4$
44[14]	104	14.7	118/117	14.7	$<1.0 \times 10^4$	$<1.0 \times 10^4$
45	104	14.7	115/115	14.7	$<1.0 \times 10^4$	$<1.0 \times 10^4$
46	120	14.7	120	14.7	$<1.0 \times 10^4$	$<1.0 \times 10^4$
47	291	16.0	120	14.7	$<1.0 \text{ X } 10^4$	$<1.0 \text{ X } 10^4$

Note:
1. Note Deleted
2. For components in HPCI system, dose is $<1.0 \times 10^4$ rads because HPCI operates following design basis events not associated with rapid depressurization and fuel failure.
3. Zone 15 is a 5-foot zone around 18-inch line 1603-18".
4. Zone 39 is a 5-foot zone around 18-inch line 7504-18".
5. Values shown are based on worst case corner room conditions following a HPCI HELB. Analysis is for a HPCI HELB in the HPCI Room with concurrent LOOP. Value applies to Unit 2 only, Unit 3 is affected to a much lesser extent due to physical distance from the HPCI Room.
6. Calculation DRE 01-0041, Rev. 0 predicted a peak post LOCA temperature of 190°F in this room without room coolers.
7. Values shown are based on worst case corner room conditions following a HPCI HELB. Analysis is for a HPCI HELB in the HPCI Room with concurrent LOOP. Value applies to Unit 3 only, Unit 2 is affected to a much lesser extent due to physical distance from the HPCI Room.

Table 3.11-2 (Continued)

ENVIRONMENTAL ZONE PARAMETERS FOR POST-ACCIDENT CONDITIONS

8. Temperature of the adjacent HPCI room of the other Unit will be 176 °F. A LOOP is considered concurrent with a HPCI line break in the HPCI room.
9. The portion of the zone on the north side of the turbine building considers a post accident radiation source (2/3-7509-24" containment air pipe) and could see higher doses than other parts of Zone 31A.
10. This temperature is with HPCI Room cooler running. Calculation DRE98-0077, Rev. 01c predicted a peak 4-hour temperature of 134°F in this zone without room cooler.
11. Section 9.4.3.1. reports design temperatures of 103°F in occupied areas, 120°F in cells and collector tank room and 150°F in the concentrator and concentrator tank waste cells.
12. Where a single value is given, it applies to the LOCA unit; the non-LOCA unit remains at normal conditions. Where two values are given, it represents the data for LOCA unit followed by the data for non-LOCA unit.
13. Equipment in this zone located below grade (505'-6") may be submerged as a result of a break in the Main Staem line or Feedwater line anywhere in this tunel. For existing installations, Section 3.6.1.1.4.3 states that no safety related equipment or wiring would be endangered by potential flooding.
14. The portion of zone 44 that is north of Column H is a part of the Turbine Building and has the following conditions for both LOCA and HELB: 120°F and 14.7 psia.

Table 3.11-3

RELATIVE HUMIDITY BY ENVIRONMENTAL ZONE

Zone	Relative Humidity (%) Normal	Spurious	Accident
1[3]	20—90	5—100	100(C)
2	20—90	5—100	100(C)
3	20—90	5—100	100(C)
4	20—90	5—100	100(C)
5	20—90	5—100	100(C)
6	20—90	5—100	100(C)
7[1]	20—90	5—100	100(NC)
7a[1]	20—90	5—100	100(NC)
7b[1]	20—90	5—100	100(NC)
8[3]	20—90	5—100	100(C)
9	20—90	5—100	100(C)
9a	20—90	5—100	100(C)
10[2]	20—90	5—100	100(C)
11[2]	20—90	5—100	100(C)
12[2]	20—90	5—100	100(C)
13[2]	20—90	5—100	100(C)
14[2]	20—90	5—100	100(C)
15[2]	20—90	5—100	100(C)
16[2]	20—90	5—100	100(C)
17[1]	20—90	5—100	100(C)
18[1]	20—90	5—100	100(C)
19[1]	20—90	5—100	100(NC)
19a	40—70	20—90	90
20[1]	20—90	5—100	100(NC)
20a[1]	20—90	5—100	100(NC)
21[3]	20—90	5—100	100(C)

Table 3.11-3 (Continued)

RELATIVE HUMIDITY BY ENVIRONMENTAL ZONE

Zone	Relative Humidity (%) Normal	Spurious	Accident
22	20—90	5—100	100(C)
23	20—90	5—100	100(C)
24[1]	20—90	5—100	100(NC)
25	20—90	5—100	100(C)
26[1]	20—90	5—100	100(NC)
27[1]	20—90	5—100	100(NC)
28	20—90	5—100	100(C)
29	20—90	5—100	100(C)
30[1]	20—90	5—100	100(NC)
30a	40—70	20—90	90
31	20—90	5—100	100(C)
31a[1]	20—90	5—100	100(NC)
32[3]	20—90	5—100	100(C)
33	20—90	5—100	100(C)
34[1]	20—90	5—100	100(NC)
35[1]	20—90	5—100	100(NC)
36[1]	20—90	5—100	100(NC)
37[1]	20—90	5—100	100(NC)
38[1]	20—90	5—100	100(NC)
39[1]	20—90	5—100	100(NC)
40[1]	20—90	5—100	100(NC)
41[3]	20—90	5—100	100(C)
42	20—90	5—100	100(C)

Table 3.11-3 (Continued)

RELATIVE HUMIDITY BY ENVIRONMENTAL ZONE

Zone	Relative Humidity (%) Normal	Spurious	Accident
43[1]	20—90	5—100	100(NC)
44[1]	20—90	5—100	100(NC)
45[1]	20—90	5—100	100(NC)
46[1]	20—90	5—100	100(NC)
47	20—90	5—100	100(C)[4]

Notes:

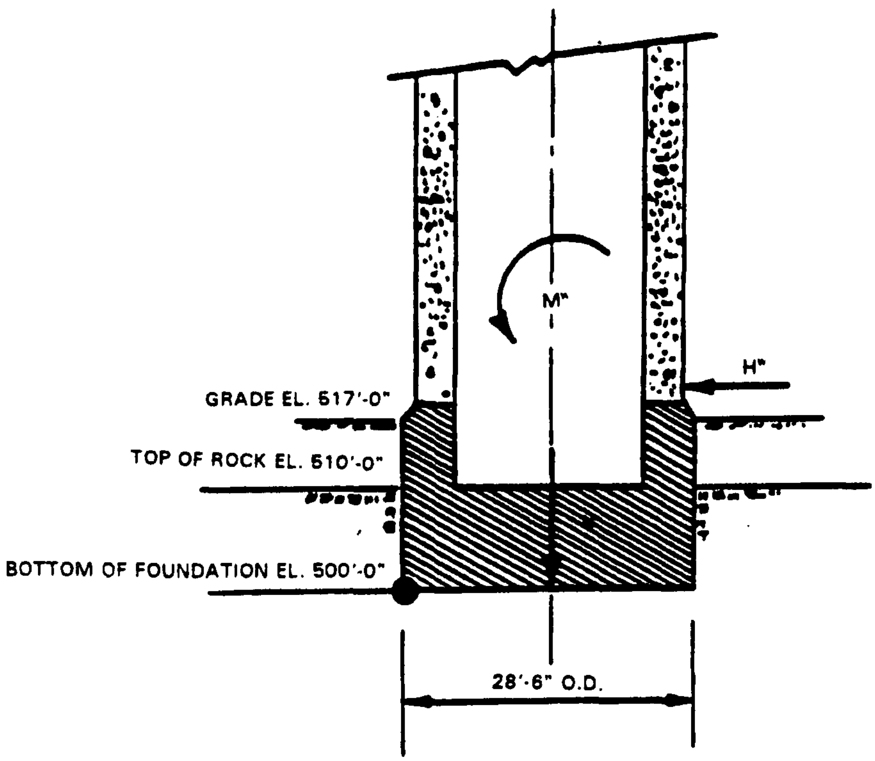
(C) = Condensing (NC) = Noncondensing

1. Maximum relative humidity for individual components containing an equipment heat source is 95%. Maximum relative humidity for area without a moisture source is 95%.

2. Maximum relative humidity for motor control centers and bus ducts in this zone is 95%.

3. Equipment in the drywell may be subject to drywell sprays of demineralized water.

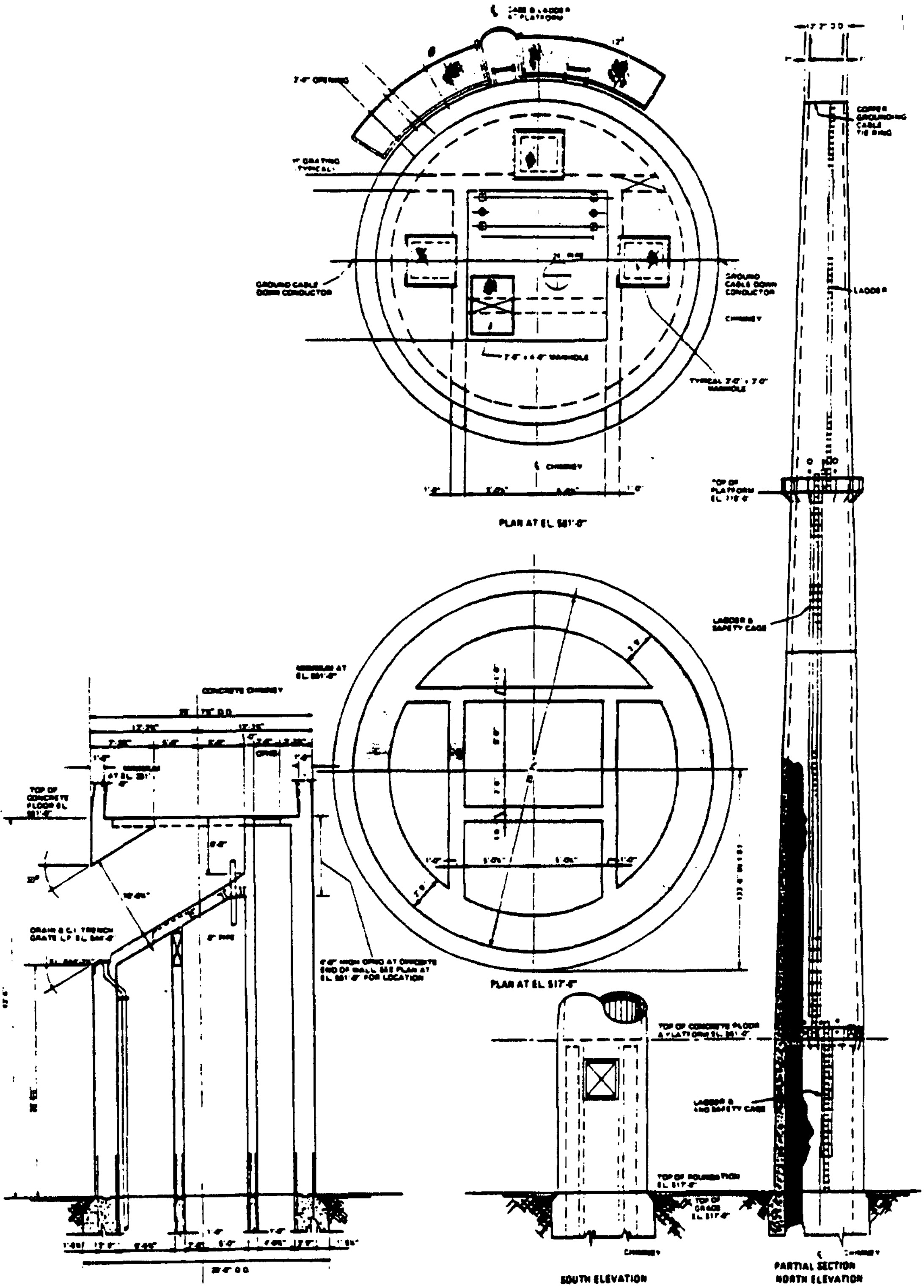
4. The accident relative humidity is 100%, Condensing only for the HPCI steam line break within the HPCI room. For all other scenarios, the accident relative humidity is 100%, Noncondensing.



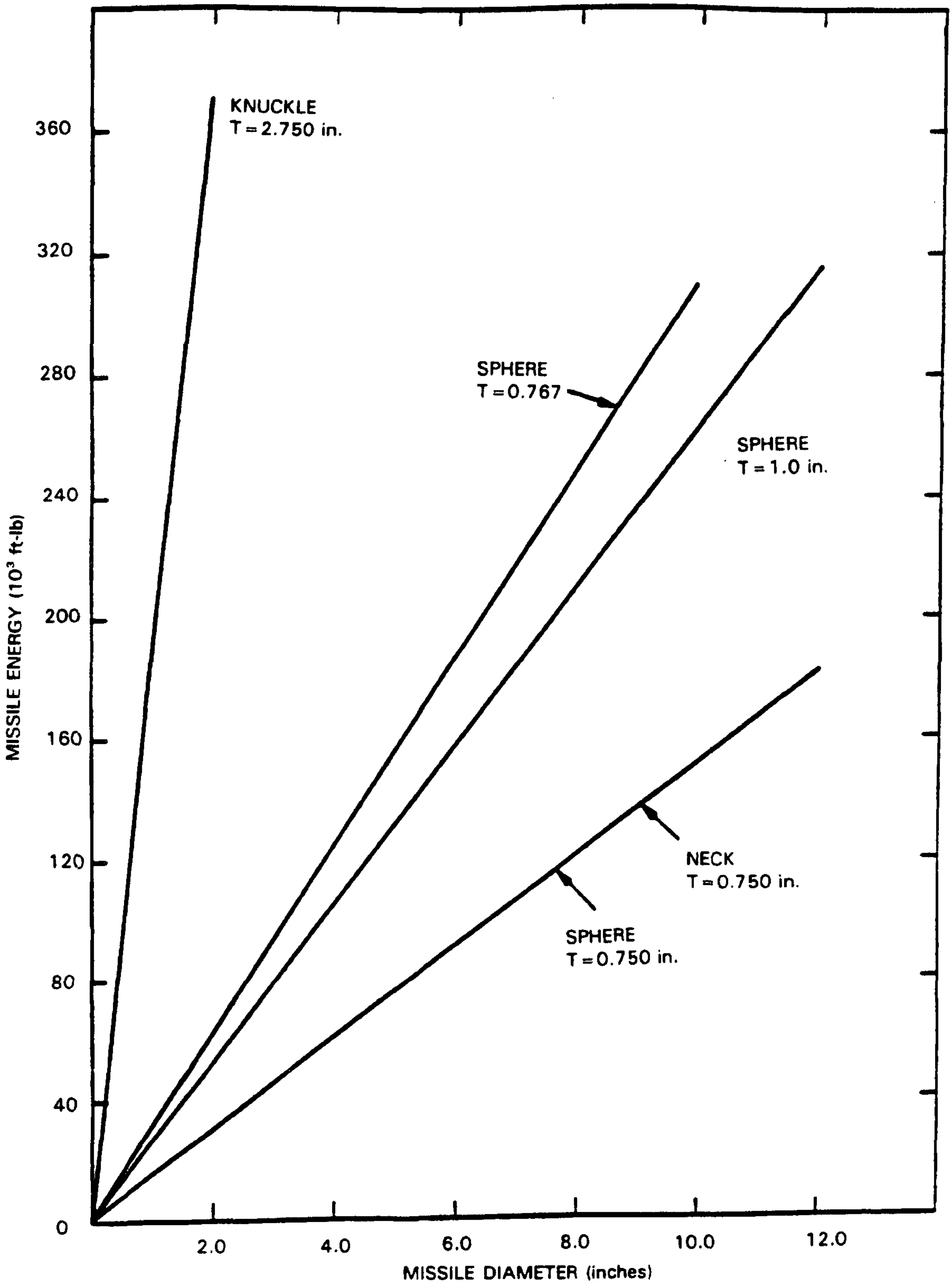
DRESDEN STATION
UNITS 2 & 3

VENTILATION CHIMNEY
OVERTURNING MODEL

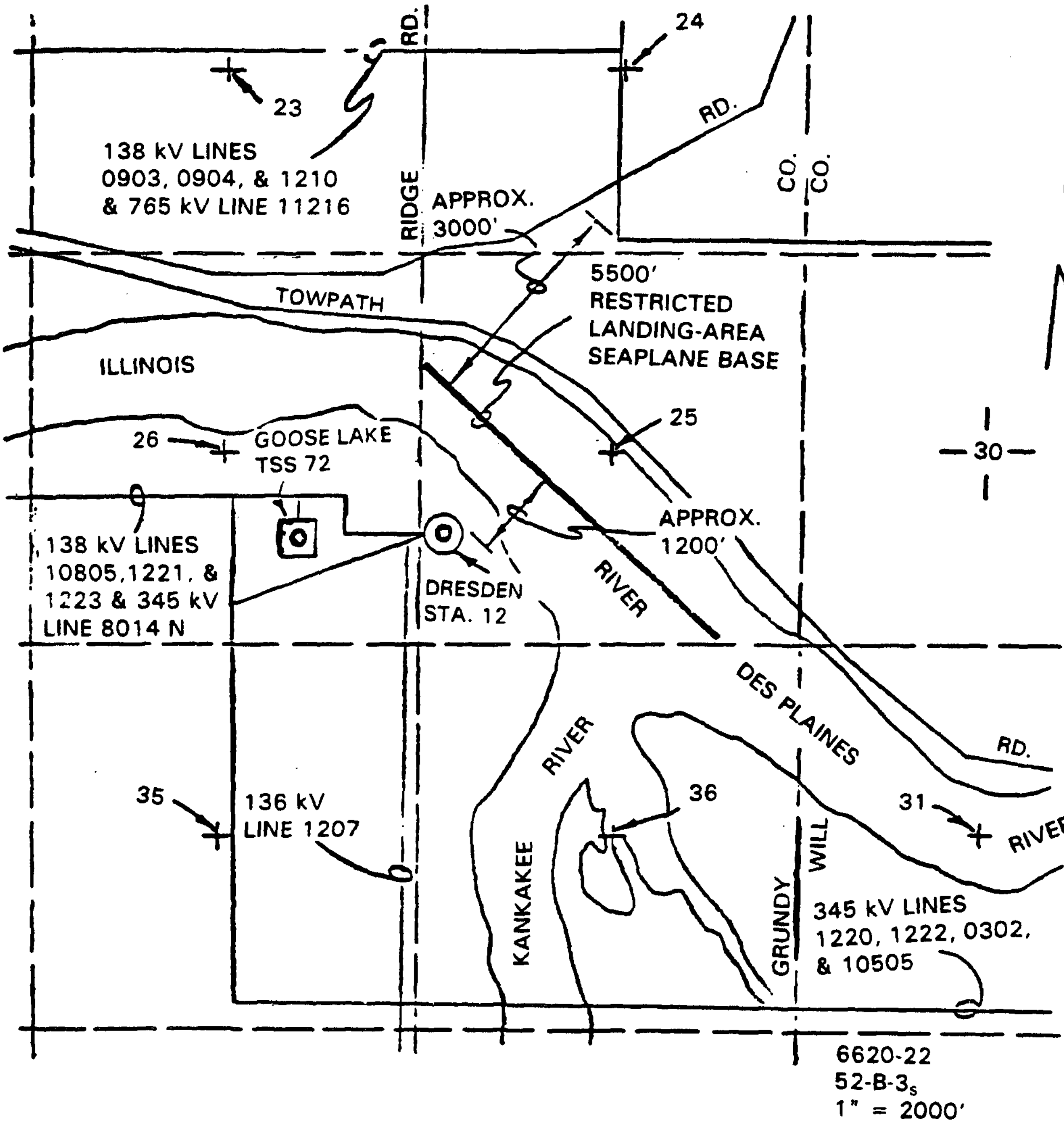
FIGURE 3.3-1



DRESDEN STATION UNITS 2 & 3
VENTILATION CHIMNEY, ELEVATIONS, AND SECTIONS
FIGURE 3.3-2



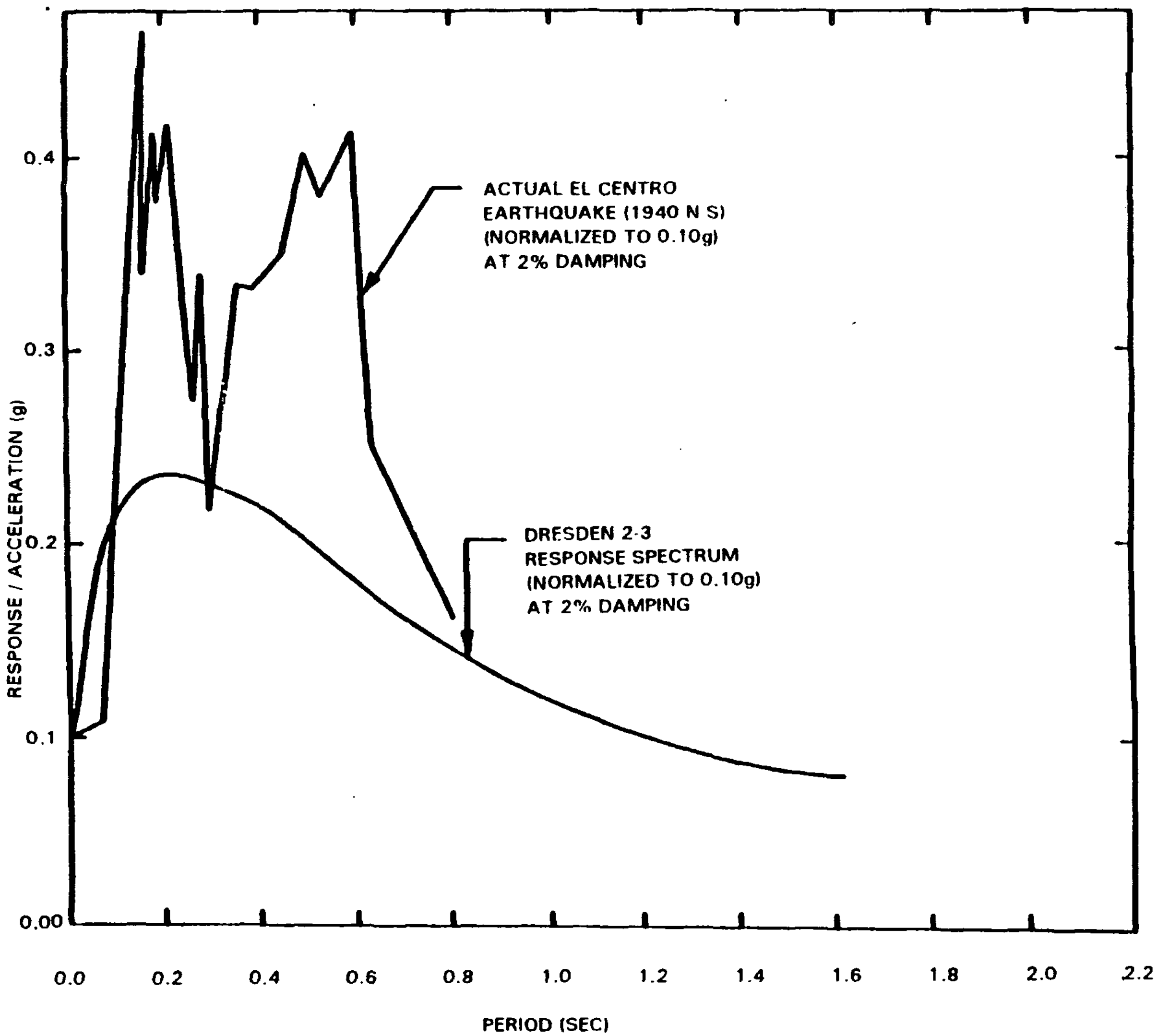
DRESDEN STATION UNITS 2 & 3
ENERGY NECESSARY TO PENETRATE DRYWELL CONTAINMENT
FIGURE 3.5-1



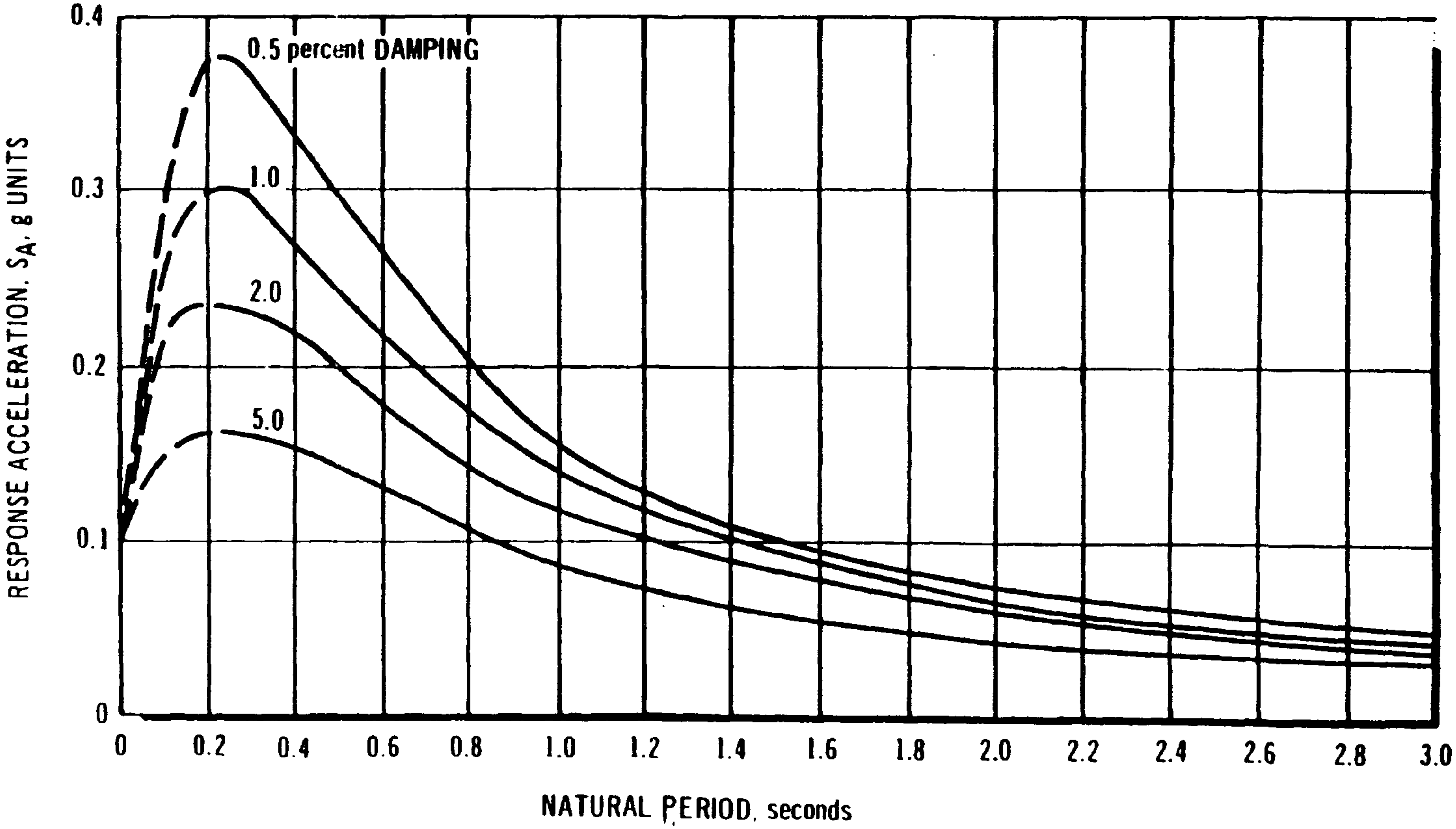
DRESDEN STATION
UNITS 2 & 3

PROPOSED RESTRICTED LANDING AREA-
SEAPLANE BASE FOR MICHAEL S.
GREENWALD LOCATED NEAR MINOOKA, IL

FIGURE 3.5-2



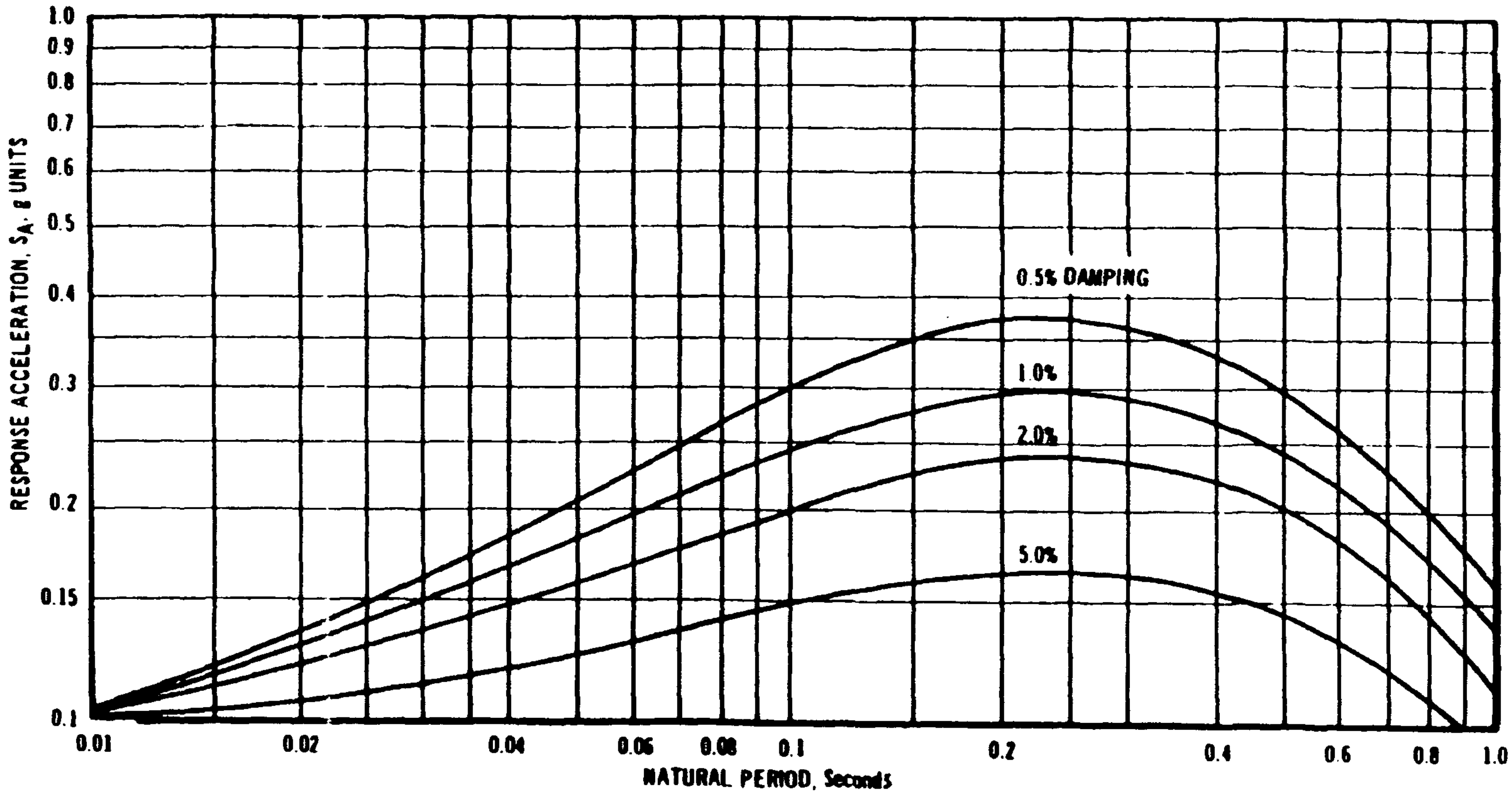
DRESDEN STATION UNITS 2 & 3
RESPONSE SPECTRUM CURVES FOR EL CENTRO EARTHQUAKE
FIGURE 3.7-1



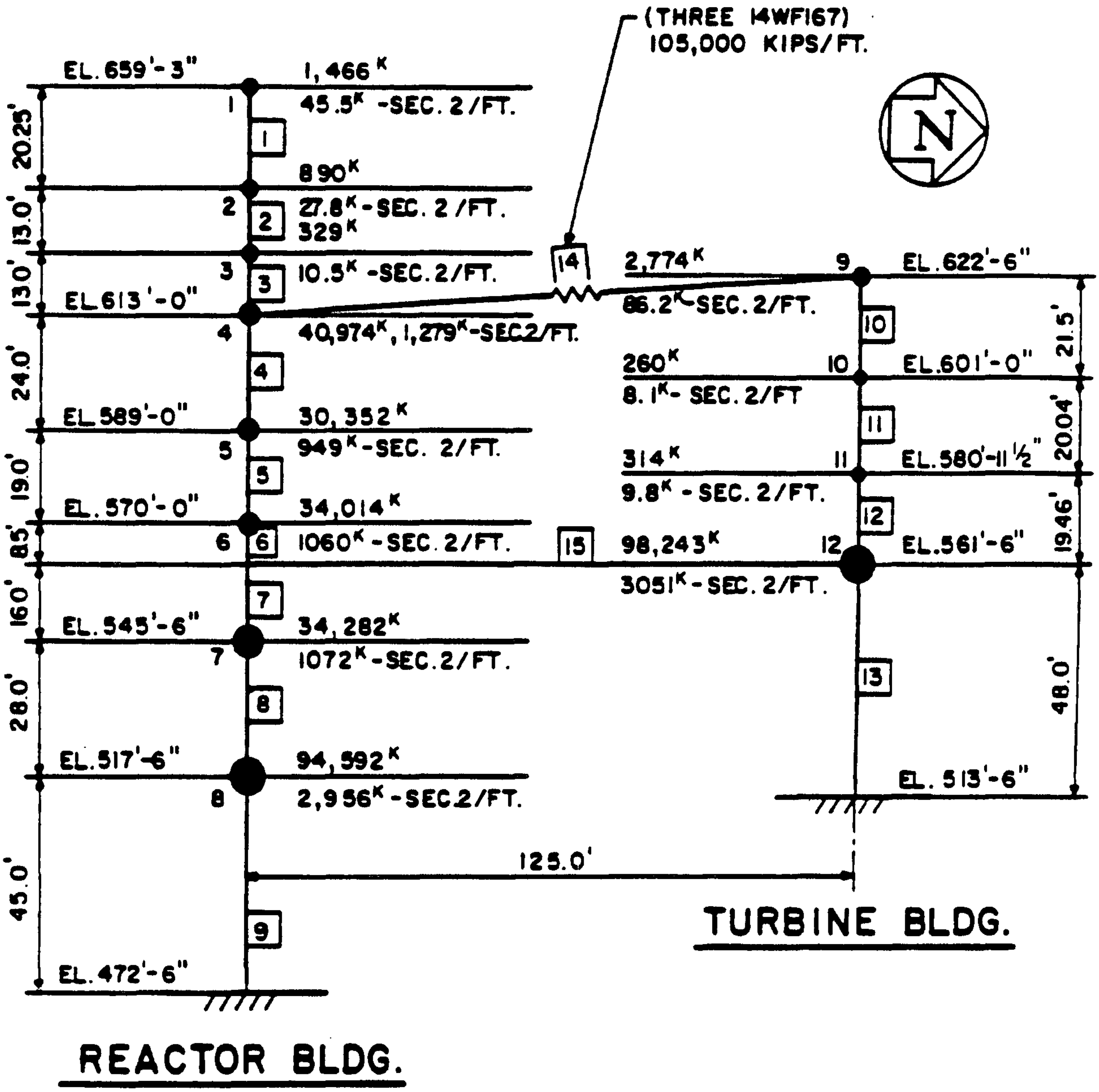
DRESDEN STATION
UNITS 2 & 3

DESIGN ACCELERATION RESPONSE SPECTRUM, LINEAR SCALE

FIGURE 3.7-2



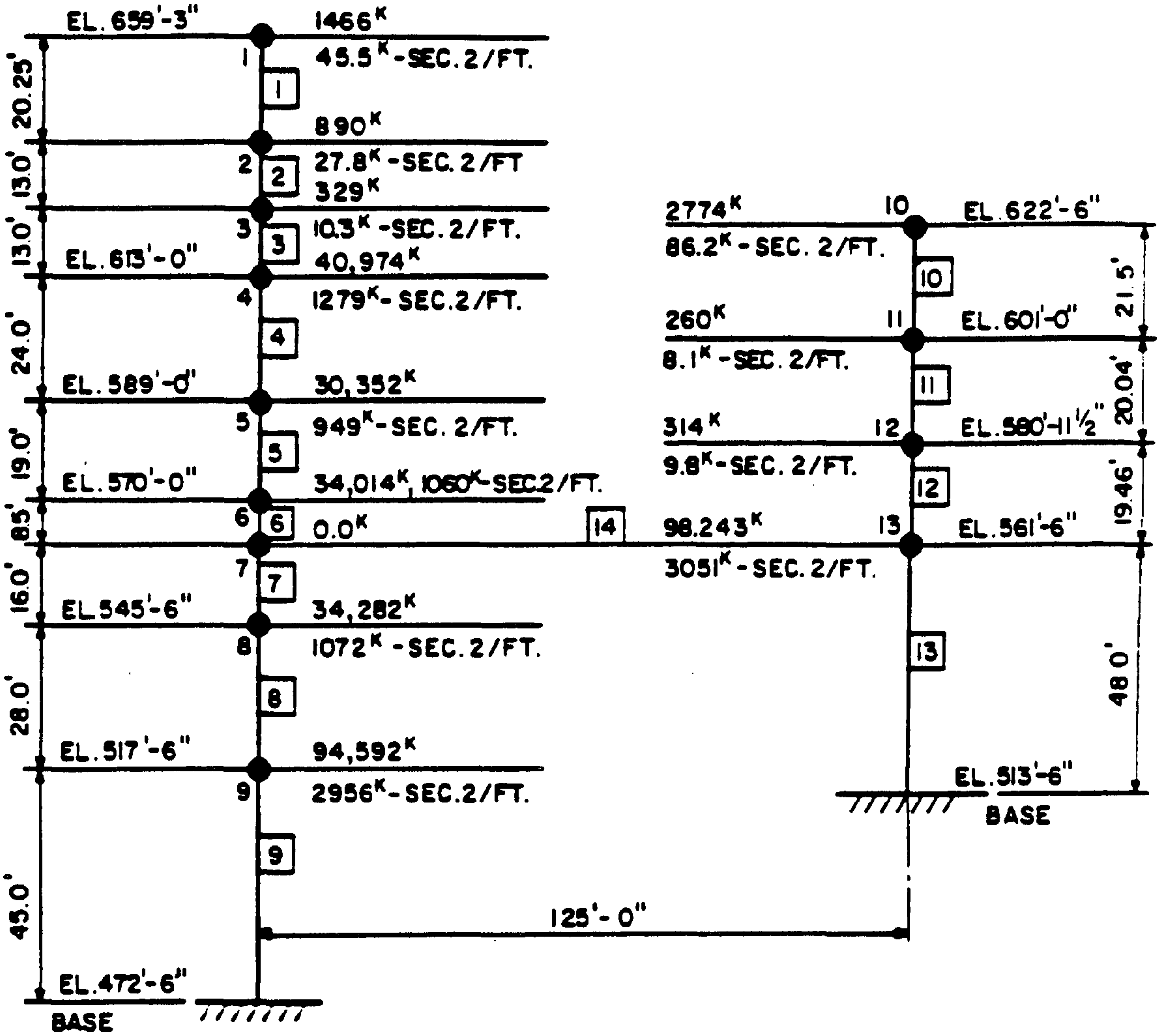
DRESDEN STATION UNITS 2 & 3
DESIGN ACCELERATION RESPONSE SPECTRUM, LOG-LOG SCALE
FIGURE 3.7-3



DRESDEN STATION
UNITS 2 & 3

MATHEMATICAL MODEL. COMBINED REACTOR-TURBINE BUILDING. EARTHQUAKE IN N-S DIRECTION

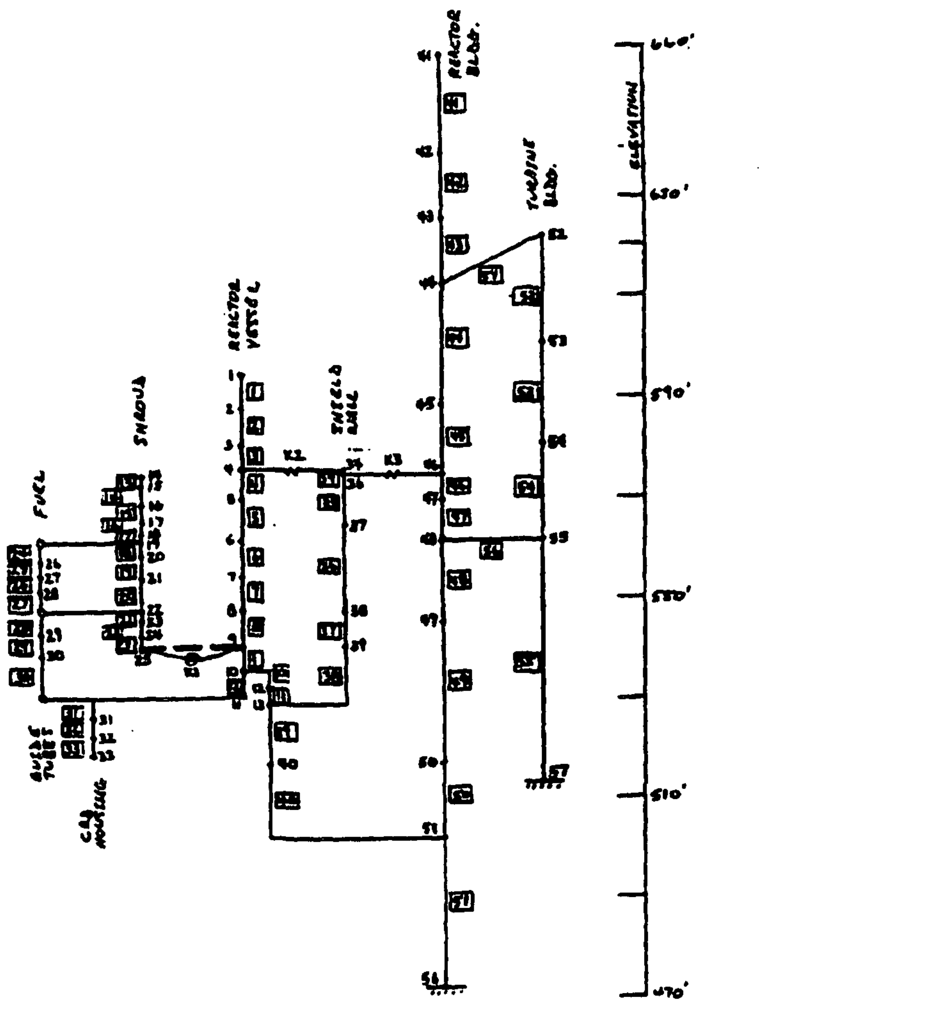
FIGURE 3.7-4



DRESDEN STATION
UNITS 2 & 3

MATHEMATICAL MODEL, COMBINED REACTOR-TURBINE BUILDING, EARTHQUAKE IN E-W DIRECTION

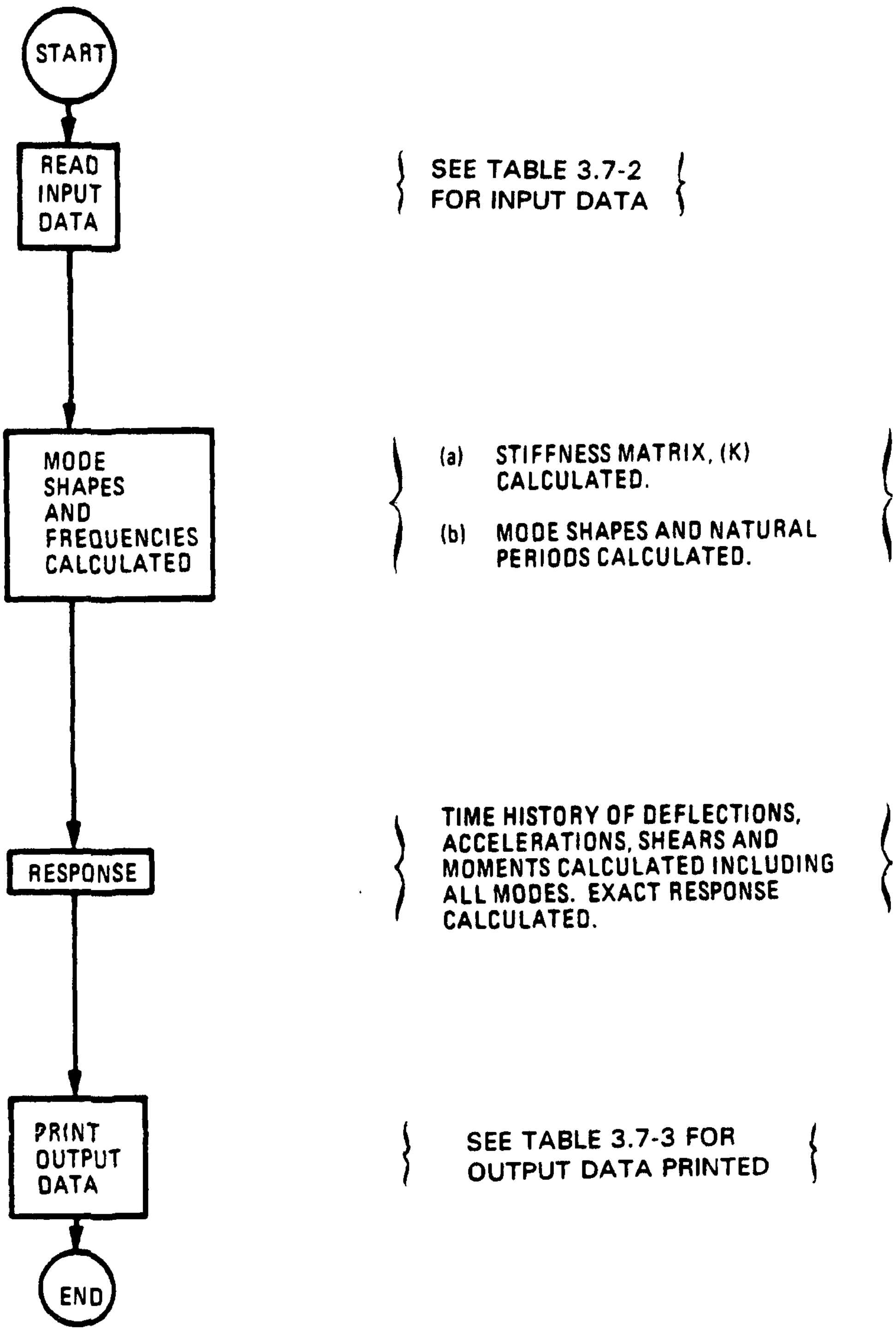
FIGURE 3.7-5



Notes:

1. The details of the modeling of the core shroud repair hardware is considered proprietary information (see GENE 771-84-1194 and section 3.9.5.3.4).

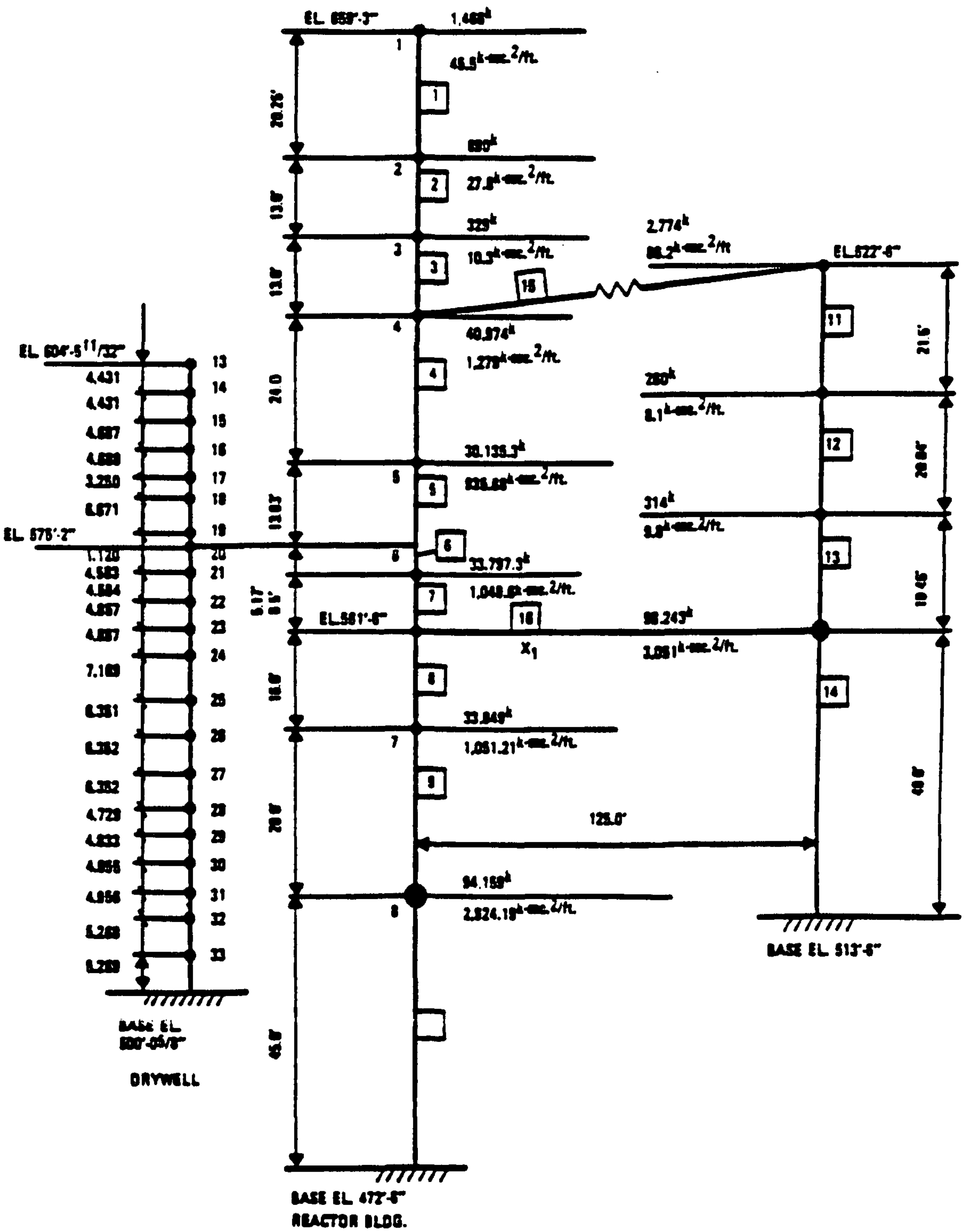
UFSAR REVISION 2, JUNE 1997
DRESDEN STATION UNITS 2 & 3
MATHEMATICAL MODEL, REACTOR PRESSURE VESSEL & INTERNALS
FIGURE 3.7–6



DRESDEN STATION
UNITS 2 & 3

SIMPIFIED BLOCK DIAGRAM OF DYNAMIC COMPUTER PROGRAM

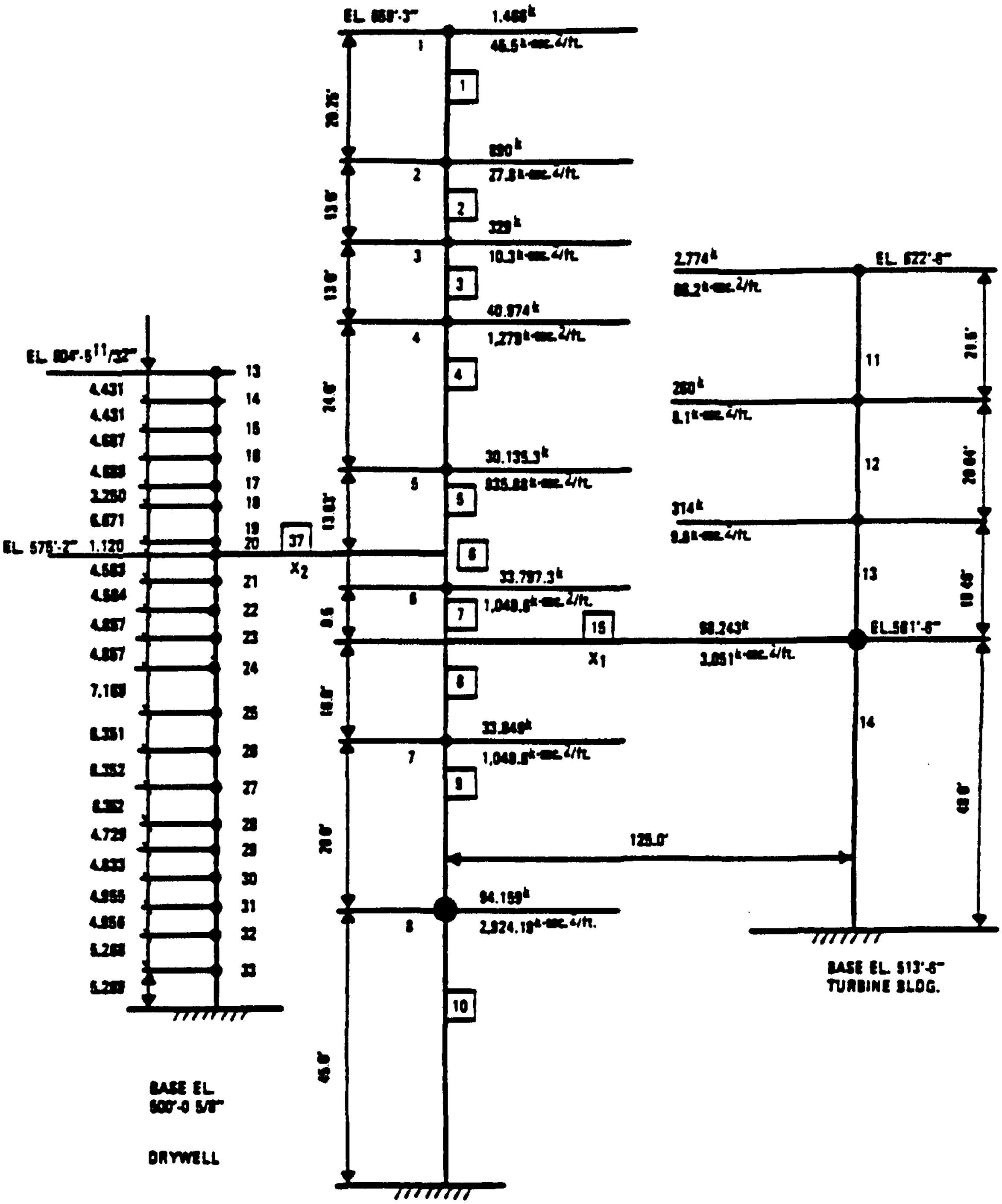
FIGURE 3.7-7



DRESDEN STATION
UNITS 2 & 3

MATHEMATICAL MODEL - EARTHQUAKE IN N-S DIRECTION

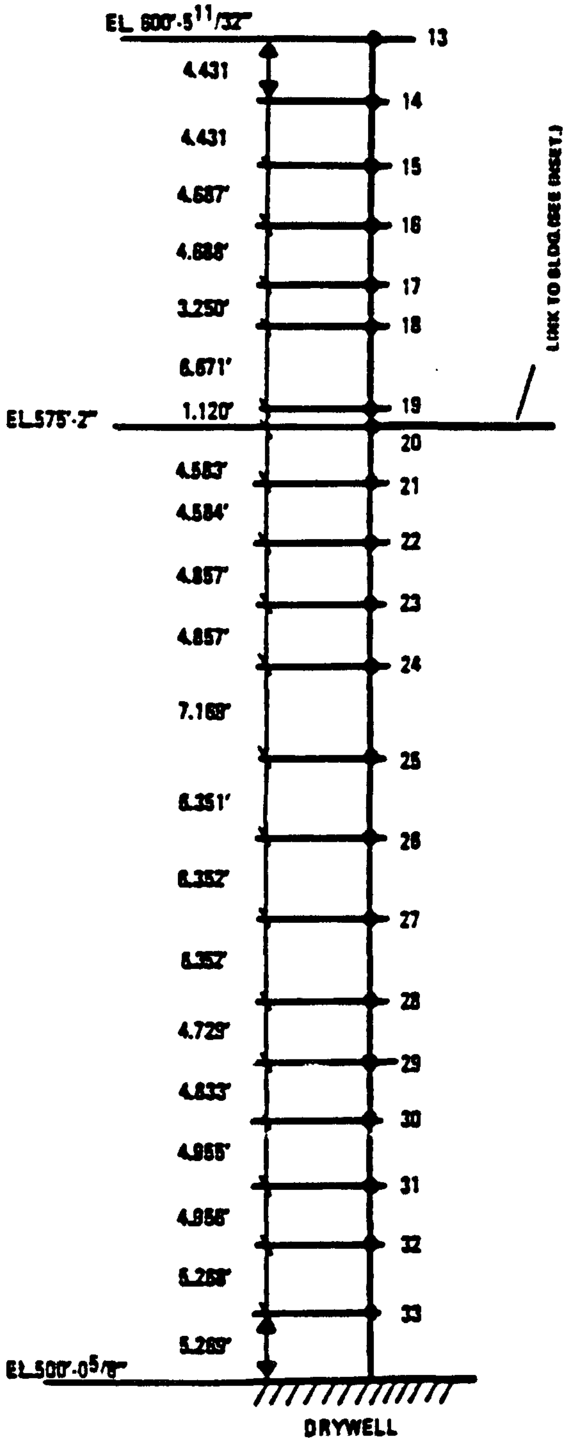
FIGURE 3.7-8

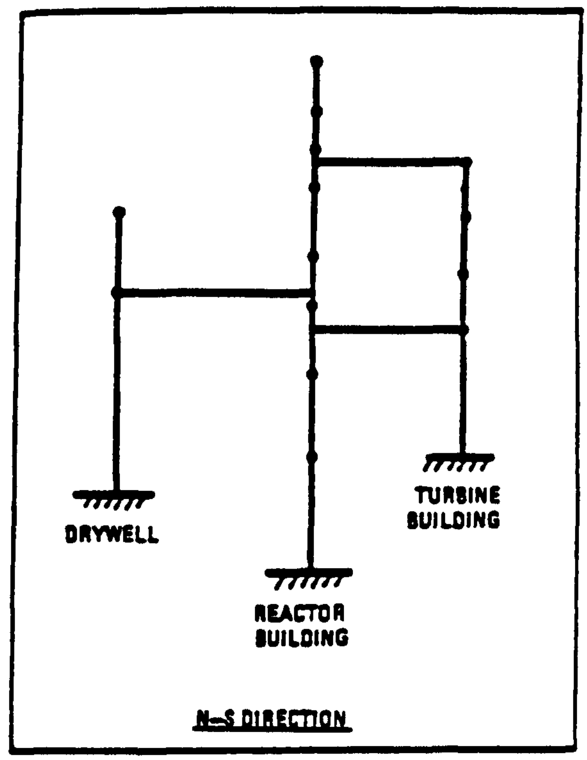


DRESDEN STATION
UNITS 2 & 3

MATHEMATICAL MODEL - EARTHQUAKE IN E-W DIRECTION

FIGURE 3.7-9

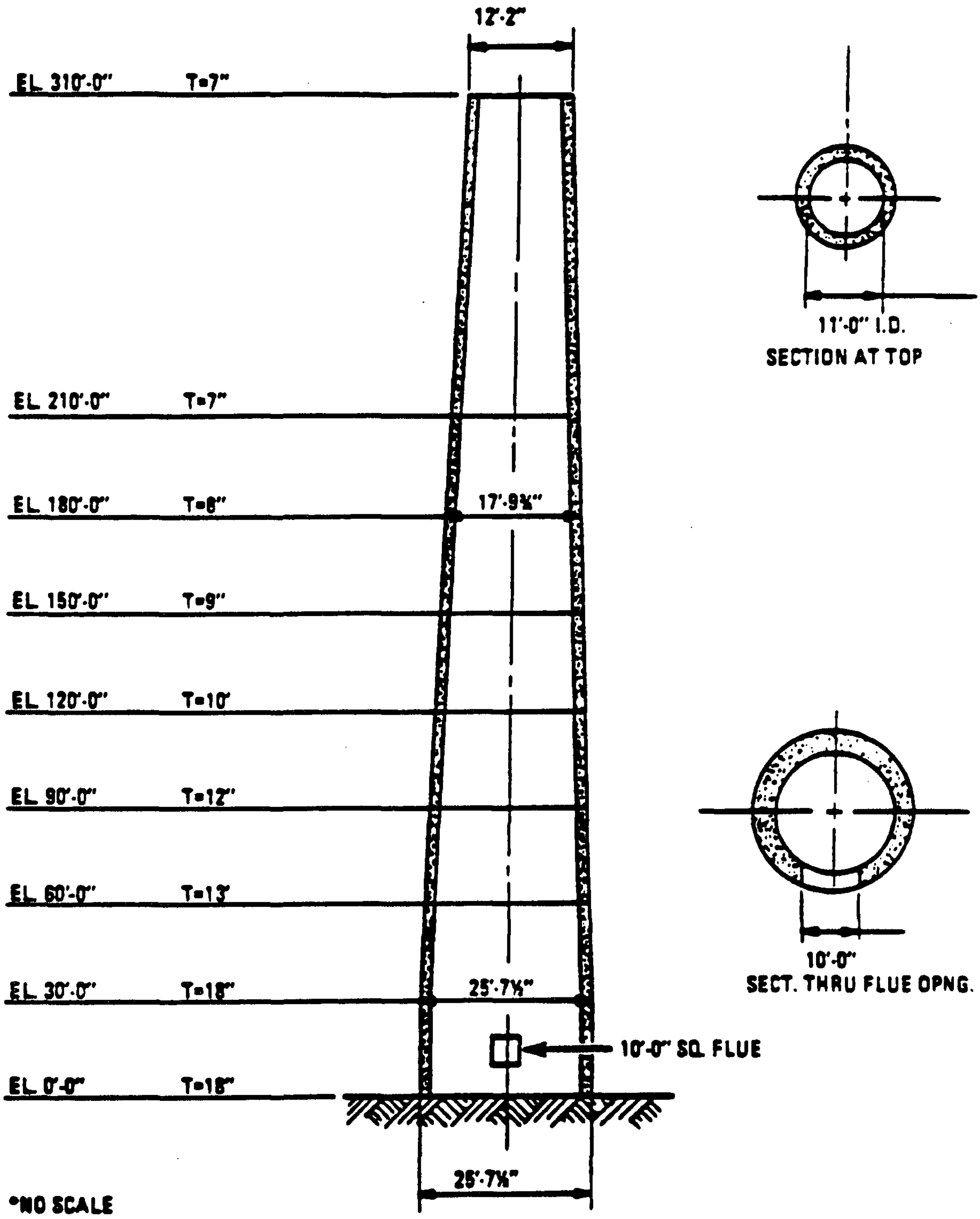




DRESDEN STATION
UNITS 2 & 3

MATHEMATICAL MODEL - DRYWELL

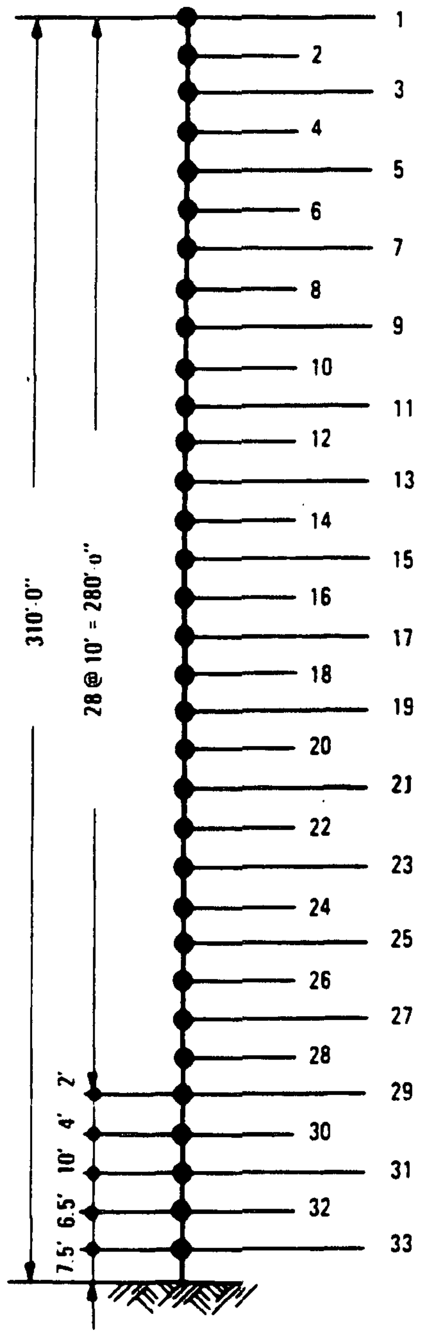
FIGURE 3.7-10



DRESDEN STATION
UNITS 2 & 3

VENTILATION CHIMNEY CONFIGURATION

FIGURE 3.7-11

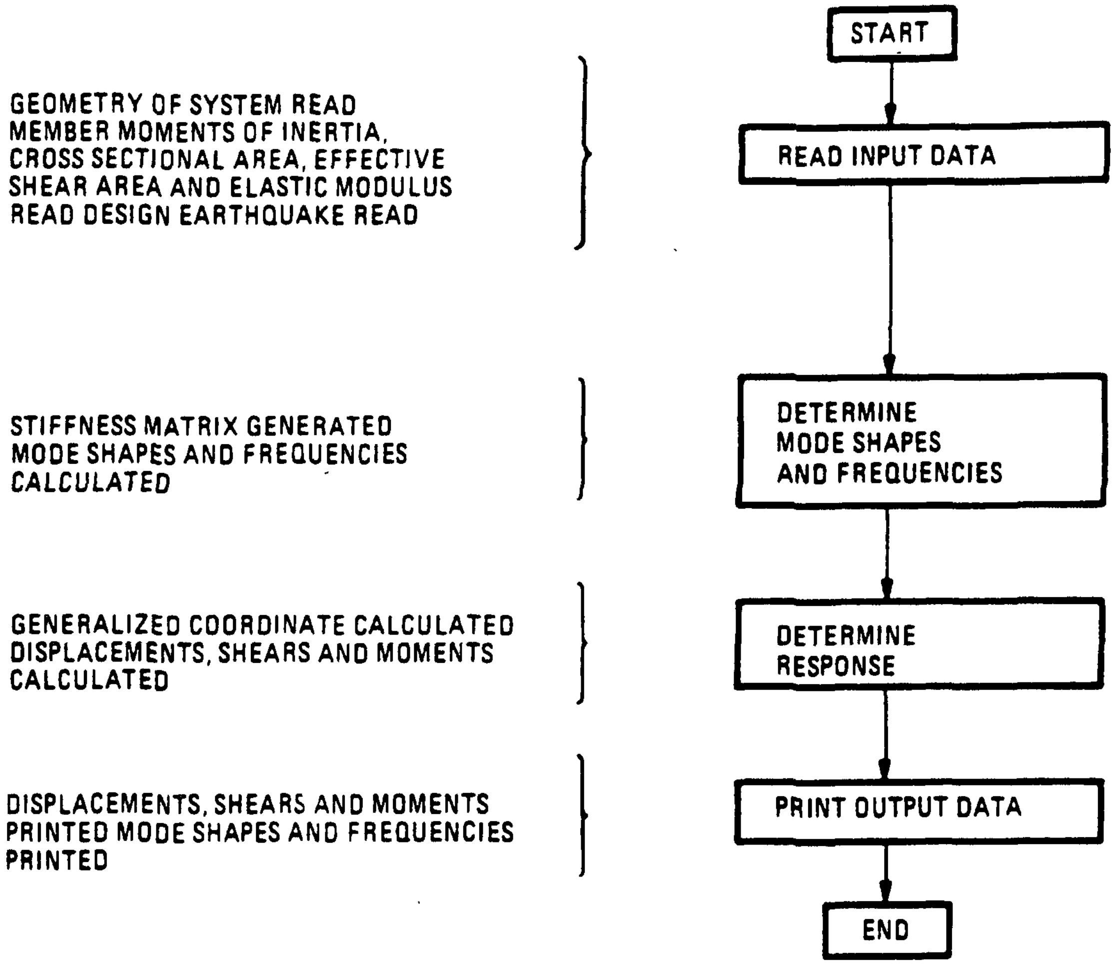


MASS	WEIGHT KIPS	MEMBER	A (ft^2)	I (ft^4)
1	16.4			
2	34.6	1-2	22.7	435
3	35.8	2-3	23.5	483
4	37.0	3-4	24.3	533
5	39.2	4-5	25.1	587
6	39.4	5-6	25.8	645
7	40.6	6-7	26.6	706
8	41.8	7-8	27.4	771
9	42.9	8-9	28.2	840
10	44.1	9-10	29.0	913
11	45.9	10-11	29.8	990
12	48.8	11-12	31.3	1097
13	52.3	12-13	33.7	1239
14	56.0	13-14	36.1	1395
15	60.0	14-15	38.6	1575
16	64.1	15-16	41.3	1786
17	68.5	16-17	44.2	2019
18	72.9	17-18	47.1	2267
19	77.7	18-19	50.1	2545
20	83.1	19-20	53.5	2863
21	87.7	20-21	57.3	3218
22	95.7	21-22	59.7	3518
23	105.3	22-23	67.9	4195
24	111.4	23-24	72.5	4692
25	117.9	24-25	76.2	5159
26	127.5	25-26	81.1	5911
27	140.3	26-27	88.9	6707
28	152.8	27-28	98.1	7668
29	95.8	28-29	105.5	7276
30	50.1	29-30	111.3	7761
31	121.8	30-31	111.3	7761
32	142.7	31-32	96.3	5669
33	111.3	32-33	111.3	7761
		33-BASE	111.3	7761

DRESDEN STATION
UNITS 2 & 3

MATHEMATICAL MODEL - VENTILATION CHIMNEY

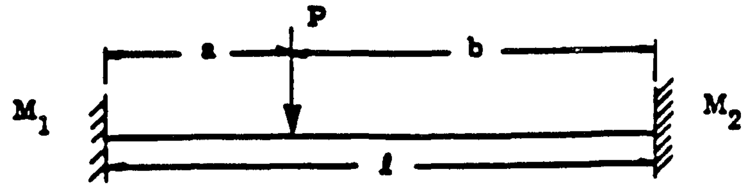
FIGURE 3.7-12



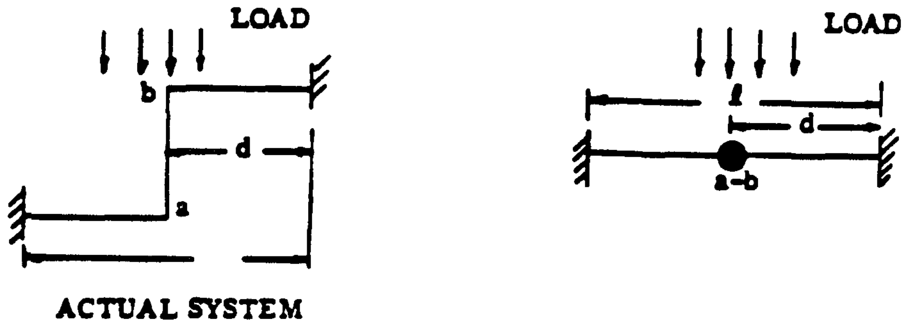
DRESDEN STATION
UNITS 2 & 3

SEISMIC COMPUTER PROGRAM PROCESS
DIAGRAM, VENTILATION CHIMNEY

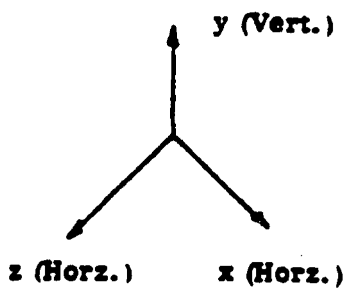
FIGURE 3.7-13



A.



B.

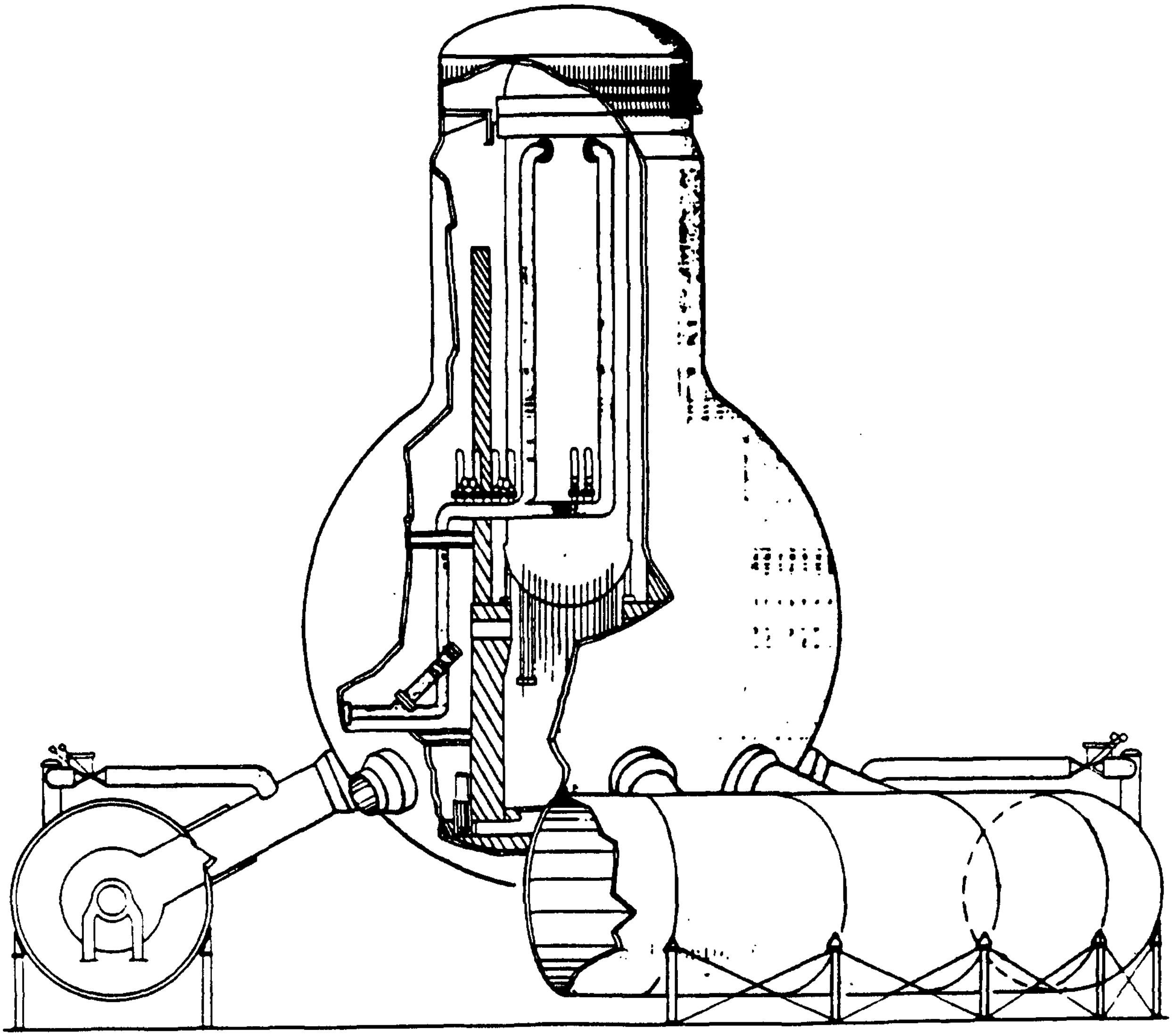


C.

DRESDEN STATION
UNITS 2 & 3

LOAD DIAGRAMS ILLUSTRATING METHOD FOR PIPING STATIC ANALYSIS

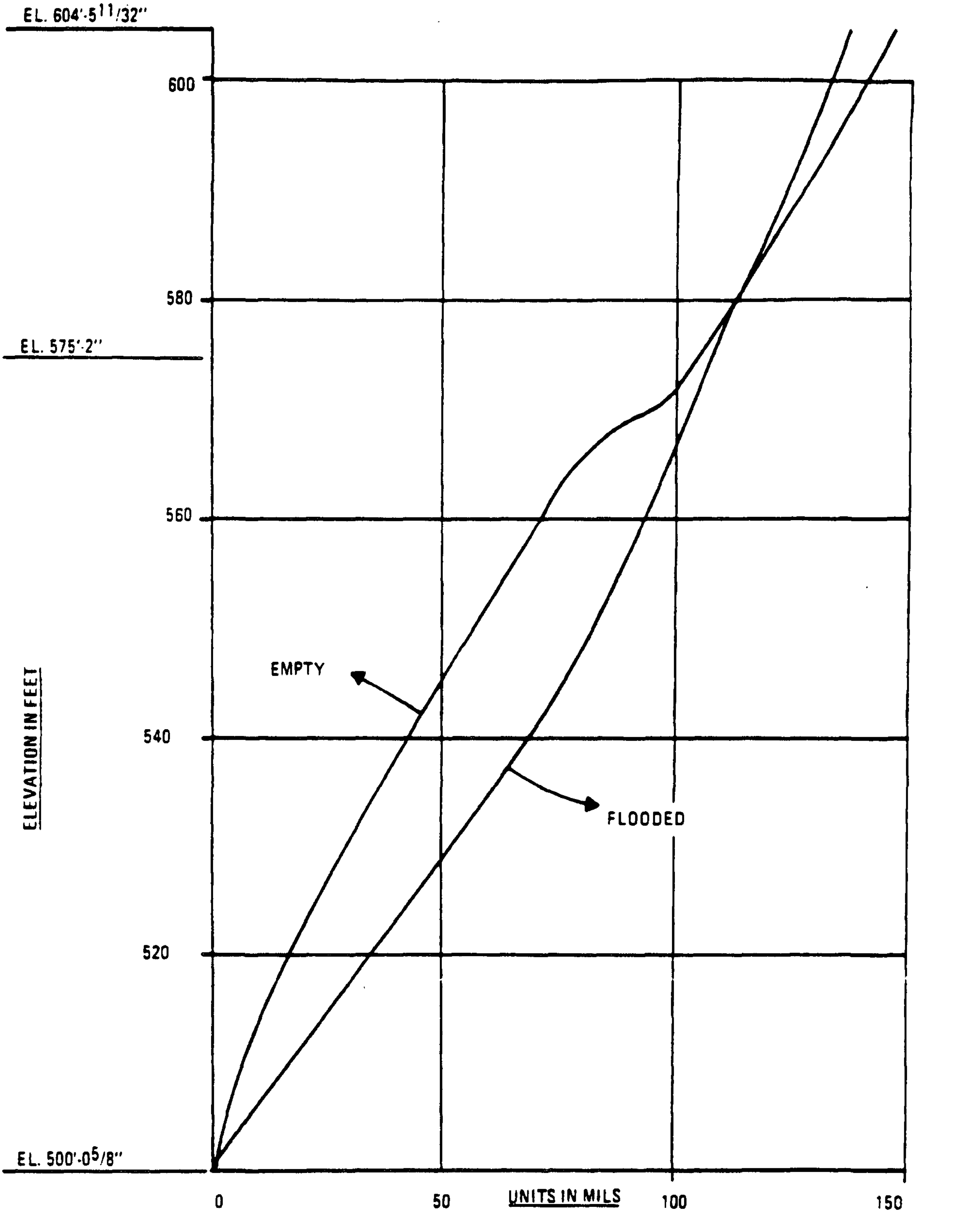
FIGURE 3.7-14



DRESDEN STATION
UNITS 2 & 3

PRESSURE SUPPRESSION CONTAINMENT SYSTEM

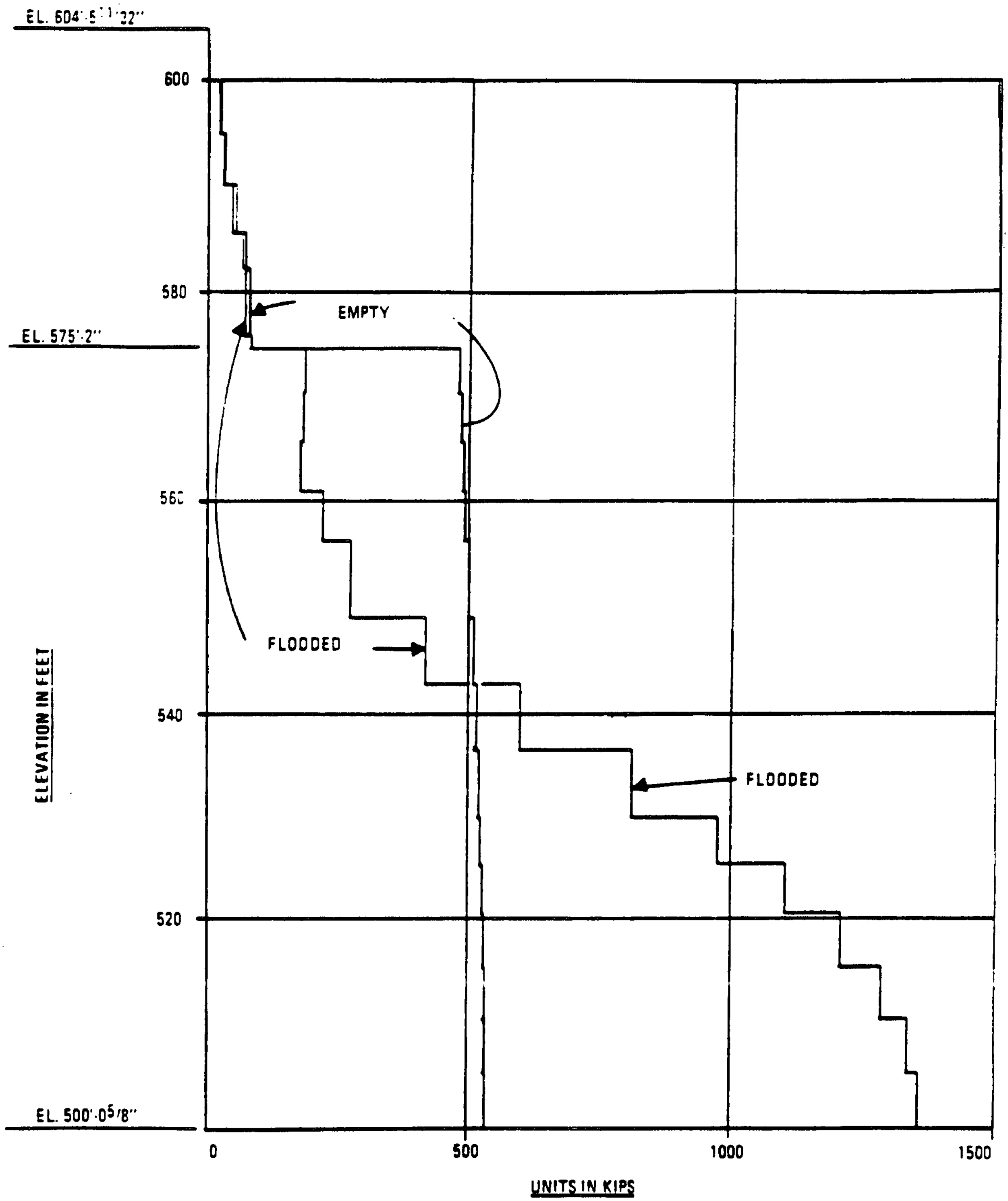
FIGURE 3.8-1



DRESDEN STATION
UNITS 2 & 3

DRYWELL OBE DISPLACEMENT DIAGRAM - N-S DIRECTION

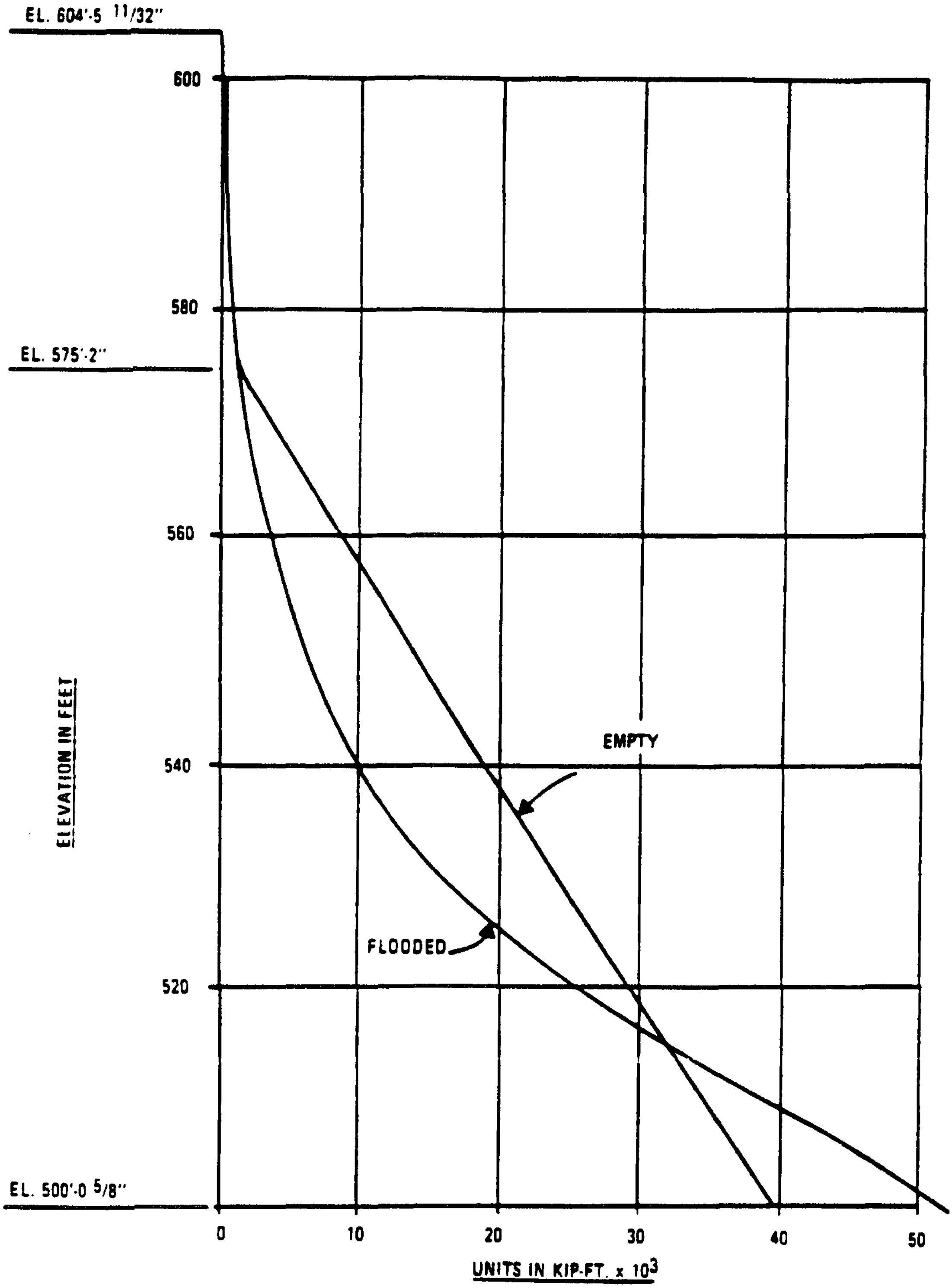
FIGURE 3.8-2



DRESDEN STATION
UNITS 2 & 3

DRYWELL OBE SHEAR DIAGRAM - N-S DIRECTION

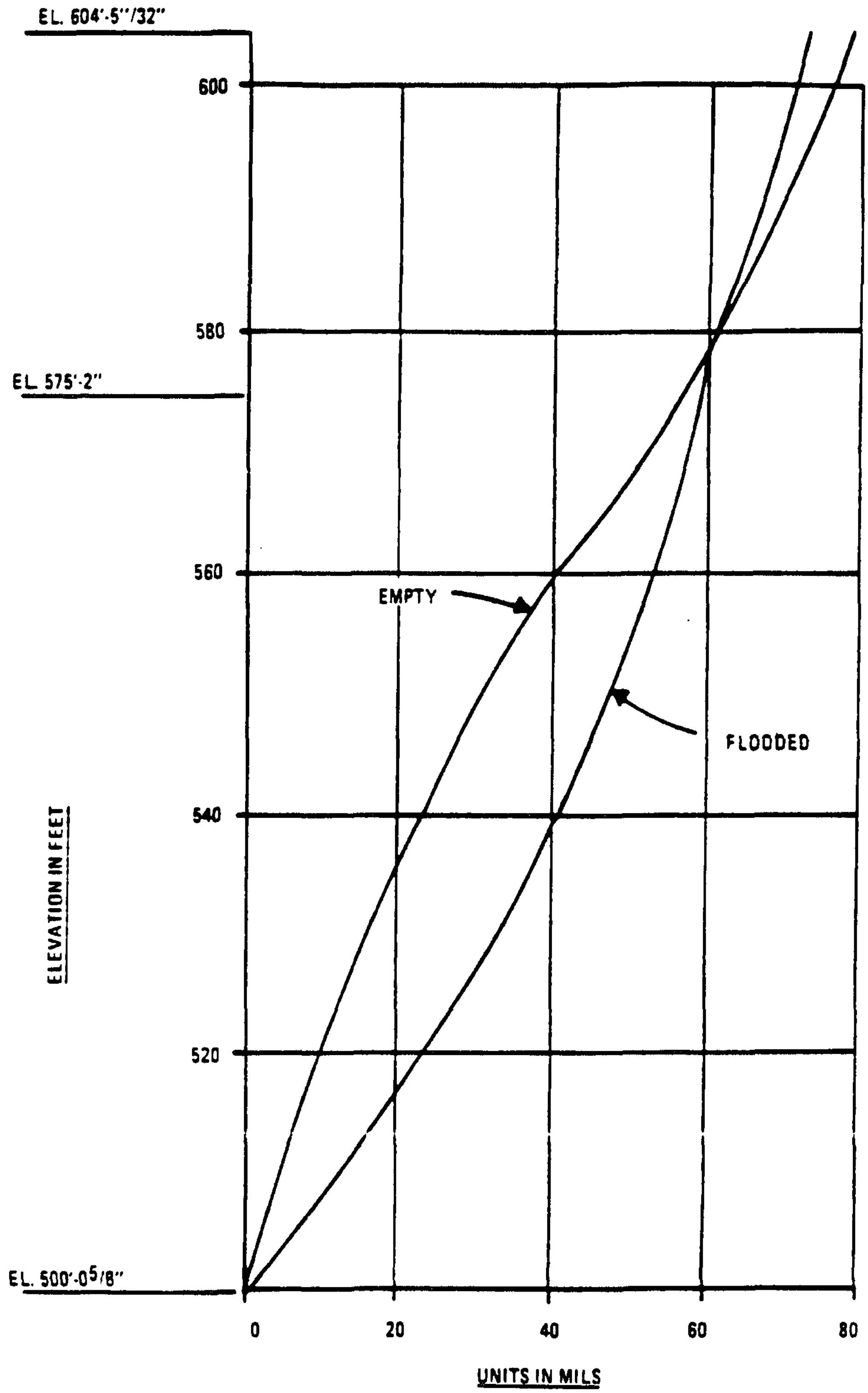
FIGURE 3.8-3



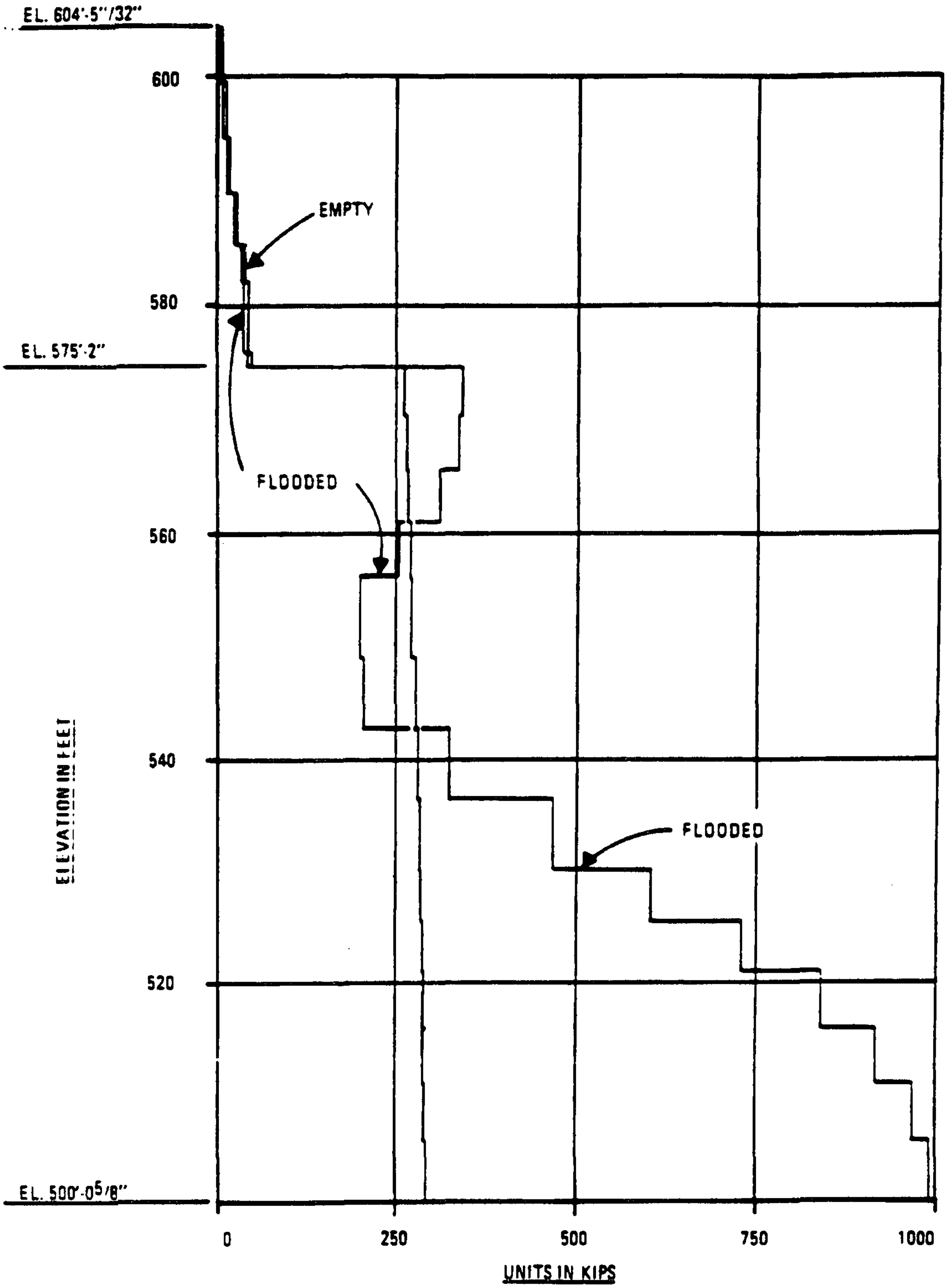
DRESDEN STATION
UNITS 2 & 3

DRYWELL OBE MOMENT DIAGRAM -
N-S DIRECTION

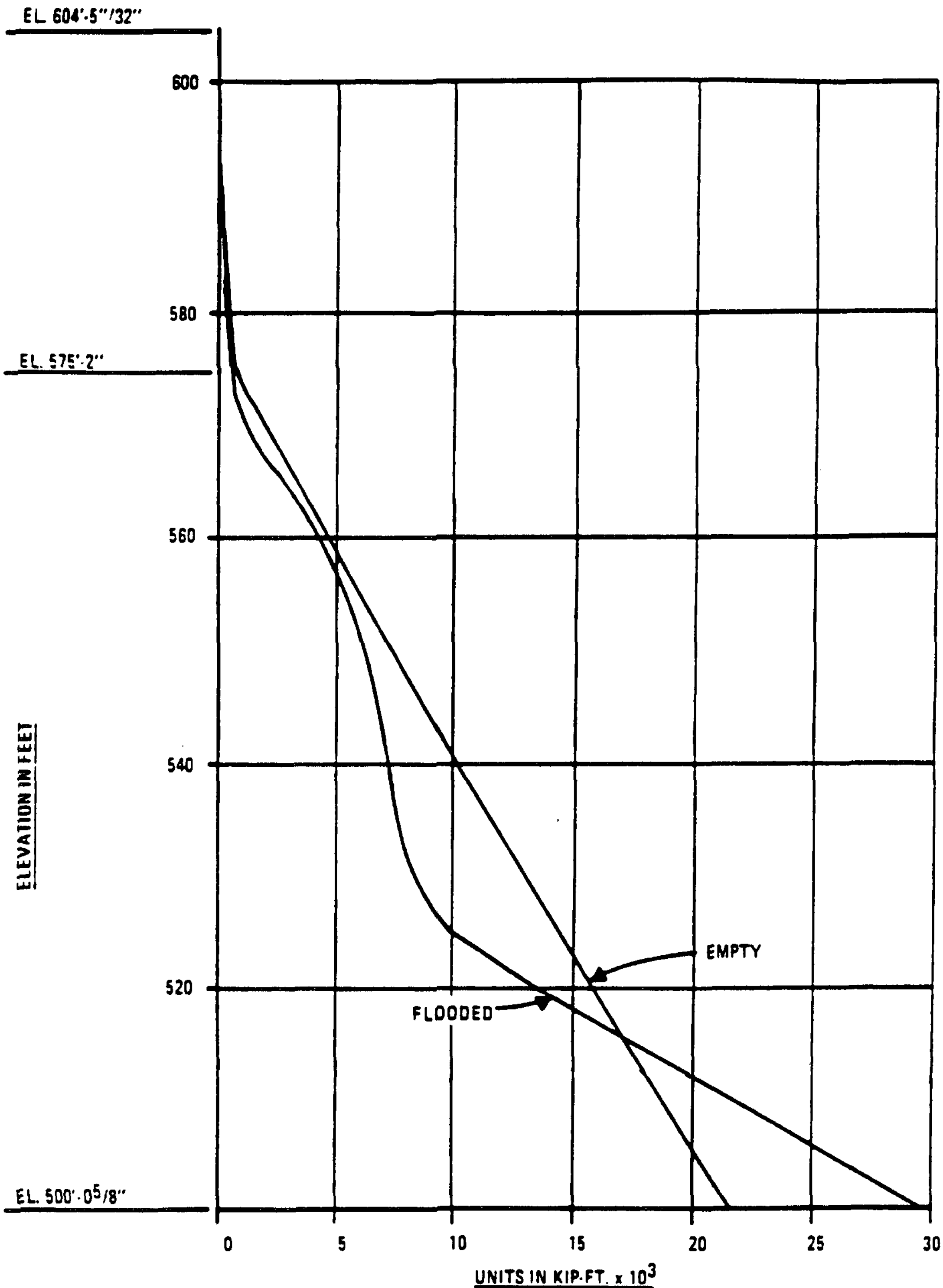
FIGURE 3.8-4



DRESDEN STATION UNITS 2 & 3
DRYWELL OBE DISPACEMENT DIAGRAM - E-W DIRECTION
FIGURE 3.8-5



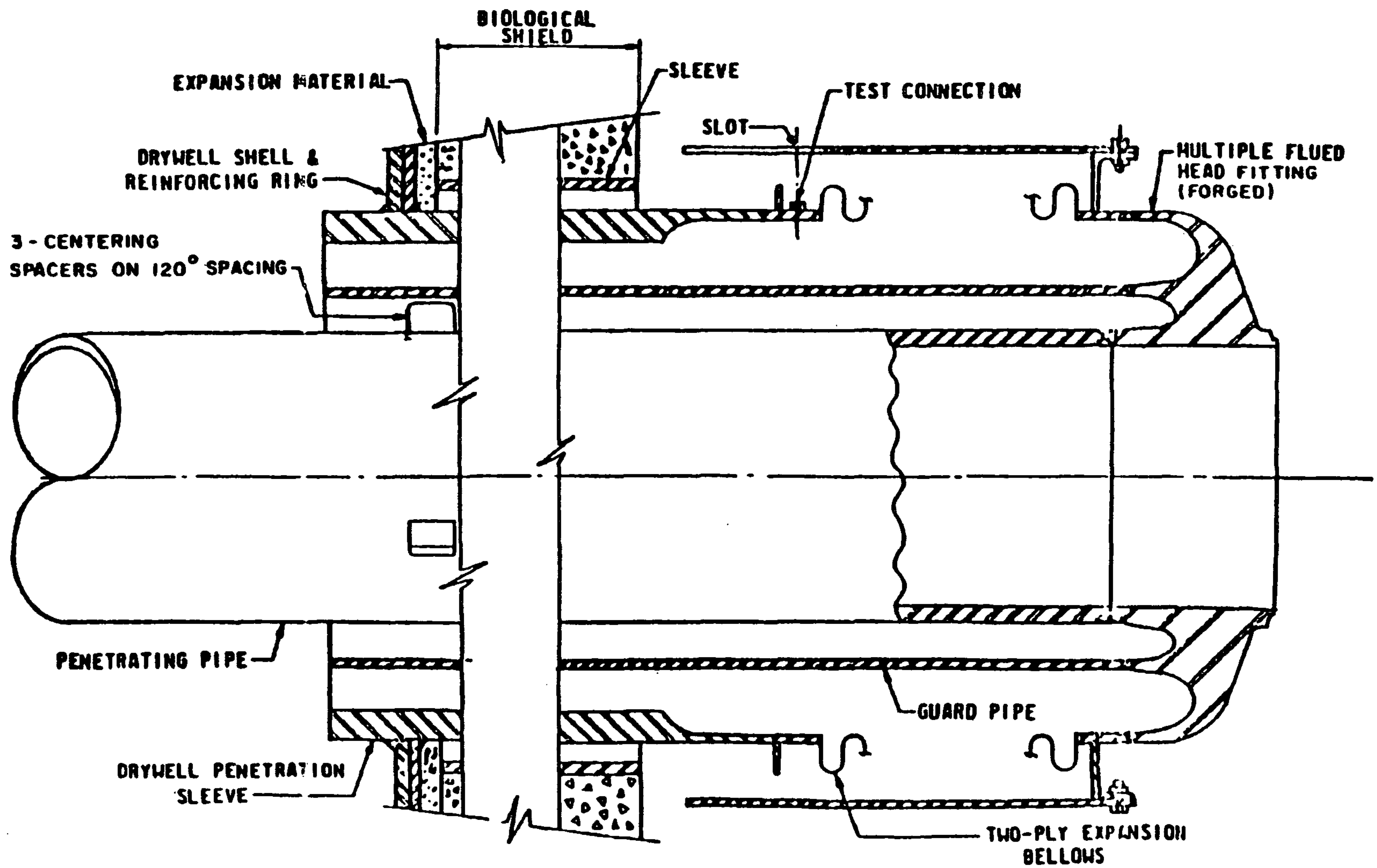
DRESDEN STATION UNITS 2 & 3
DRYWELL OBE SHEAR DIAGRAM - E-W DIRECTION
FIGURE 3.8-6



DRESDEN STATION
UNITS 2 & 3

DRYWELL OBE MOMENT DIAGRAM -
E-W DIRECTION

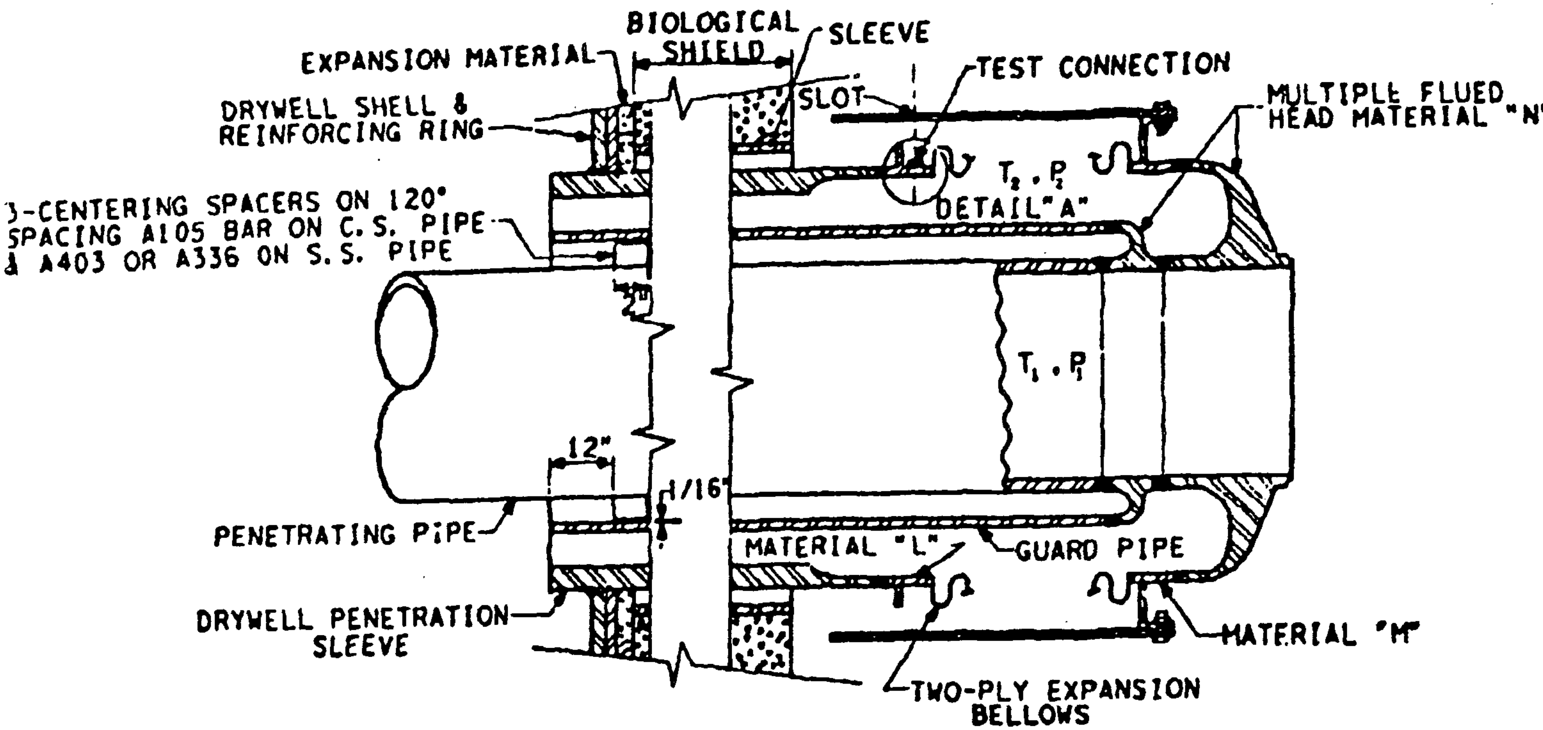
FIGURE 3.8-7



DRESDEN STATION
UNITS 2 & 3

TYPE "1" PIPING PENETRATION ASSEMBLY

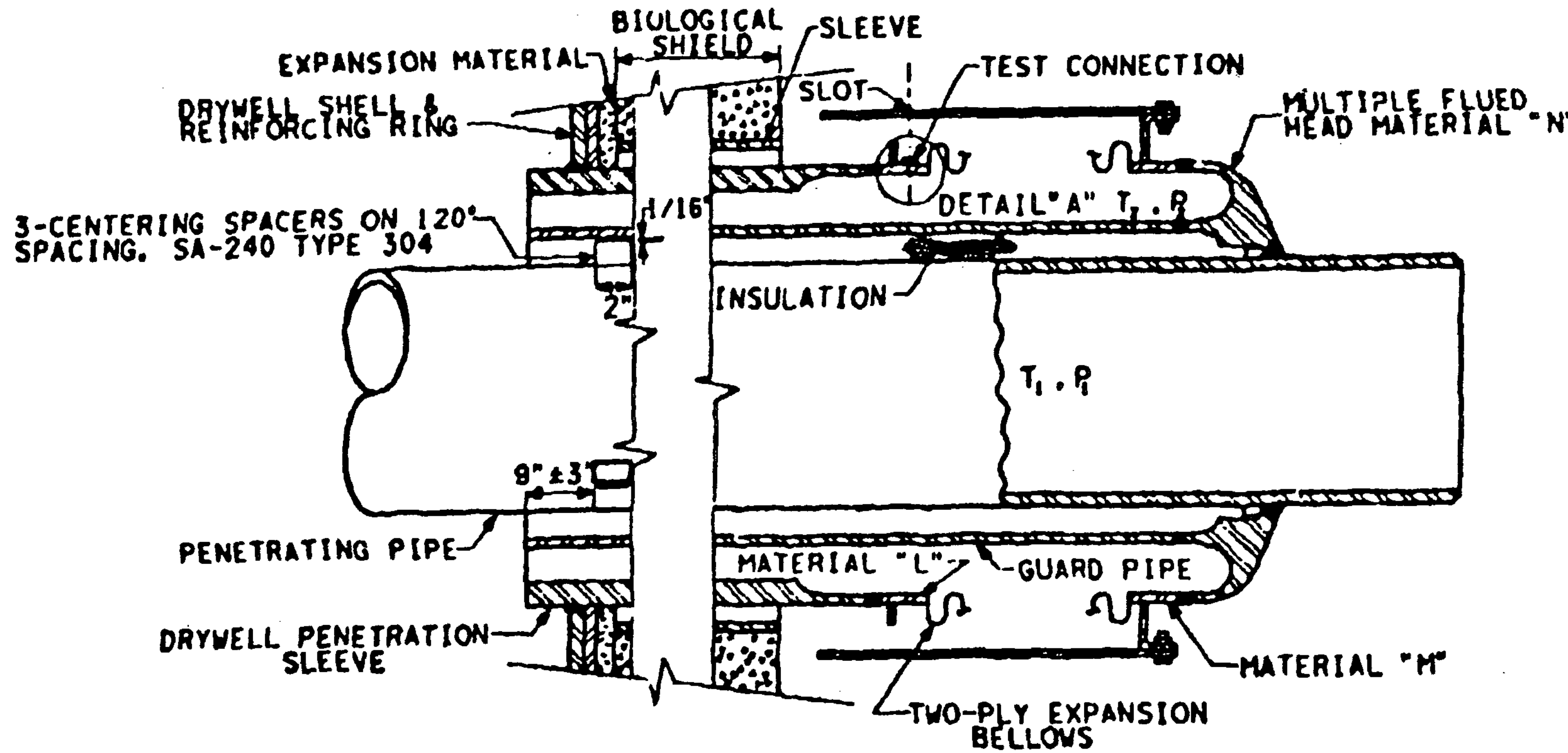
FIGURE 3.8-8



Note: See drawing M-97 for detail information.

FOR INFORMATION ONLY

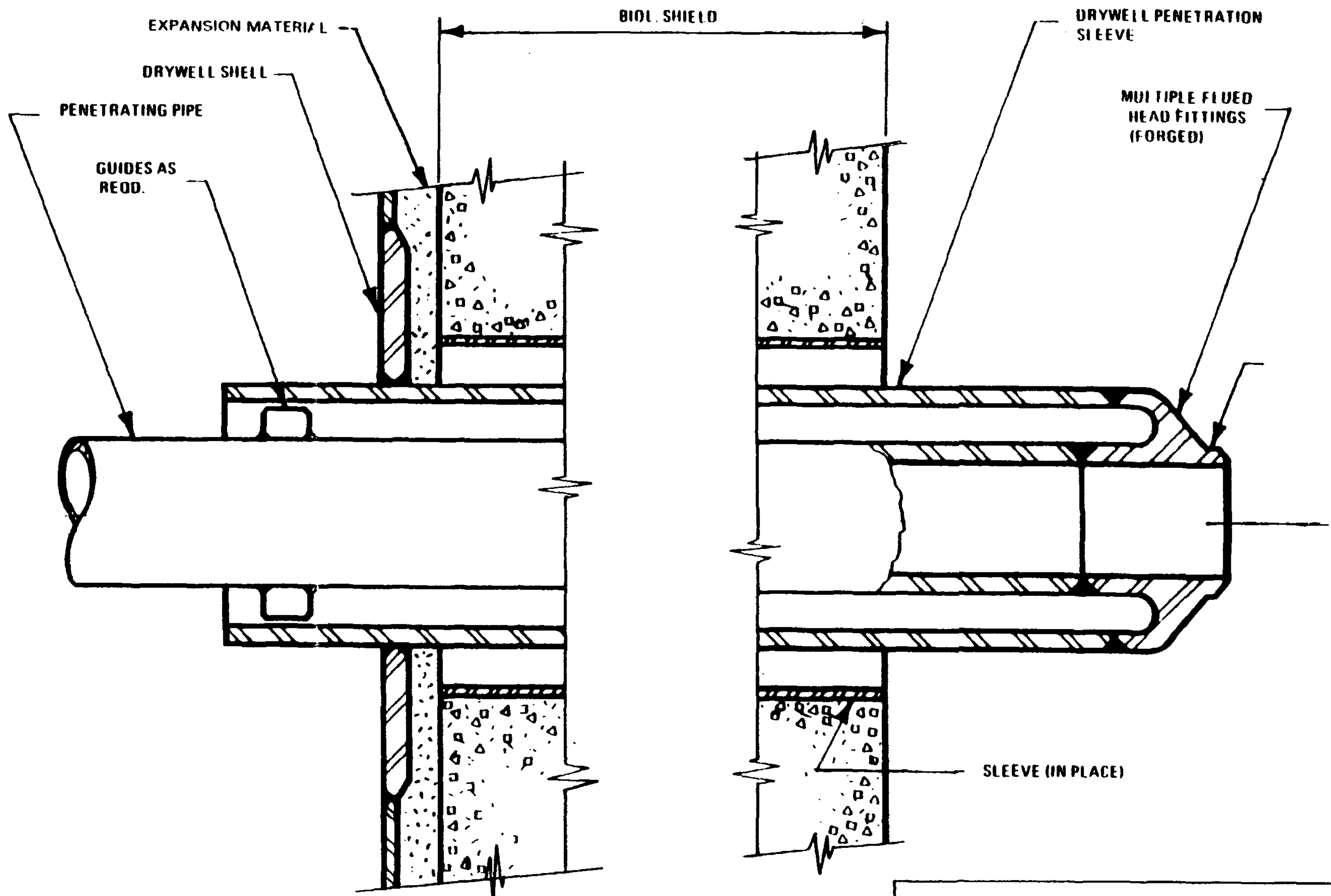
UFSAR REVISION 4, JUNE 2001
DRESDEN STATION UNITS 2 & 3
TYPE "1A" PIPING PENETRATION ASSEMBLY
FIGURE 3.8-9A



Note: See drawing M-97 for detail information.

FOR INFORMATION ONLY

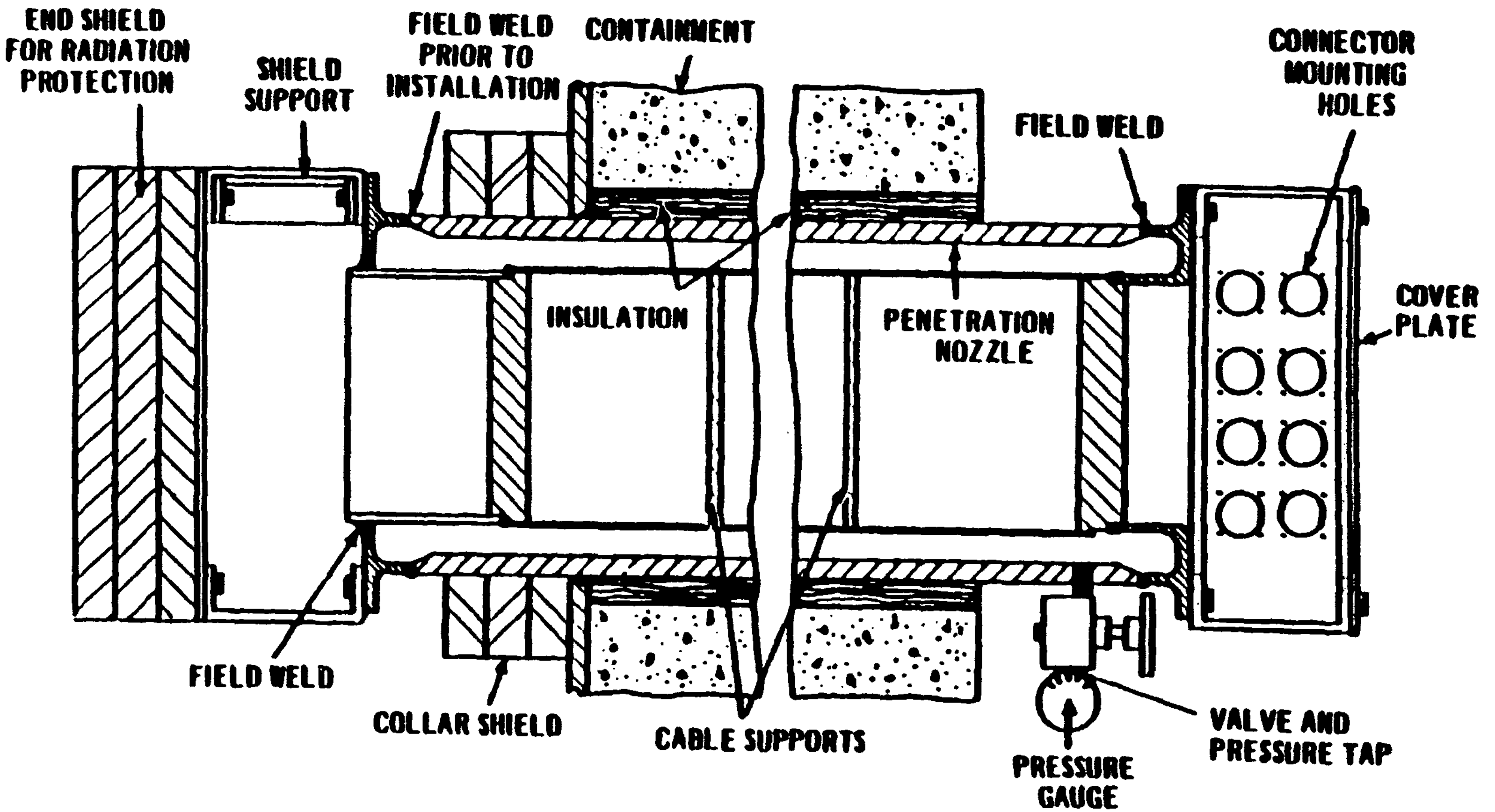
UFSAR REVISION 4, JUNE 2001
DRESDEN STATION UNITS 2 & 3
TYPE "1B" PIPING PENETRATION ASSEMBLY
FIGURE 3.8-9B



DRESDEN STATION
UNITS 2 & 3

TYPE "2" PIPING PENETRATION ASSEMBLY

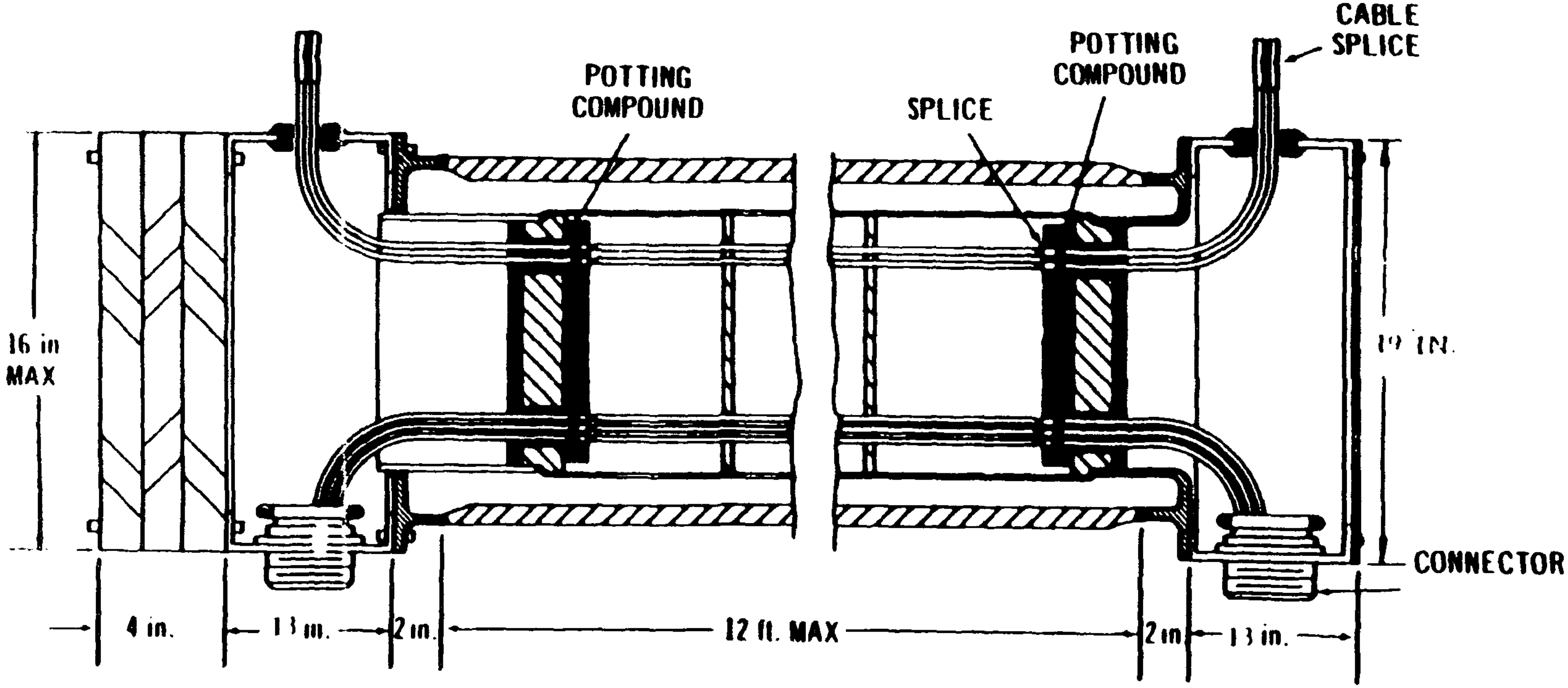
FIGURE 3.8-10



DRESDEN STATION
UNITS 2 & 3

TYPICAL ELECTRICAL PENETRATION ASSEMBLY CANISTER

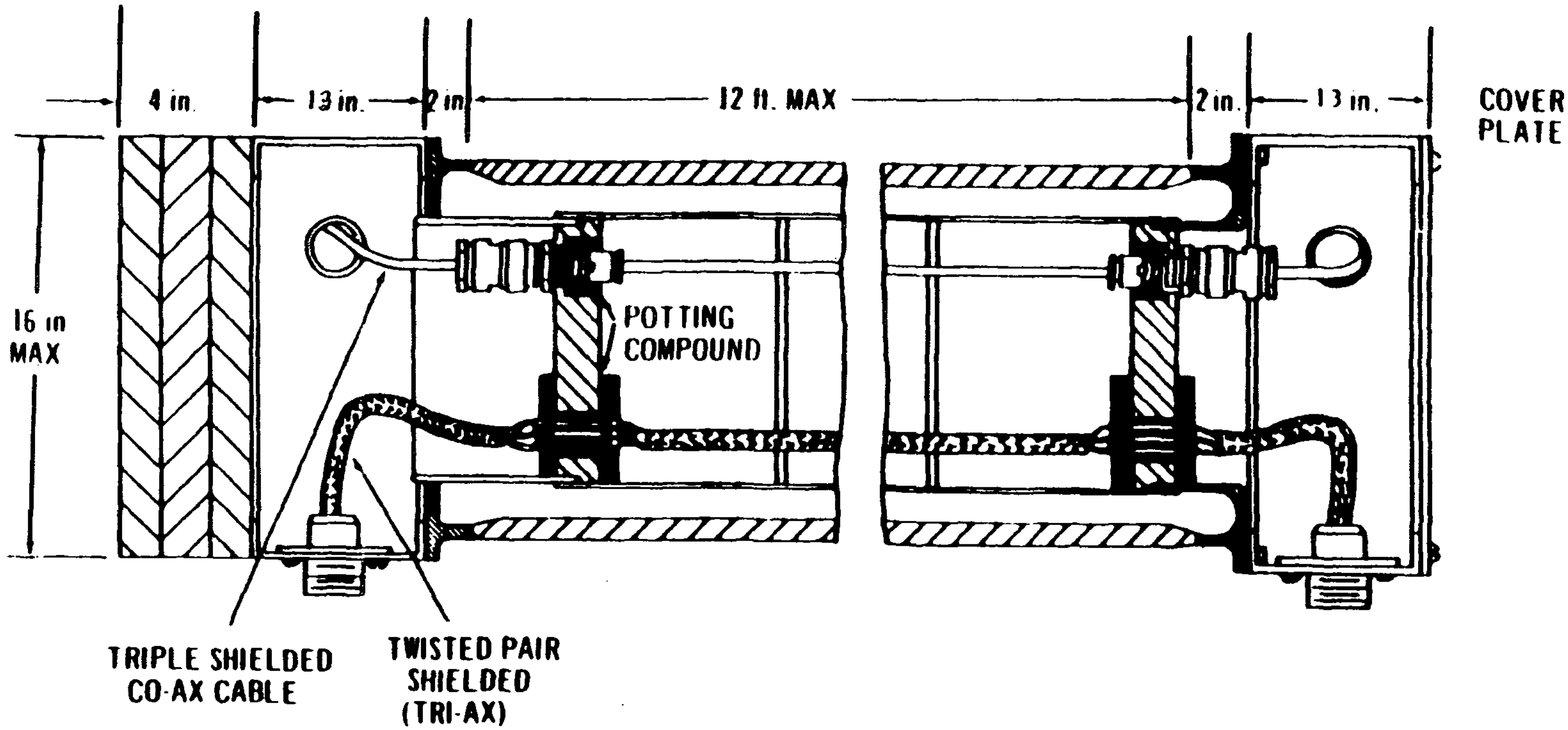
FIGURE 3.8-11



DRESDEN STATION
UNITS 2 & 3

LOW VOLTAGE POWER AND CONTROL
ELECTRICAL PENETRATION ASSEMBLY

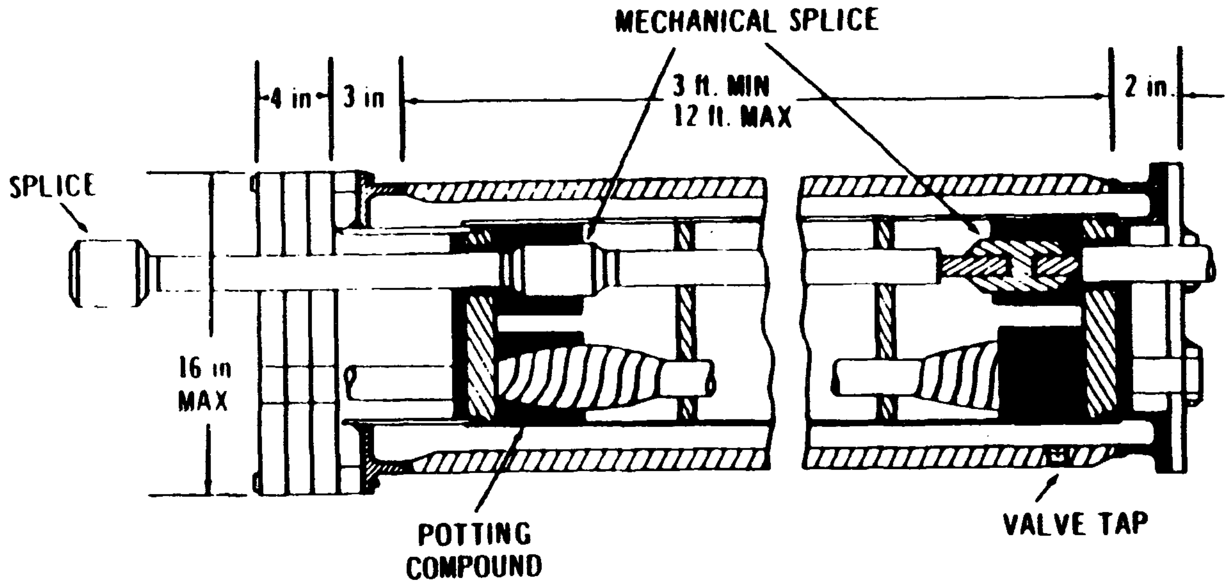
FIGURE 3.8-12



DRESDEN STATION
UNITS 2 & 3

SHILDED SIGNAL CABLE
ELECTRICAL PENETRATION ASSEMBLY

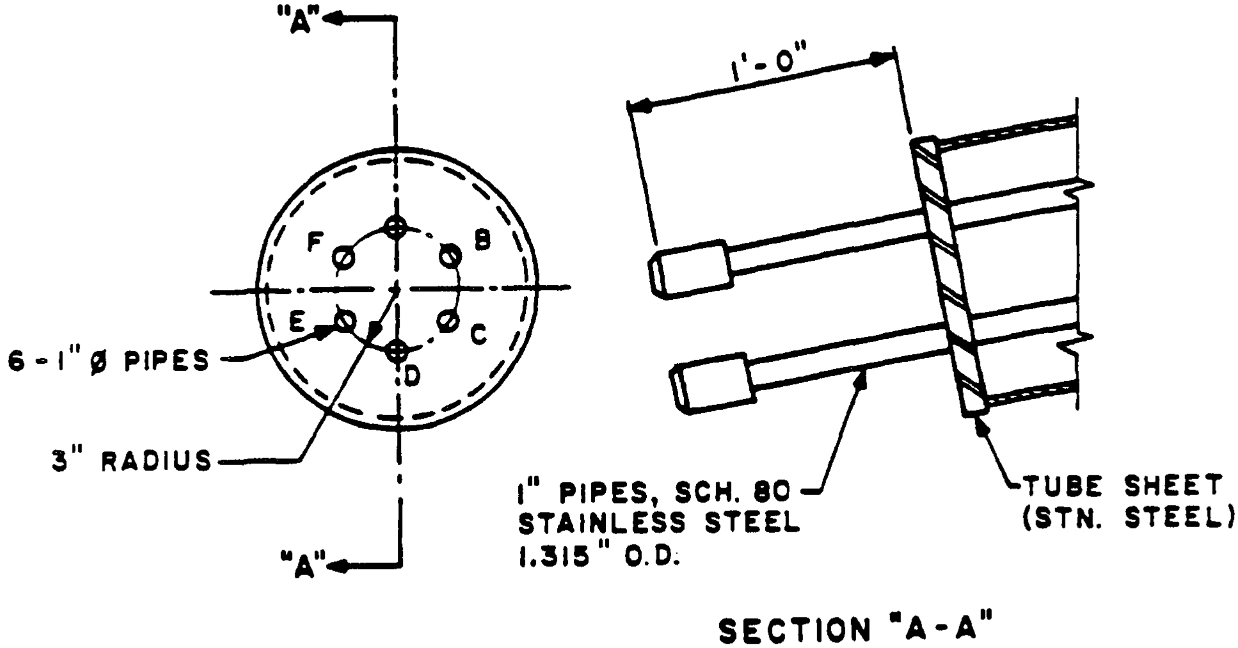
FIGURE 3.8-13



DRESDEN STATION
UNITS 2 & 3

HIGH VOLTAGE POWER
ELECTRICAL PENETRATION ASSEMBLY

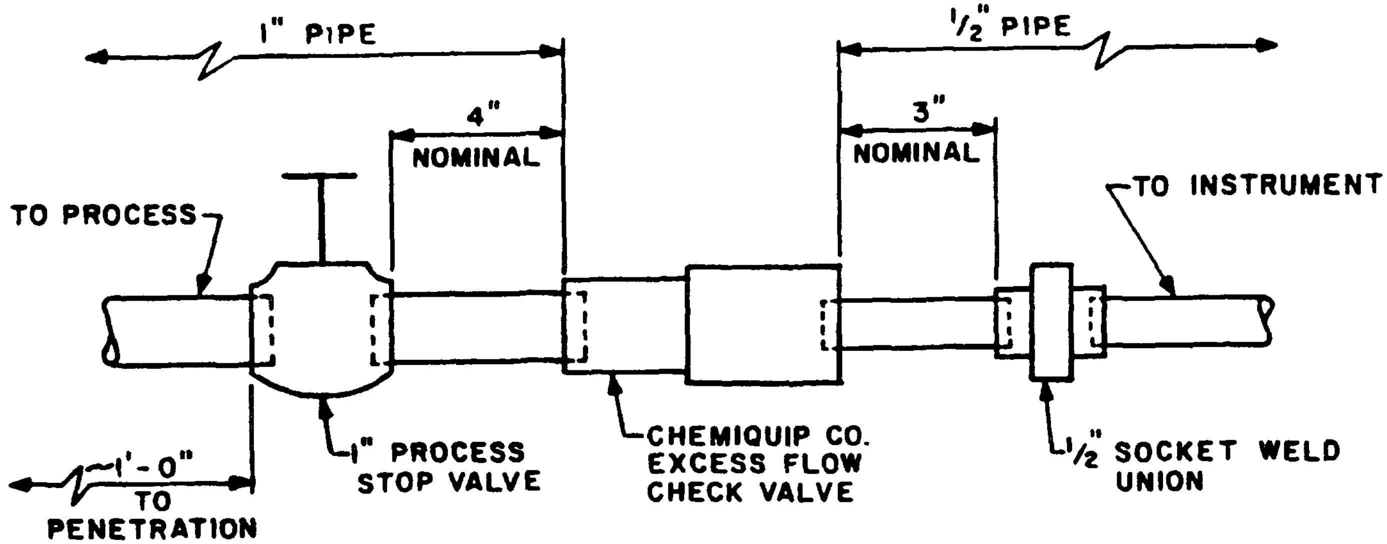
FIGURE 3.8-14



DRESDEN STATION
UNITS 2 & 3

CONTAINMENT VESSEL INSTRUMENT LINE PENETRATION

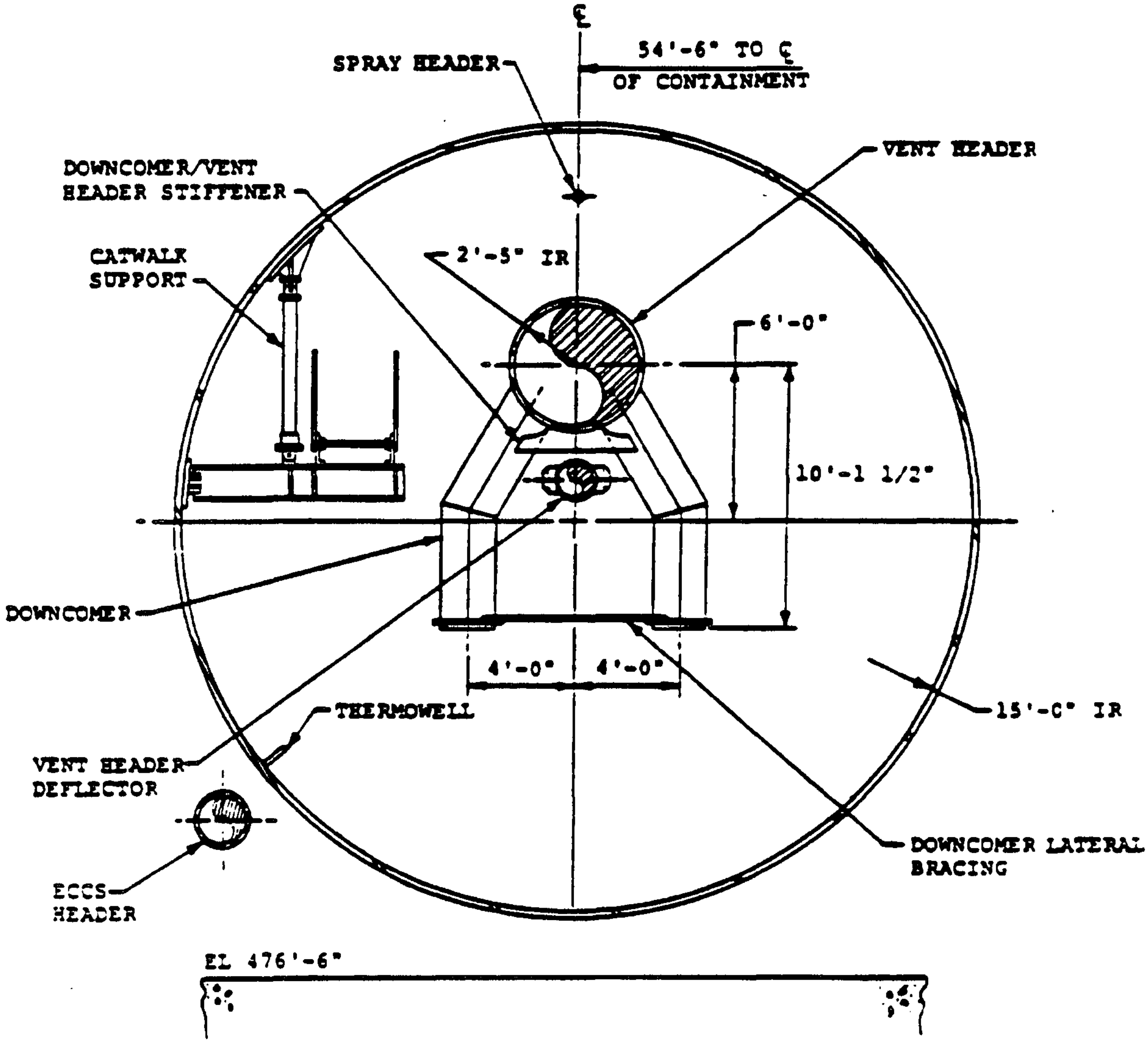
FIGURE 3.8-15



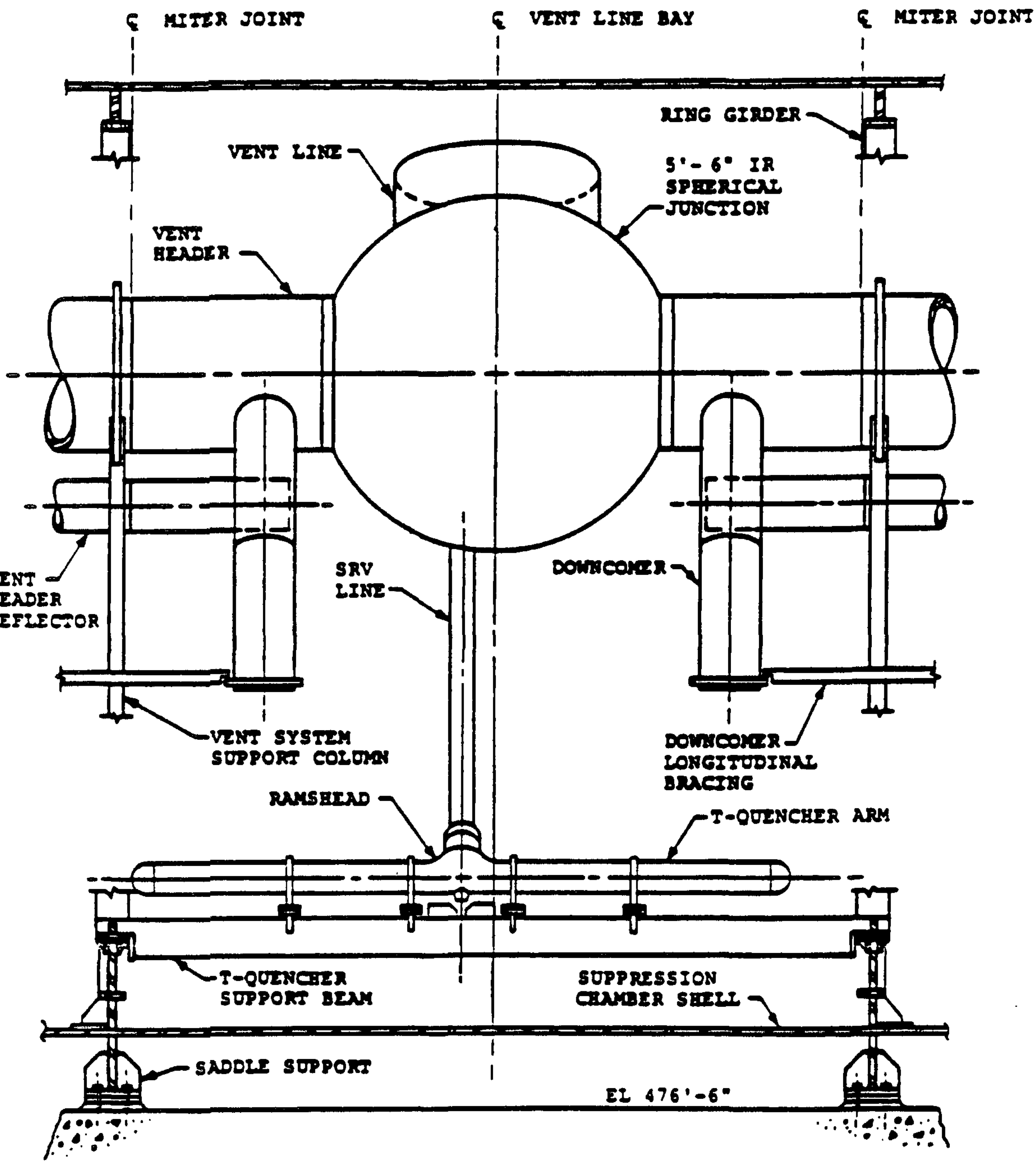
NOTES:

1. VALVES TO BE LOCATED IN HORIZONTAL PIPE RUN AND SLOPED AT 1/4" PER FT. MINIMUM.
2. VALVES TO BE INSTALLED AT ACCESSIBLE LOCATION.

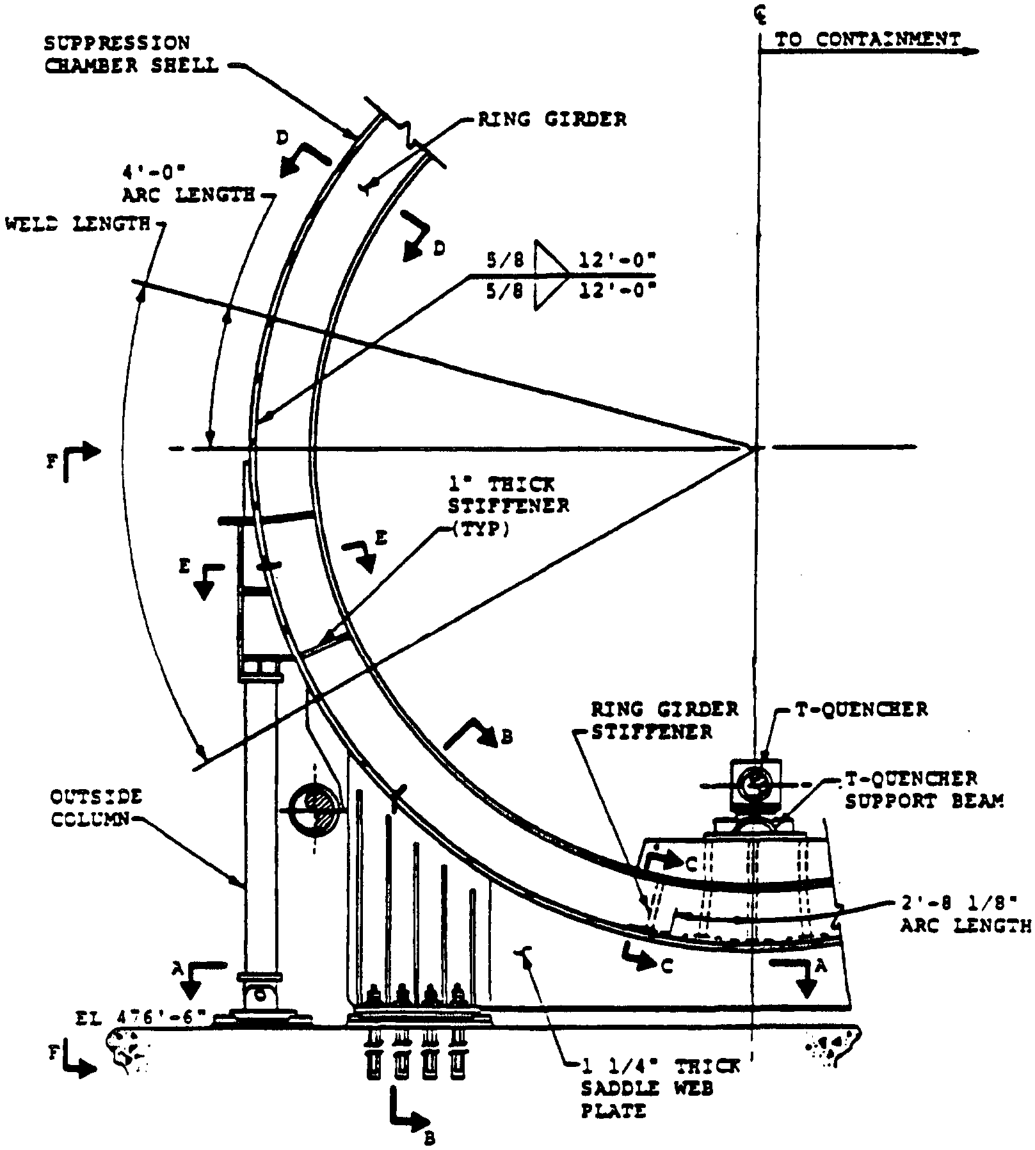
DRESDEN STATION UNITS 2 & 3
PROCESS STOP VALVE AND EXCESS FLOW CHECK VALVE PIPING
FIGURE 3.8-16



DRESDEN STATION UNITS 2 & 3
SUPPRESSION CHAMBER SECTION - MIDBAY NON-VENT LINE BAY
FIGURE 3.8-17

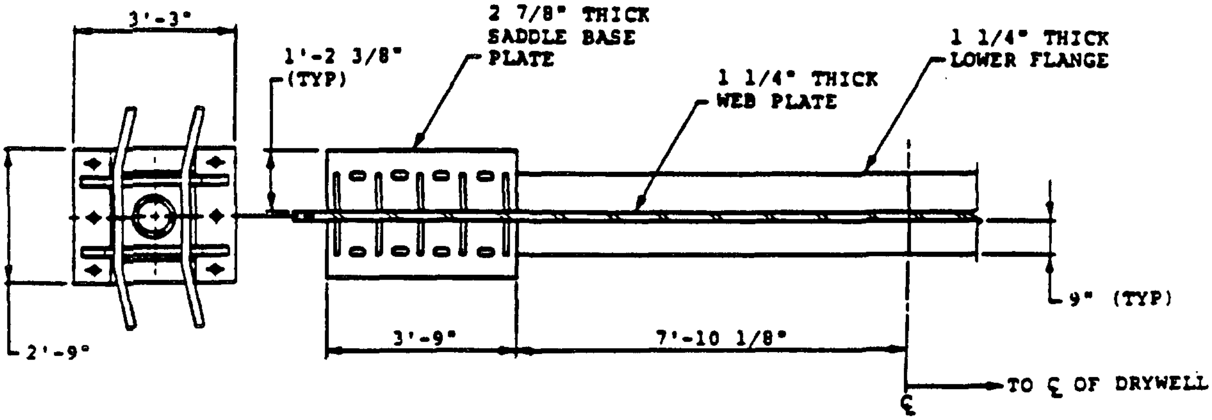


DRESDEN STATION UNITS 2 & 3
DEVELOPED VIEW OF SUPPRESSION CHAMBER SEGMENT
FIGURE 3.8-18



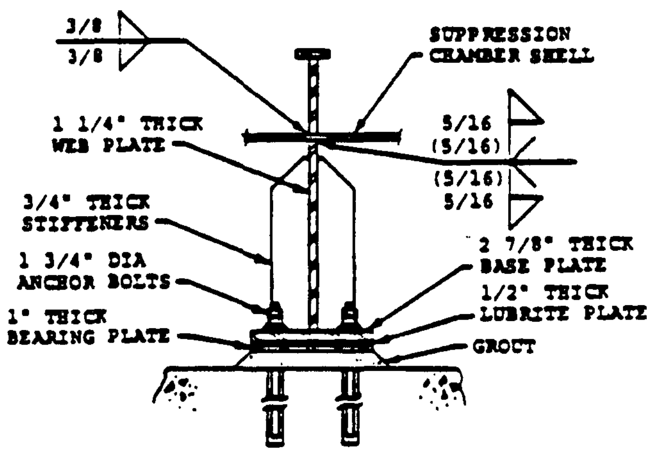
1. SEE FIGURES 3.8-20 THROUGH 3.8-23 FOR SECTIONS AND VIEWS, RESPECTIVELY.

DRESDEN STATION UNITS 2 & 3
SUPPRESSION CHAMBER RING GIRDER AND VERTICAL SUPPORTS - PARTIAL ELEVATION VIEW
FIGURE 3.8-19



SECTION A-A

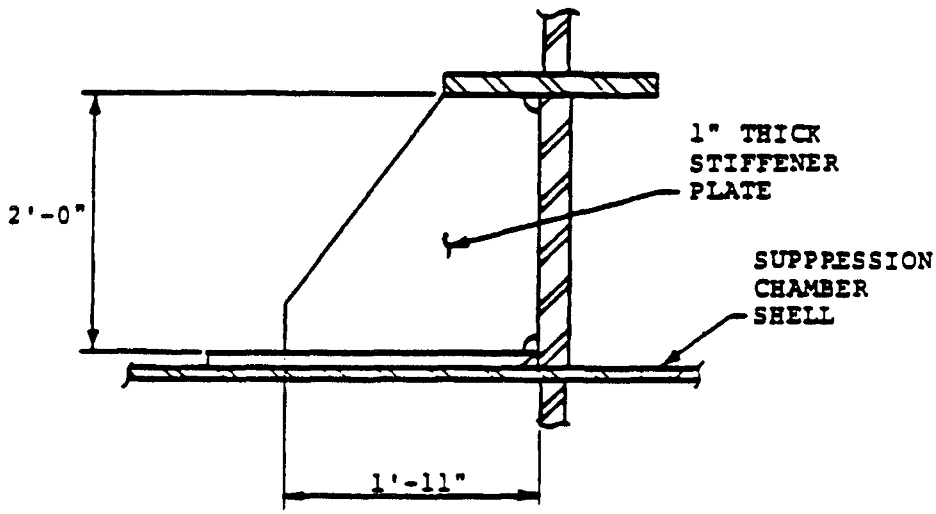
(FROM FIGURE 3.8-19)



SECTION B-B

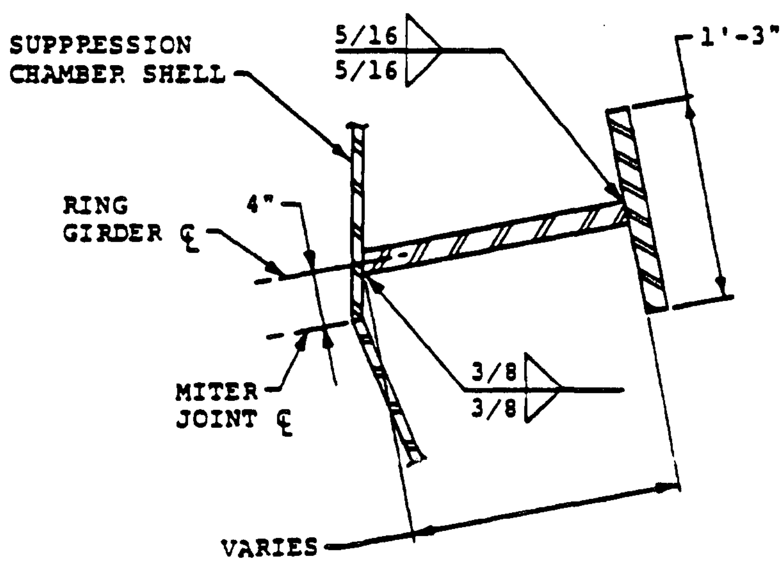
(FROM FIGURE 3.8-19)

DRESDEN STATION UNITS 2 & 3
SUPPRESSION CHAMBER VERTICAL SUPPORT BASE PLATES - PARTIAL PLAN VIEW AND DETAILS
FIGURE 3.8-20



SECTION C-C

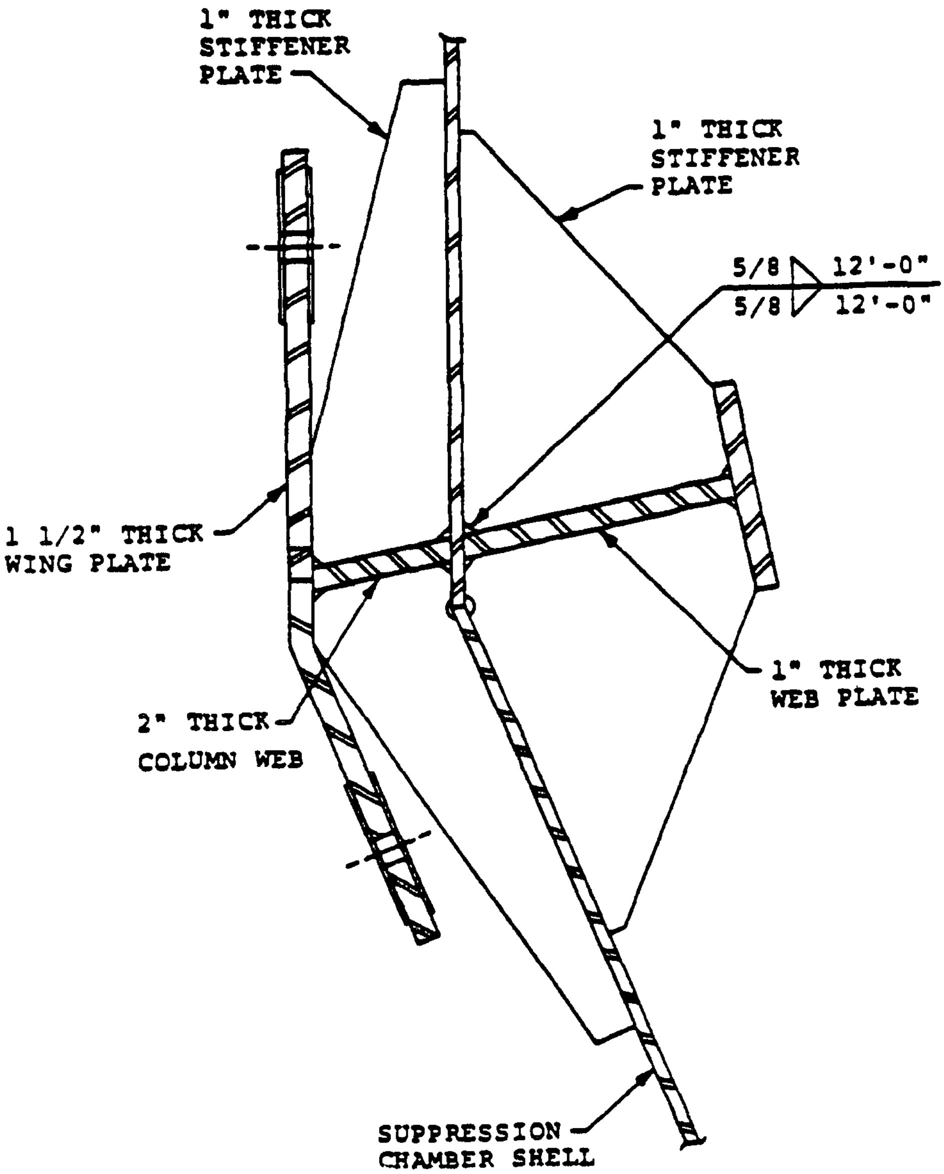
(FROM FIGURE 3.8-19)



SECTION D-D

(FROM FIGURE 3.8-19)

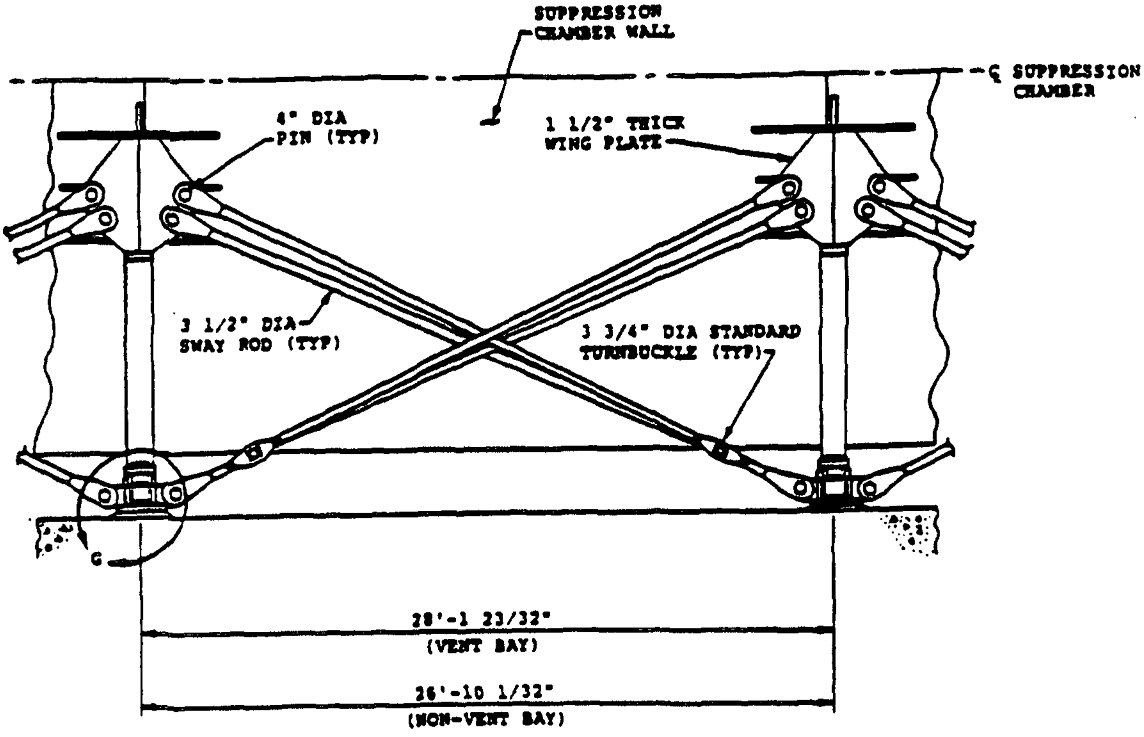
DRESDEN STATION UNITS 2 & 3
SUPPRESSION CHAMBER RING GIRDER DETAILS
FIGURE 3.8-21



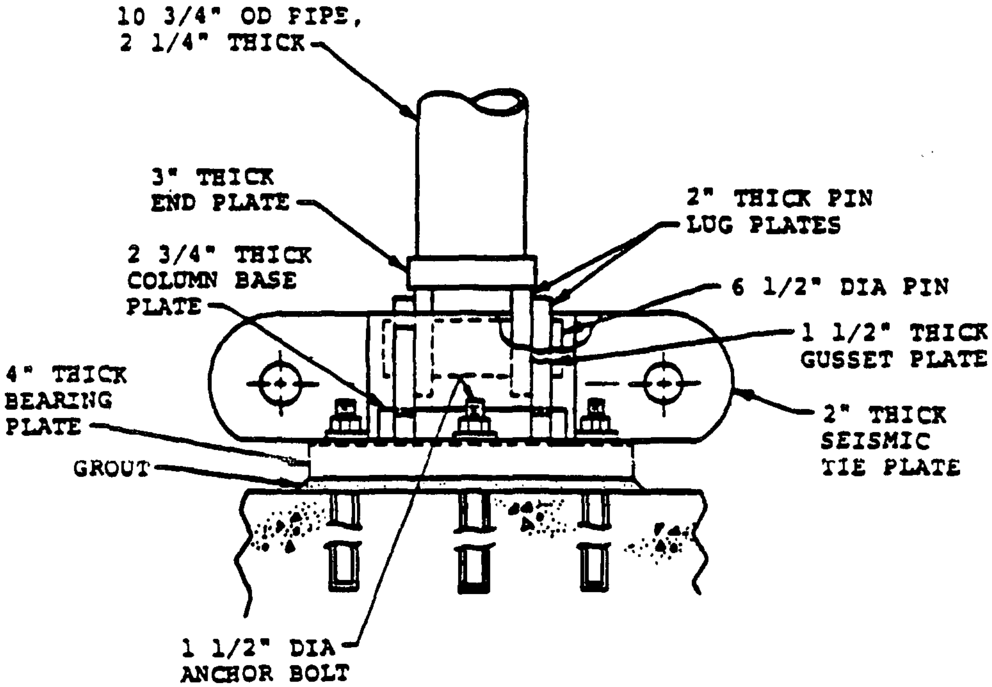
SECTION E-E

(FROM FIGURE 3.8-19)

DRESDEN STATION UNITS 2 & 3
SUPPRESSION CHAMBER RING GIRDER AND COLUMN CONNECTION DETAILS
FIGURE 3.8-22



VIEW F-F
(FROM FIGURE 3.8-19)

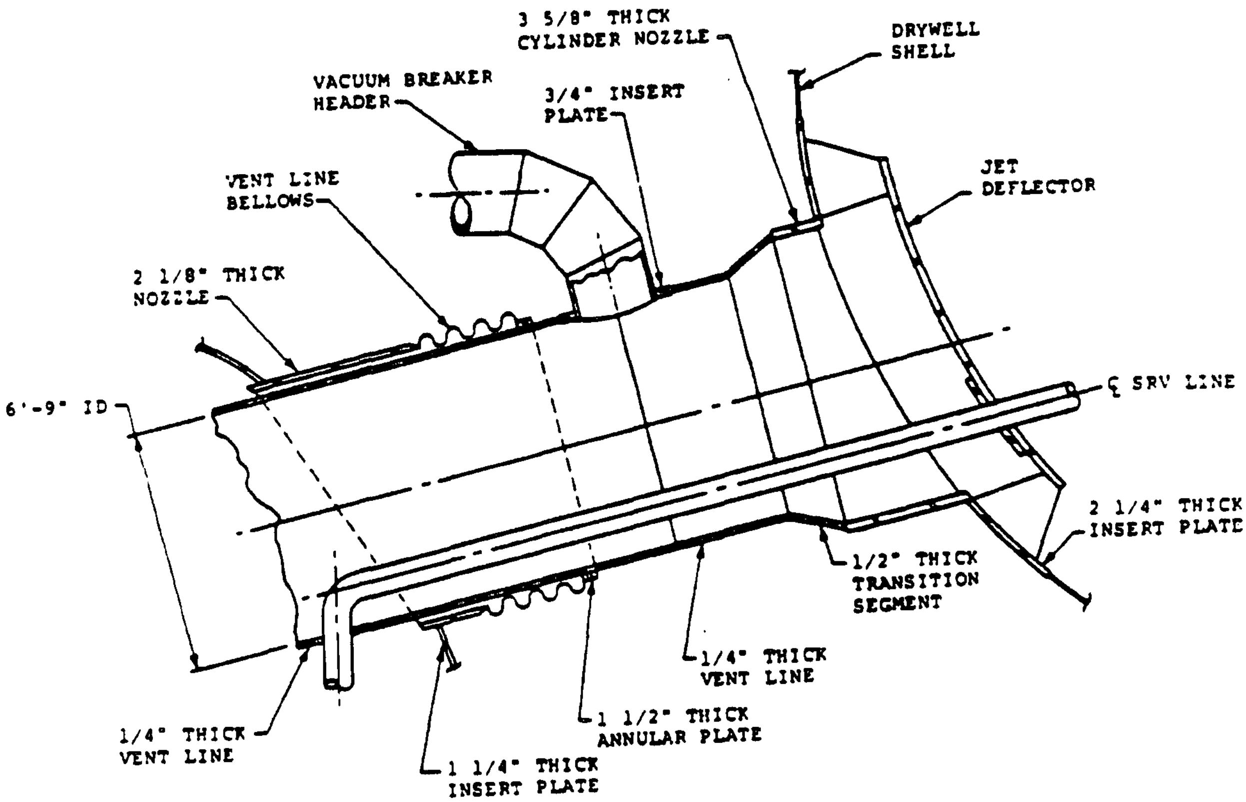


DETAIL G

DRESDEN STATION
UNITS 2 & 3

SUPPRESSION CHAMBER SEISMIC RESTRAINT

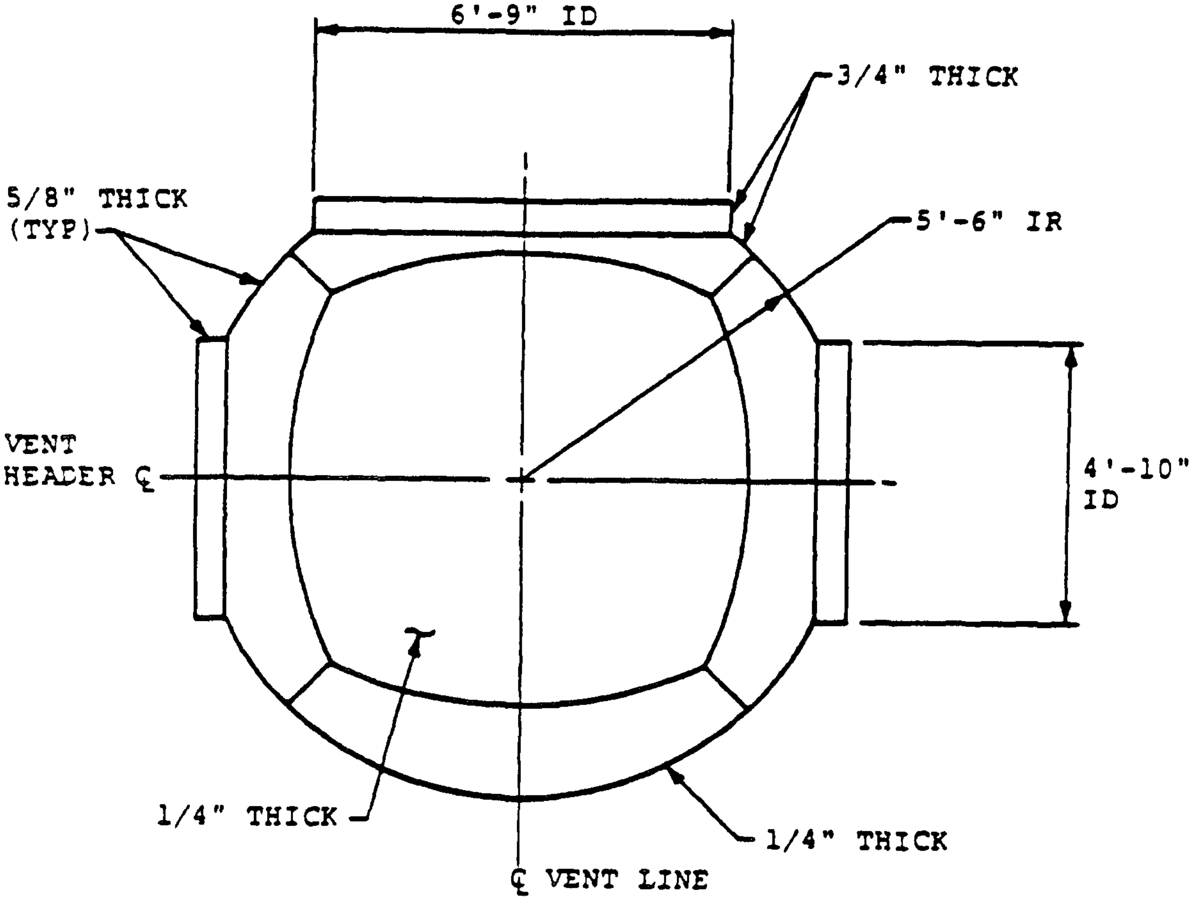
FIGURE 3.8-23



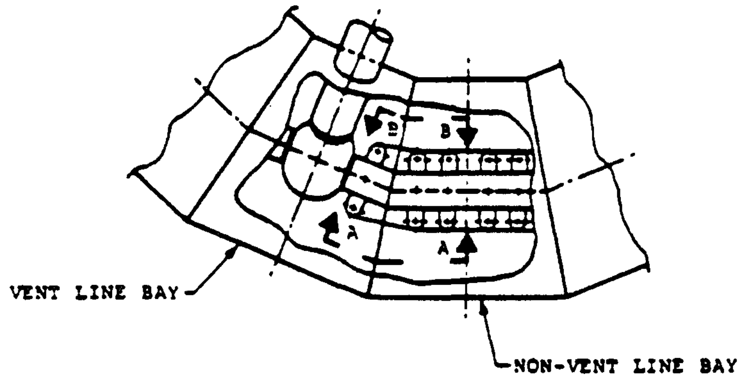
DRESDEN STATION
UNITS 2 & 3

VENT LINE DETAILS - UPPER END

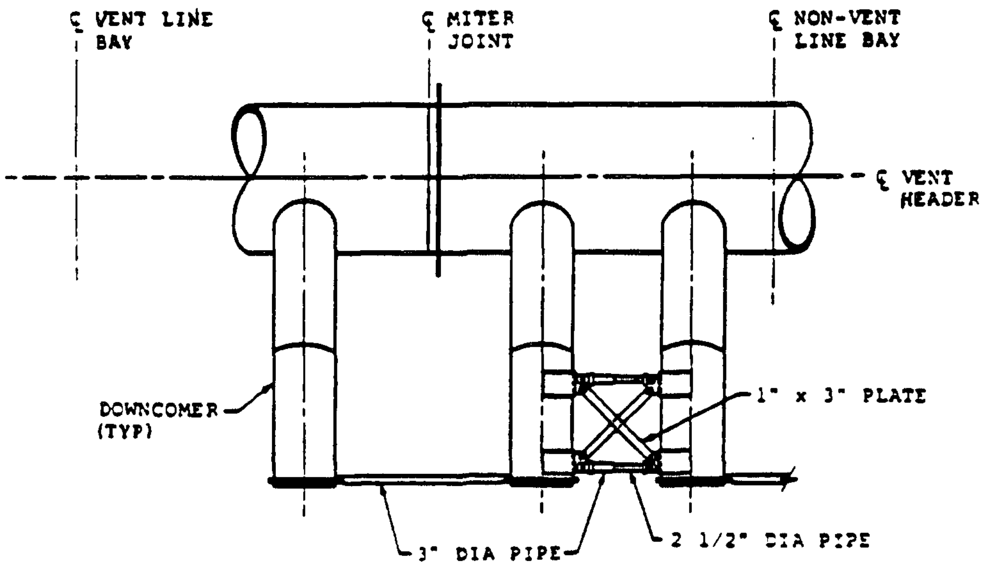
FIGURE 3.8-24



DRESDEN STATION UNITS 2 & 3
VENT LINE - VENT HEADER SPHERICAL JUNCTION
FIGURE 3.8-25



PARTIAL PLAN VIEW OF SUPPRESSION CHAMBER

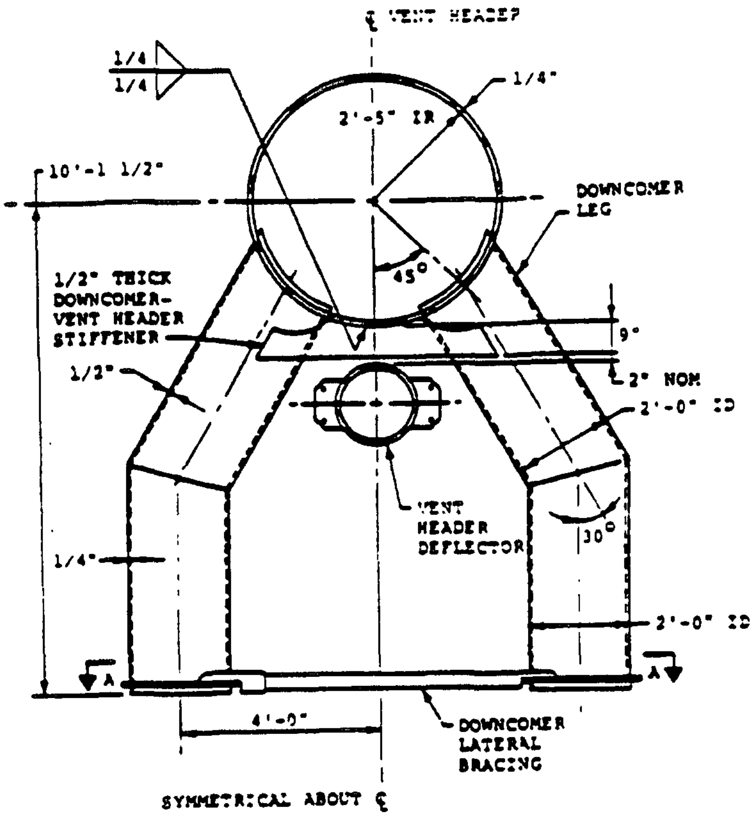


VIEW A-A

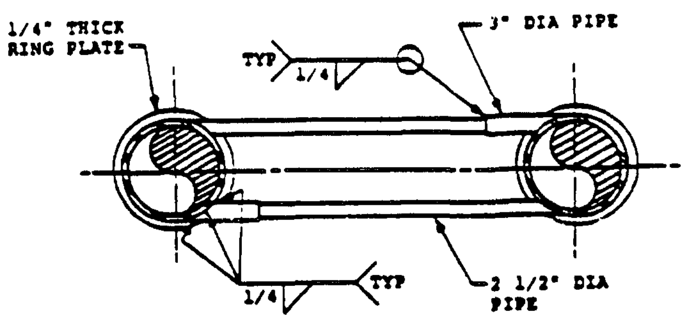
VIEW B-B (OPPOSITE HAND)

1. VENT HEADER DEFLECTOR AND VENT HEADER COLUMNS NOT SHOWN FOR CLARITY.

DRESDEN STATION UNITS 2 & 3
DEVELOPED VIEW OF DOWNCOMER LONGITUDINAL BRACING SYSTEM
FIGURE 3.8-26



ELEVATION VIEW

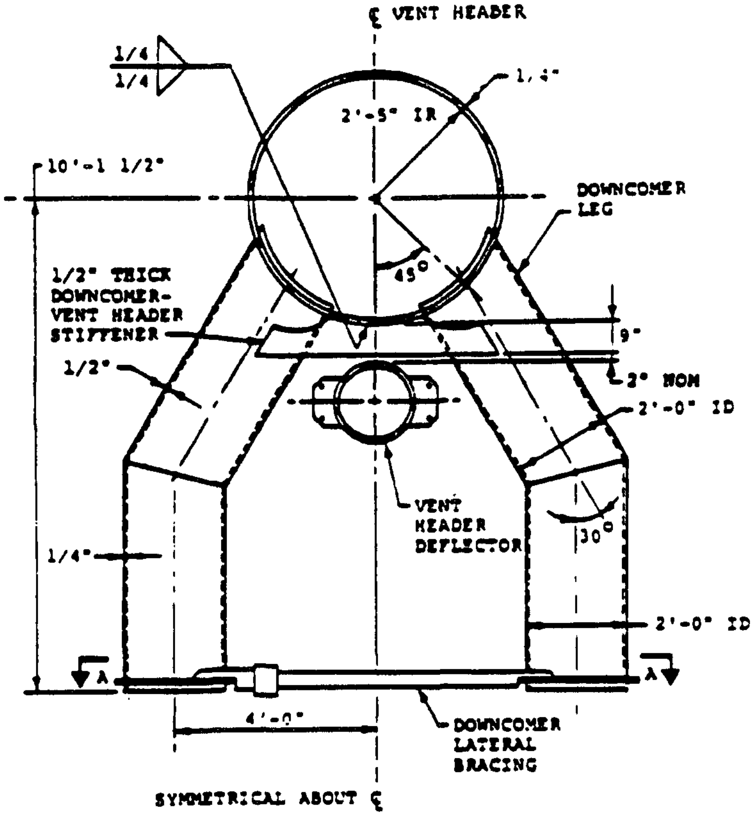


SECTION A-A

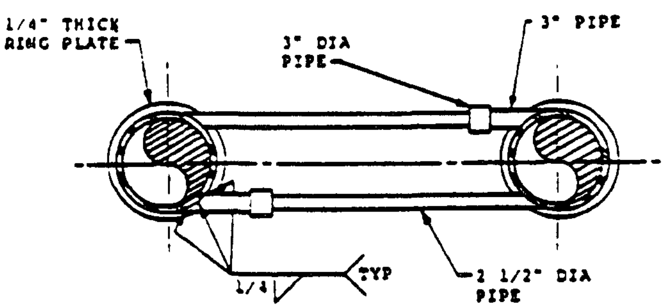
DRESDEN STATION
UNITS 2 & 3

DOWNCOMER-TO-VENT HEADER
INTERSECTION DETAILS - DRESDEN UNIT 2

FIGURE 3.8-27



ELEVATION VIEW

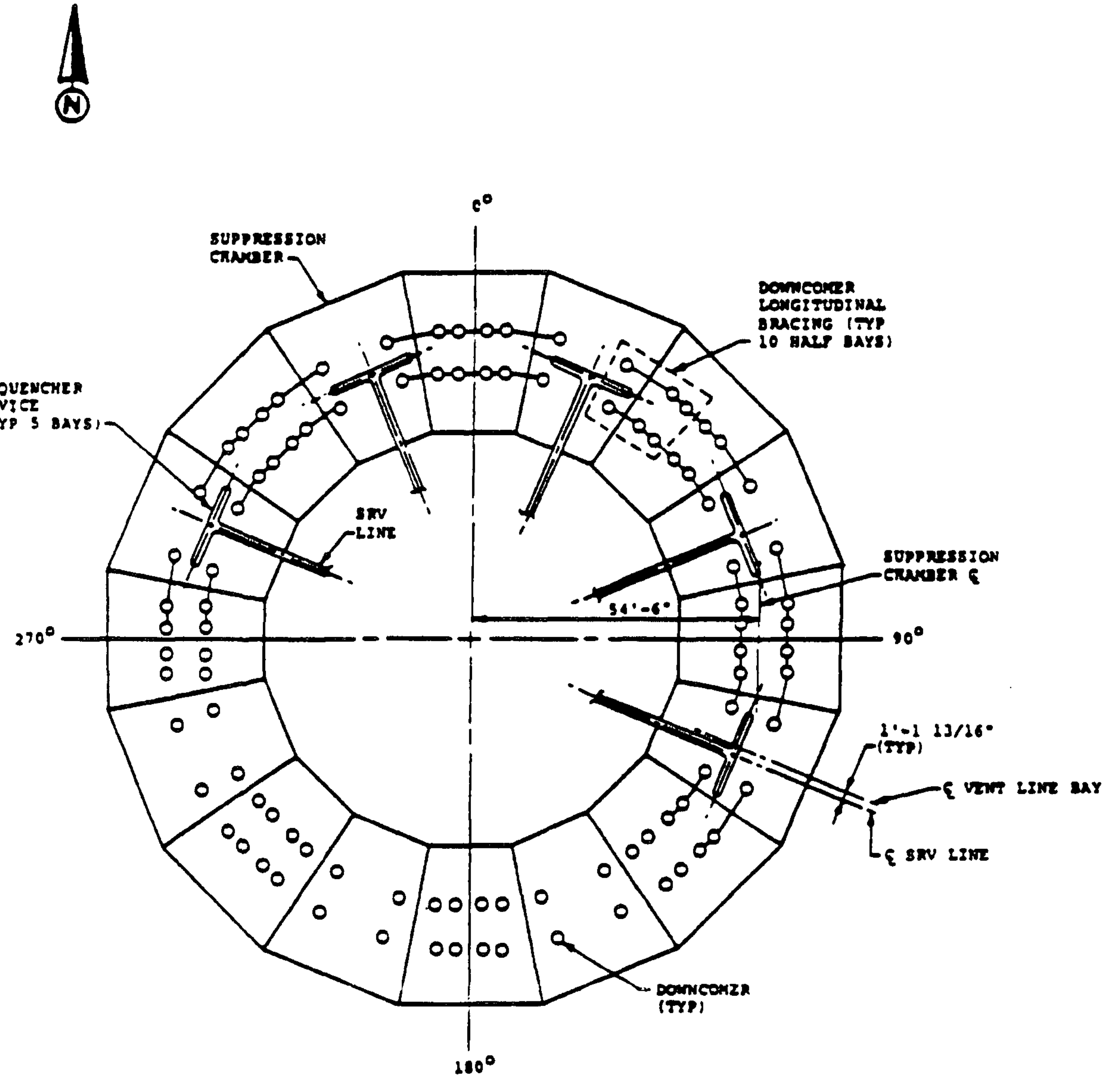


SECTION A-A

DRESDEN STATION
UNITS 2 & 3

DOWNCOMER-TO-VENT HEADER
INTERSECTION DETAILS - DRESDEN UNIT 3

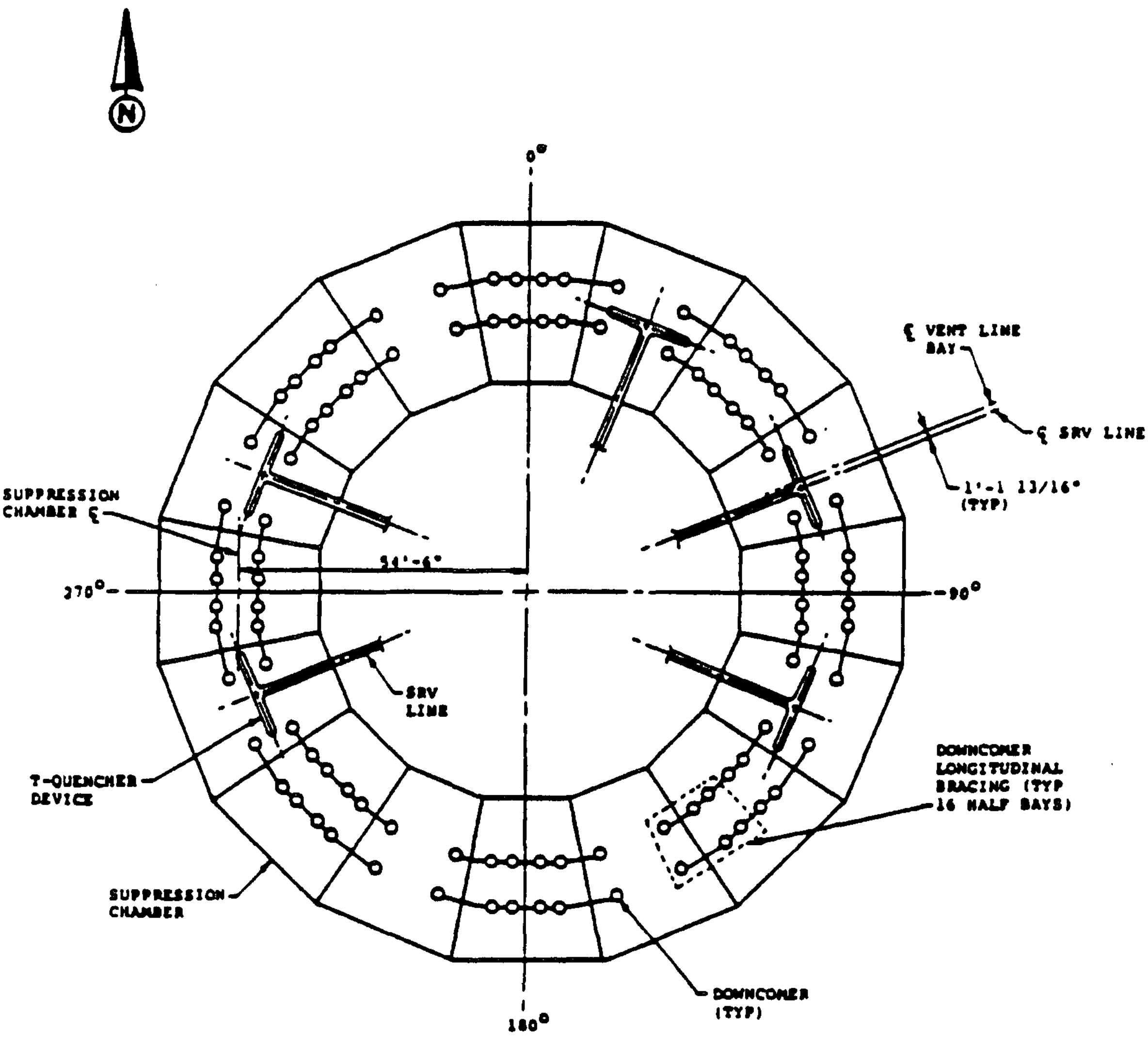
FIGURE 3.8-28



DRESDEN STATION
UNITS 2 & 3

DOWNCOMER LONGITUDINAL BRACING SYSTEM CONFIGURATION - DRESDEN UNIT 2

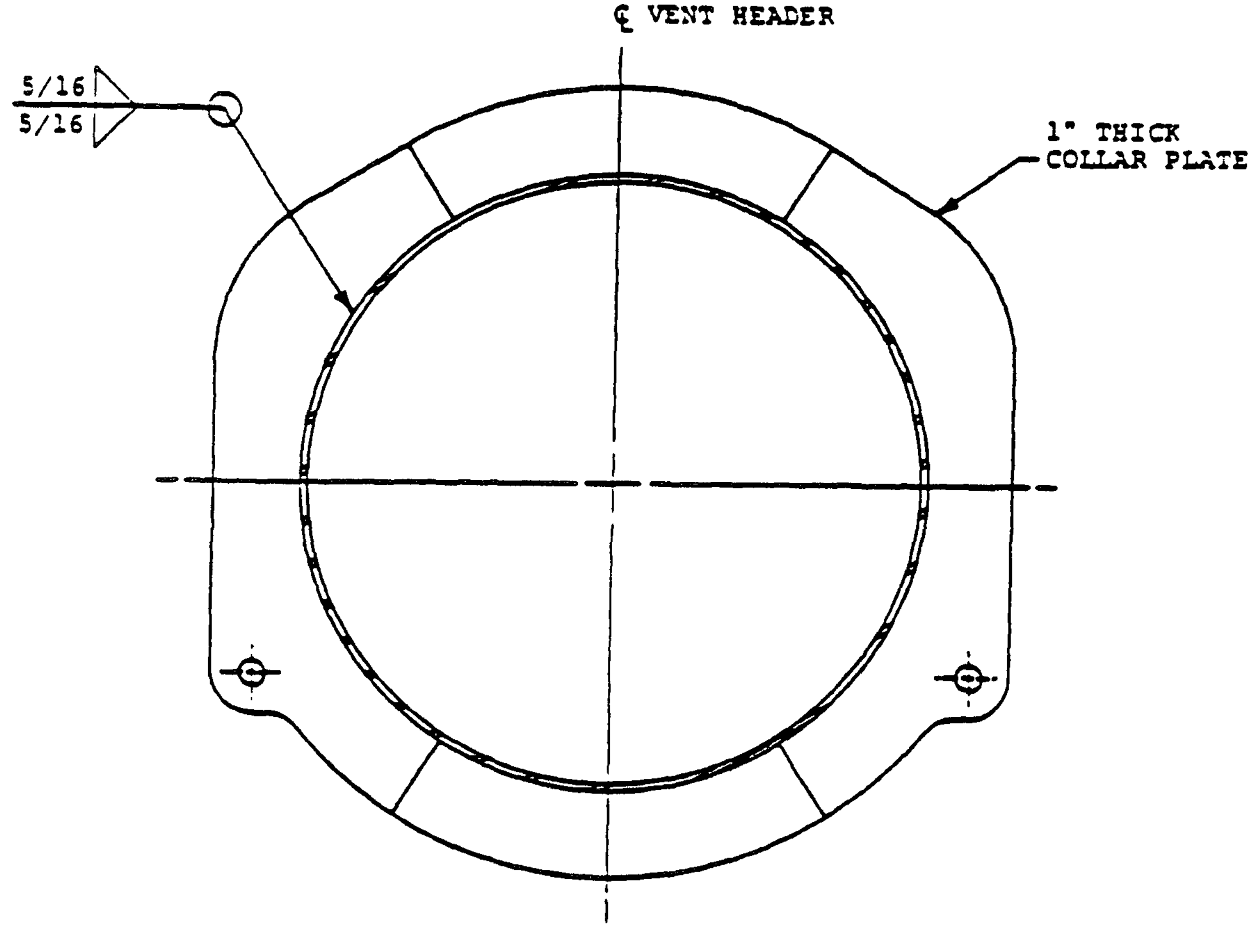
FIGURE 3.8-29



DRESDEN STATION
UNITS 2 & 3

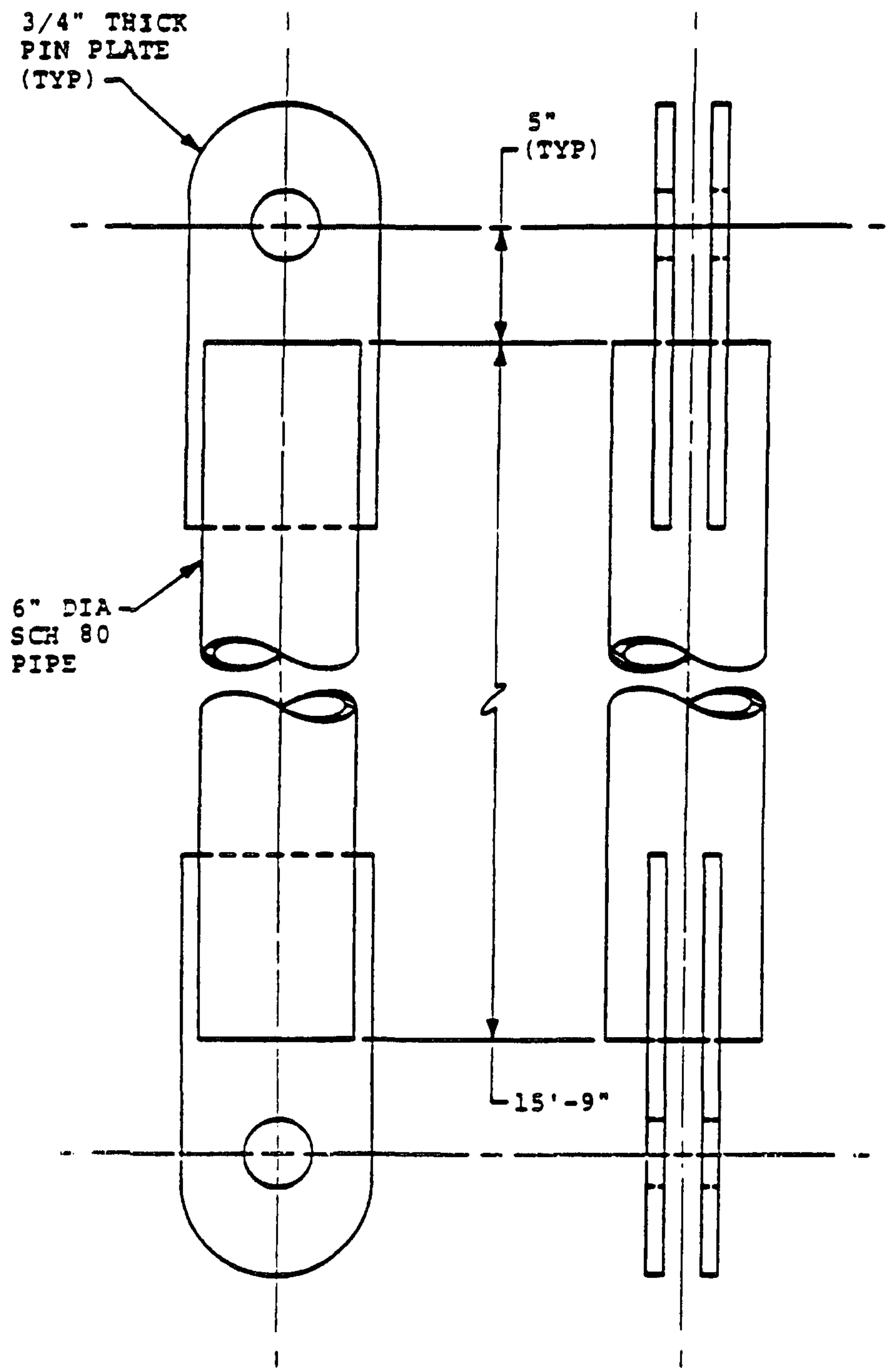
DOWNCOMER LONGITUDINAL BRACING SYSTEM CONFIGURATION - DRESDEN UNIT 3

FIGURE 3.8-30



SECTION THROUGH VENT HEADER
SUPPORT COLLAR

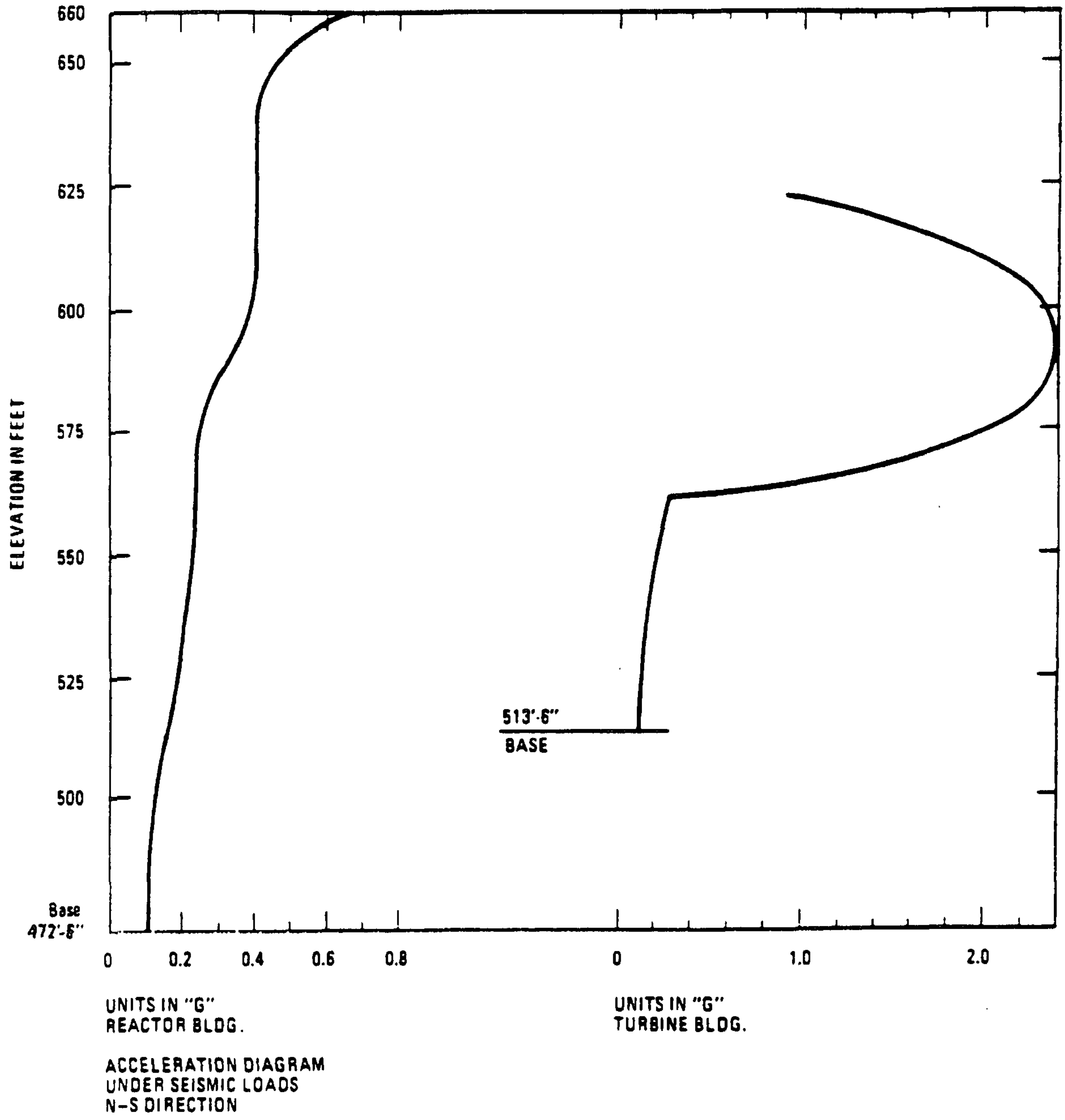
DRESDEN STATION UNITS 2 & 3
VENT HEADER SUPPORT COLLAR PLATE DETAILS
FIGURE 3.8-31



DRESDEN STATION
UNITS 2 & 3

VENT SYSTEM SUPPORT COLUMN DETAILS

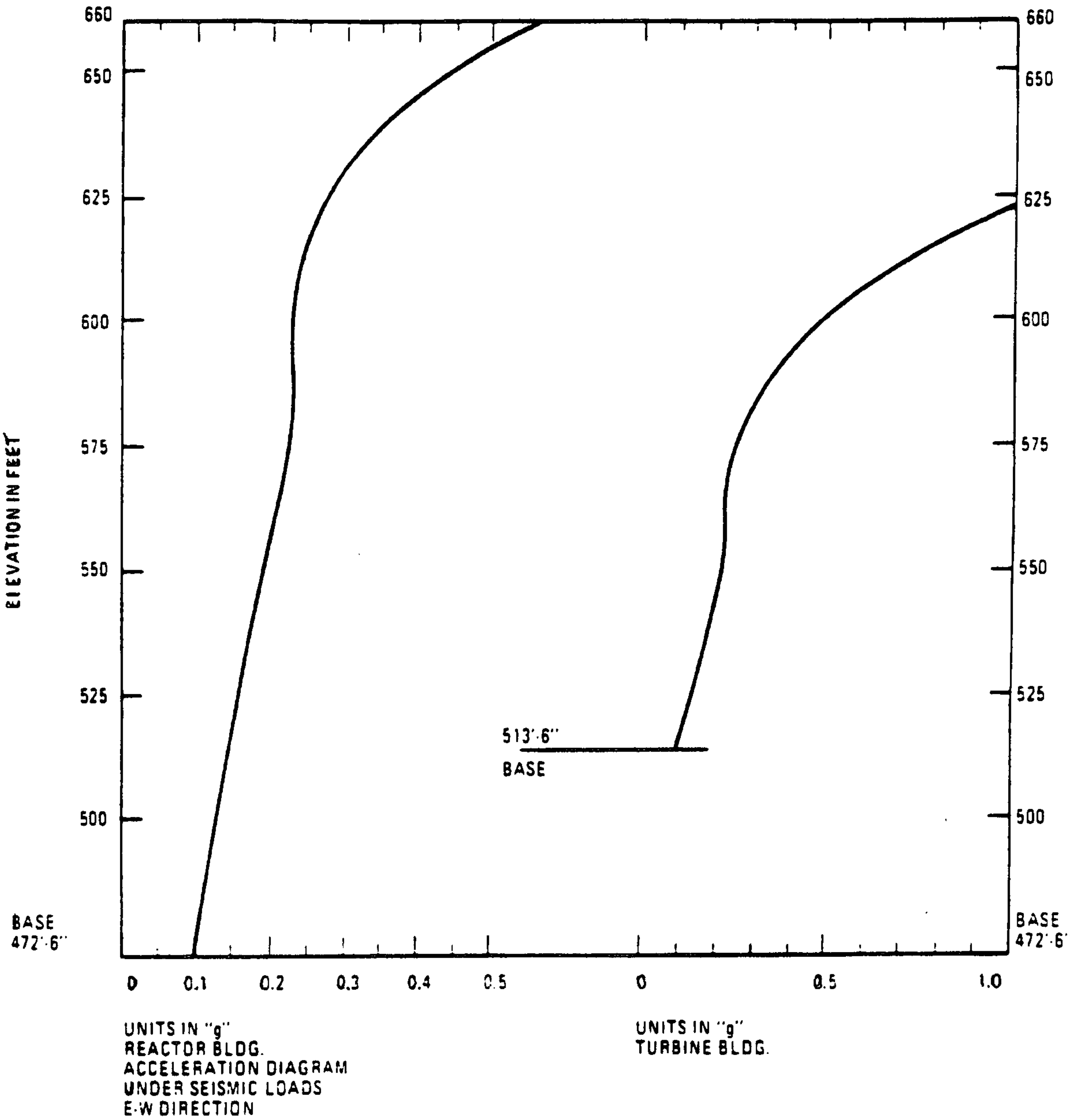
FIGURE 3.8-32



DRESDEN STATION
UNITS 2 & 3

OBE SEISMIC ACCELERATION DIAGRAM.
REACTOR & TURBINE BUILDINGS
(N-S DIRECTION)

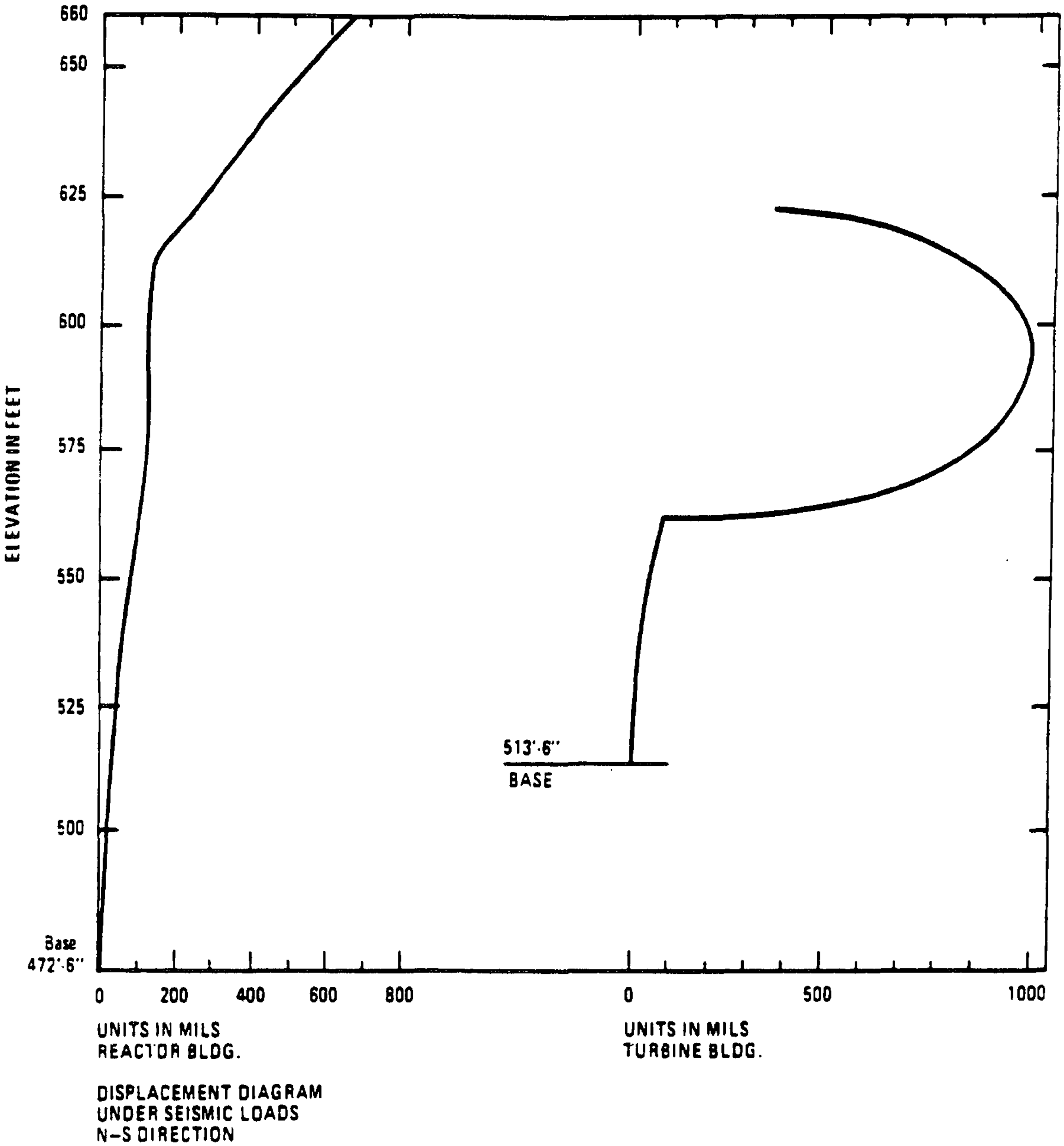
FIGURE 3.8-33



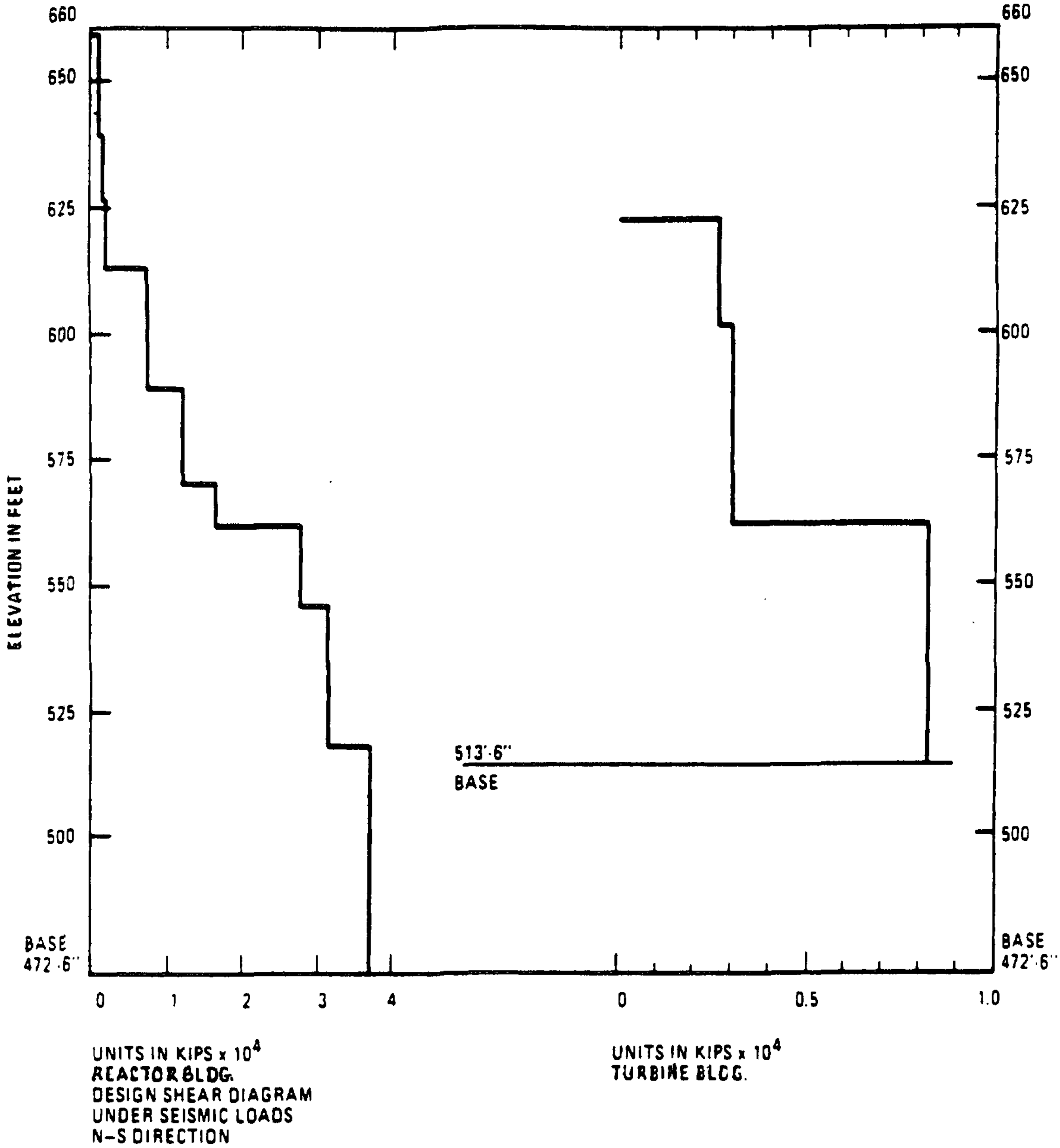
DRESDEN STATION
UNITS 2 & 3

OBE SEISMIC ACCELERATION DIAGRAM.
REACTOR & TURBINE BUILDINGS
(E-W DIRECTION)

FIGURE 3.8-34



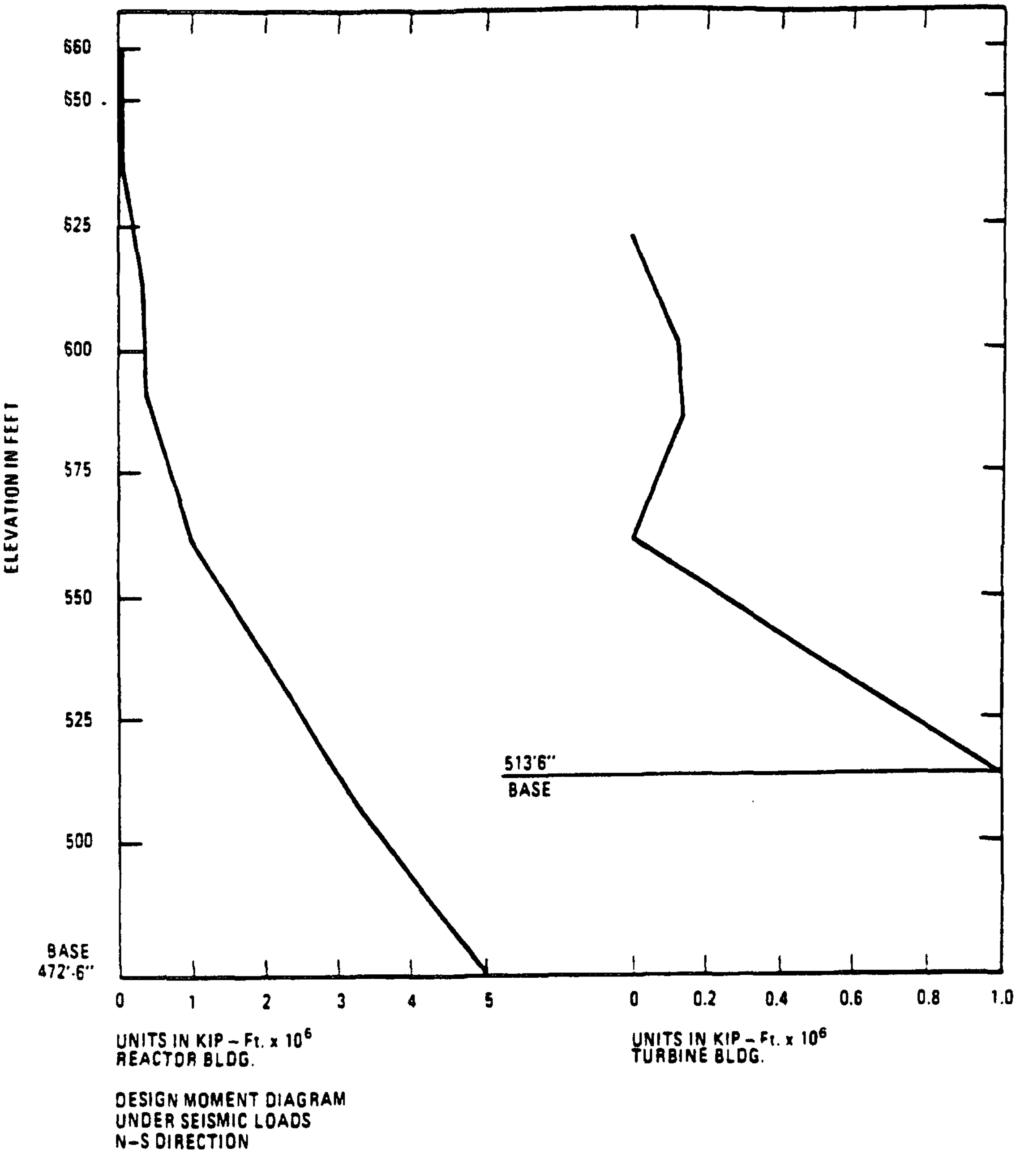
DRESDEN STATION UNITS 2 & 3
REACTOR & TURBINE BUILDINGS OBE SEISMIC DISPLACEMENT DIAGRAM (N-S DIRECTION)
FIGURE 3.8-35



DRESDEN STATION
UNITS 2 & 3

REACTOR & TURBINE BUILDINGS OBE
SEISMIC SHEAR DIAGRAM
(N-S DIRECTION)

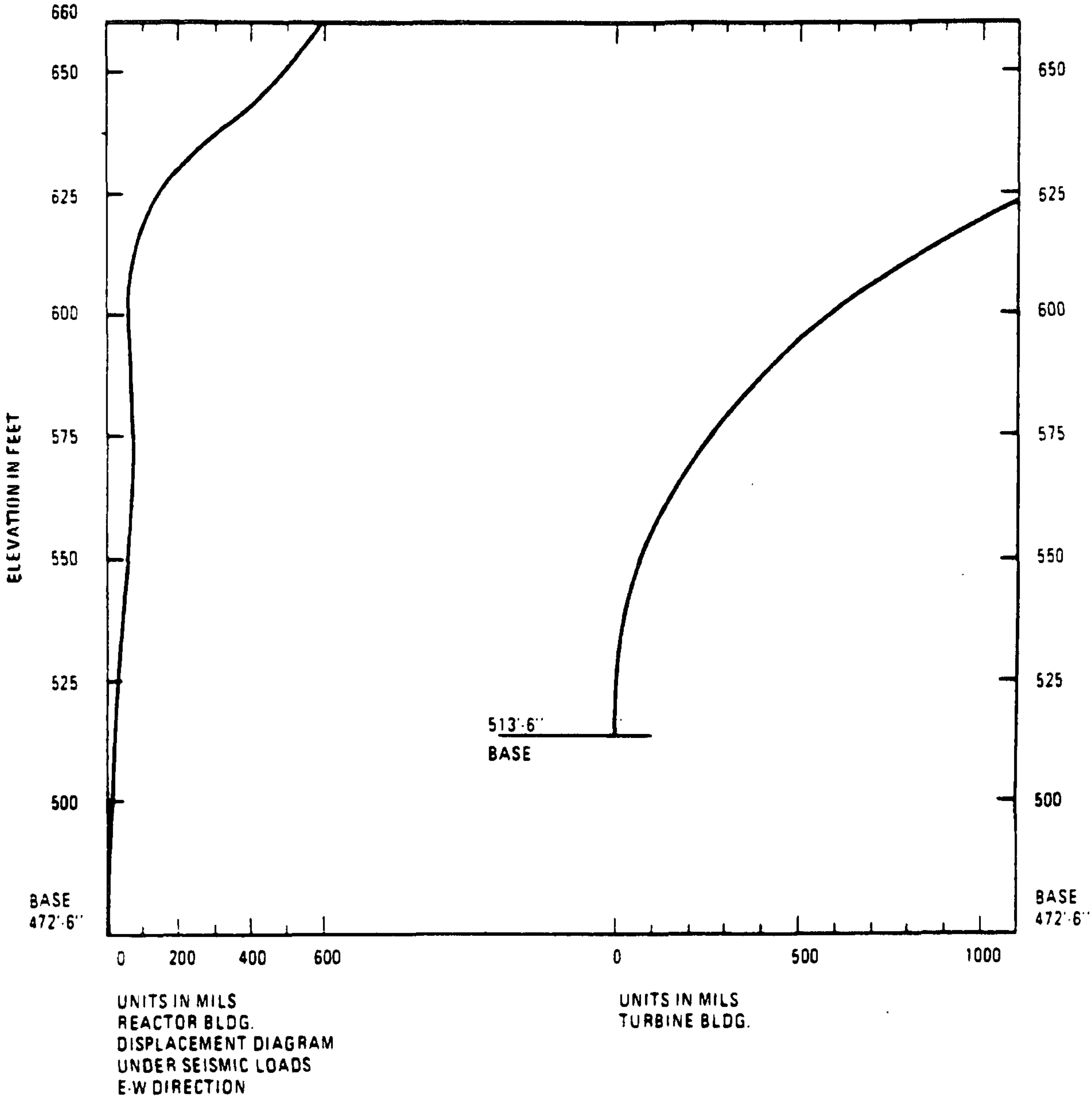
FIGURE 3.8-36



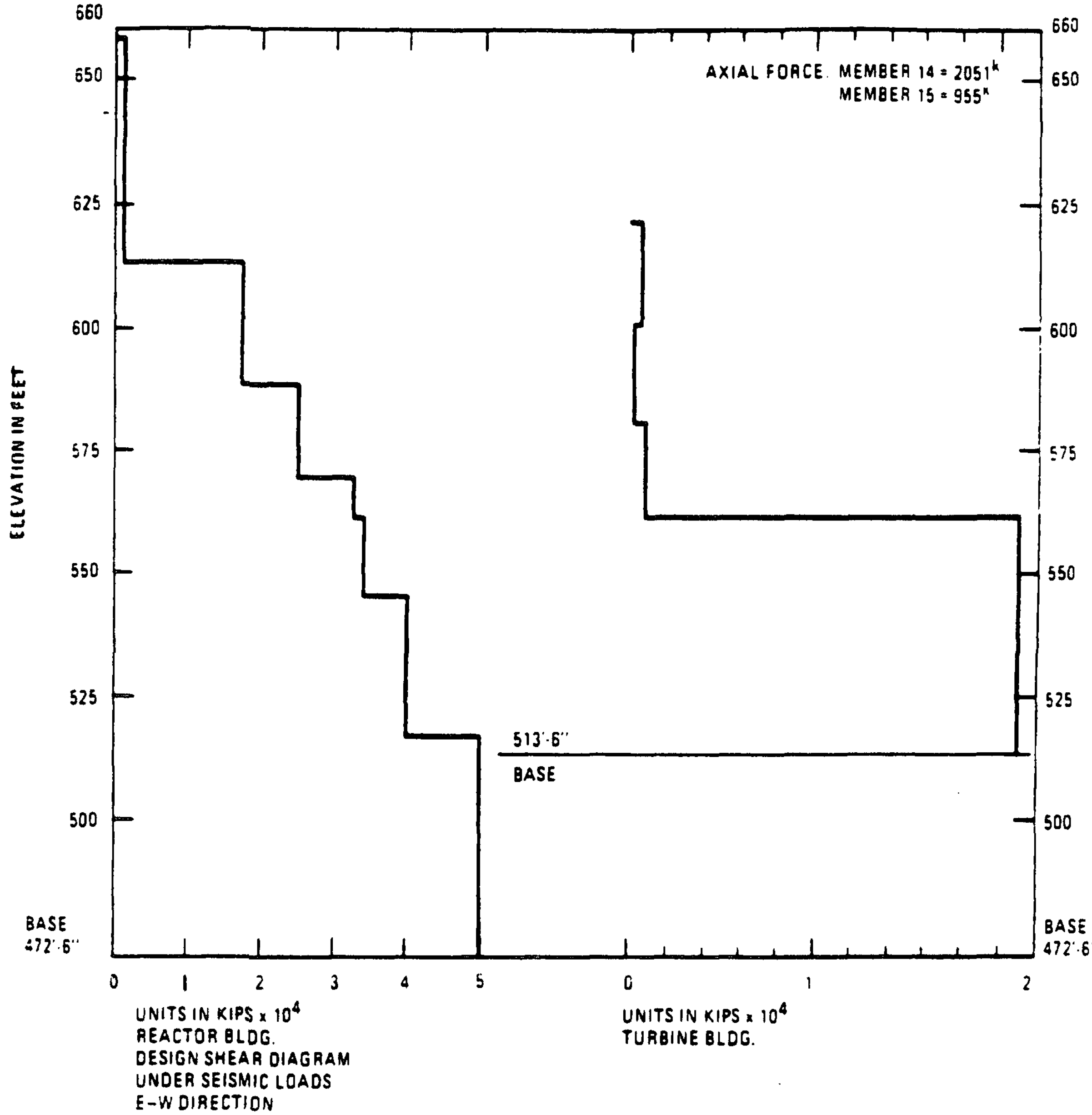
DRESDEN STATION
UNITS 2 & 3

REACTOR & TURBINE BUILDINGS OBE
SEISMIC MOMENT DIAGRAM
(N-S DIRECTION)

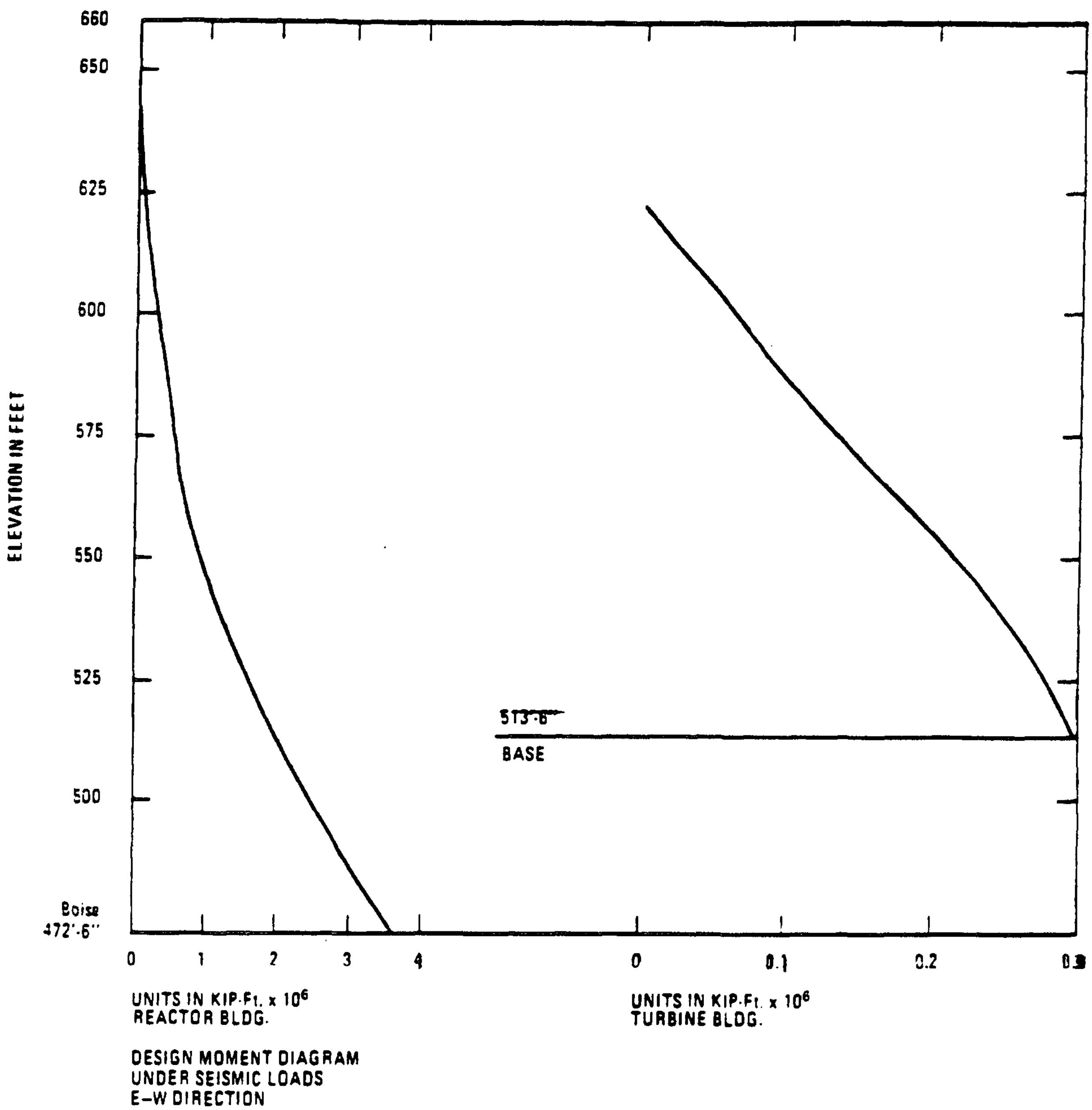
FIGURE 3.8-37



DRESDEN STATION UNITS 2 & 3
REACTOR & TURBINE BUILDINGS OBE SEISMIC DISPLACEMENT DIAGRAM (E-W DIRECTION)
FIGURE 3.8-38



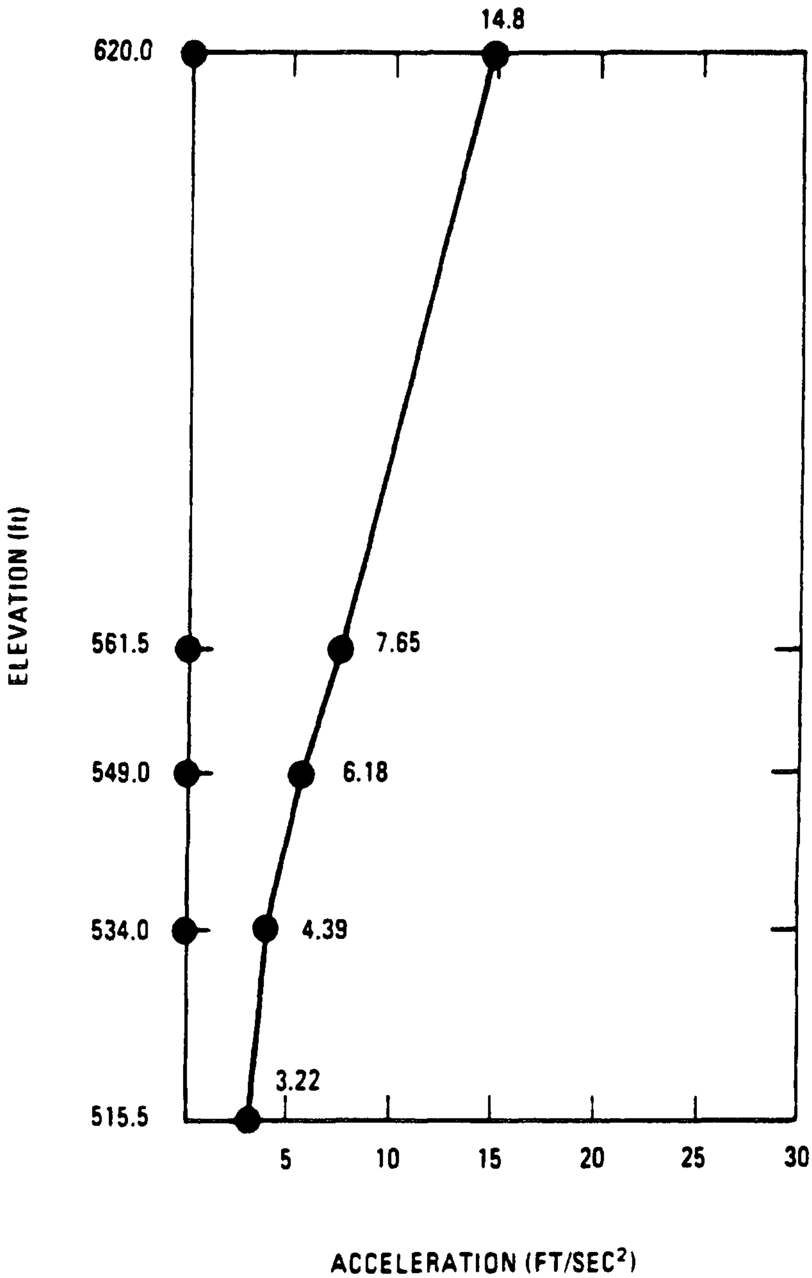
DRESDEN STATION UNITS 2 & 3
REACTOR & TURBINE BUILDINGS OBE SEISMIC SHEAR DIAGRAM (E-W DIRECTION)
FIGURE 3.8-39



DRESDEN STATION
UNITS 2 & 3

REACTOR & TURBINE BUILDINGS OBE
SEISMIC MOMENT DIAGRAM
(E-W DIRECTION)

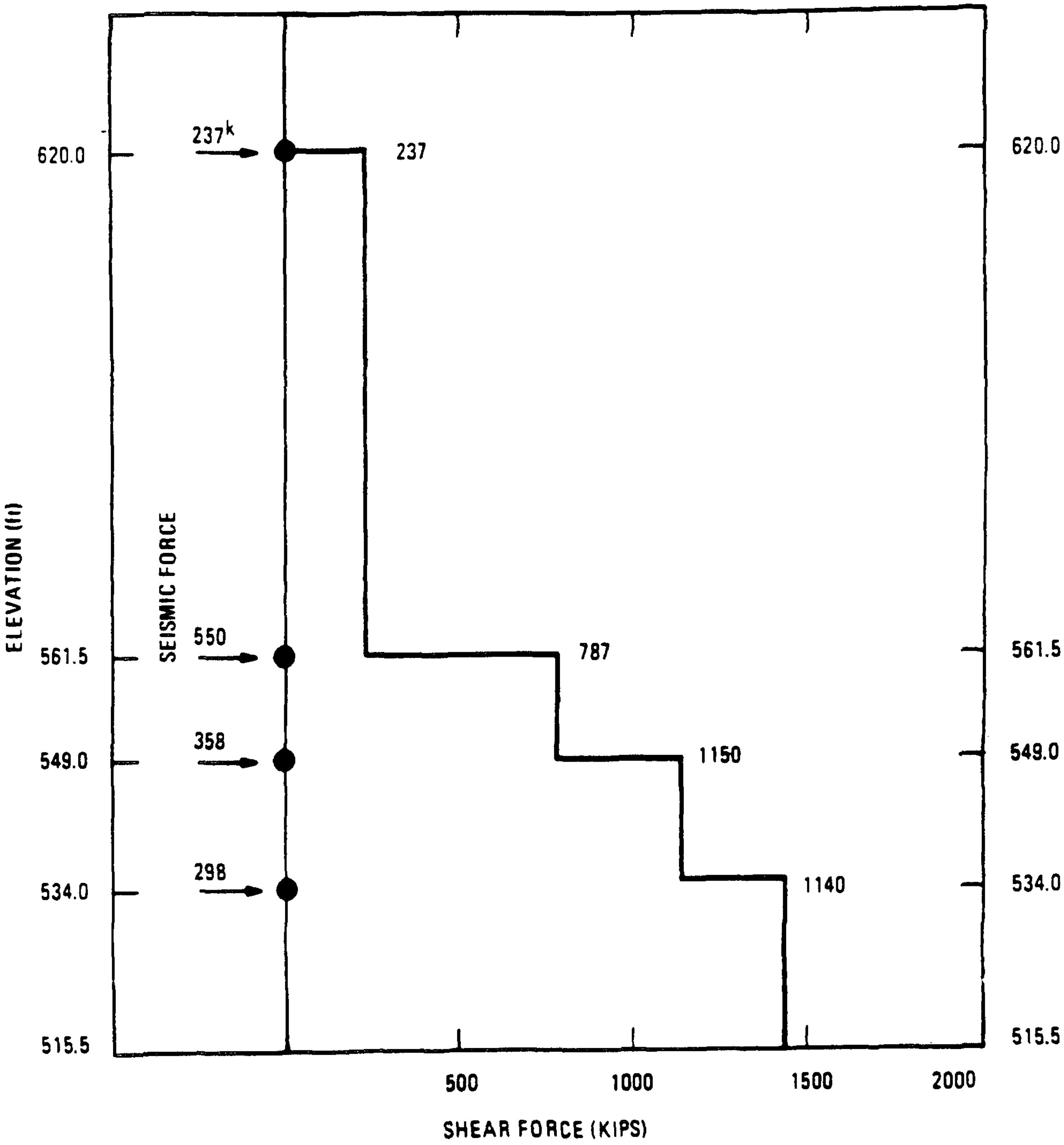
FIGURE 3.8-40



DRESDEN STATION
UNITS 2 & 3

CONTROL ROOM OBE ACCELERATING DIAGRAM (ABOUT MINOR PRINCIPAL AXIS - 5% DAMPING)

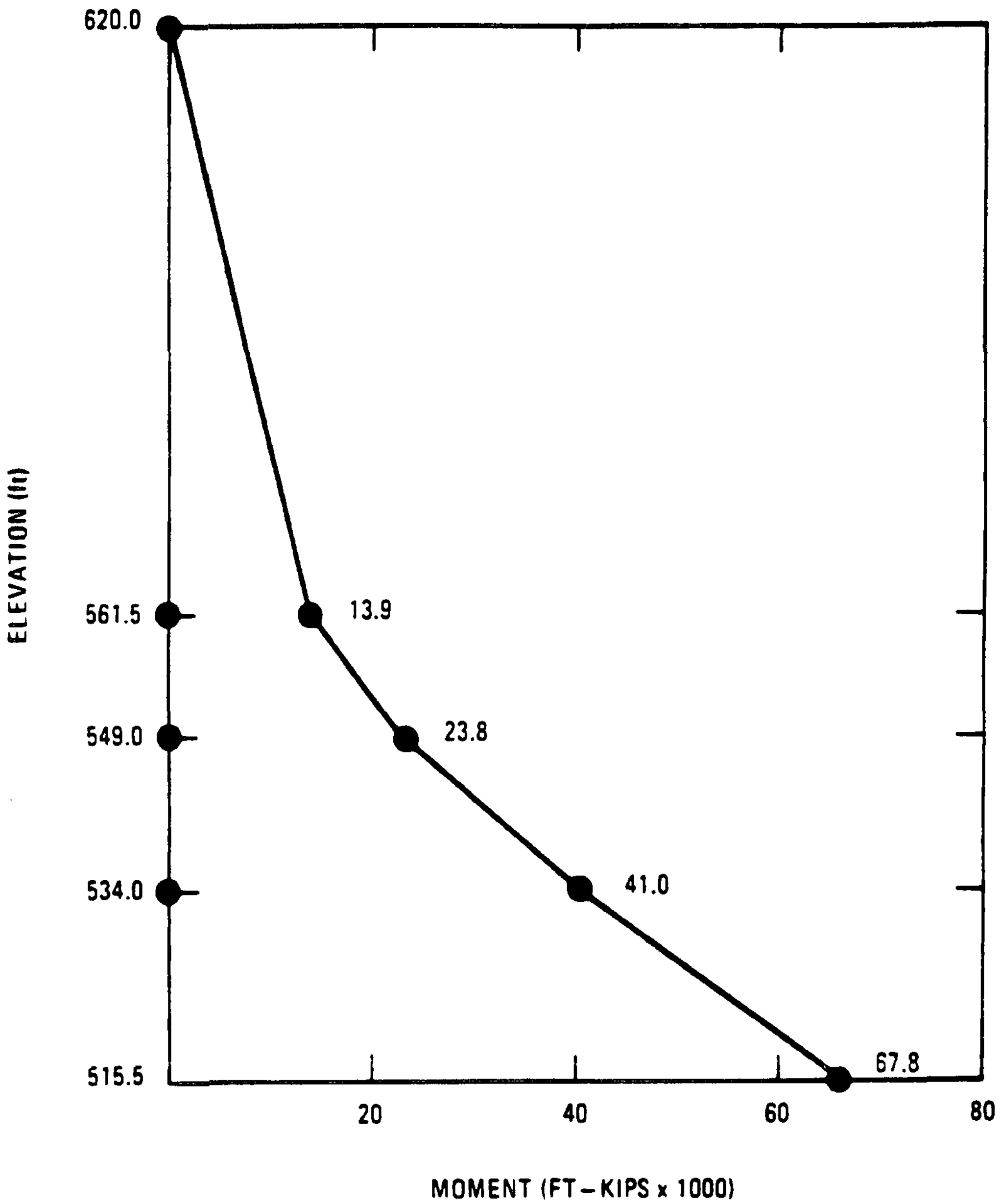
FIGURE 3.8-41



DRESDEN STATION
UNITS 2 & 3

CONTROL ROOM OBE SHEAR DIAGRAM

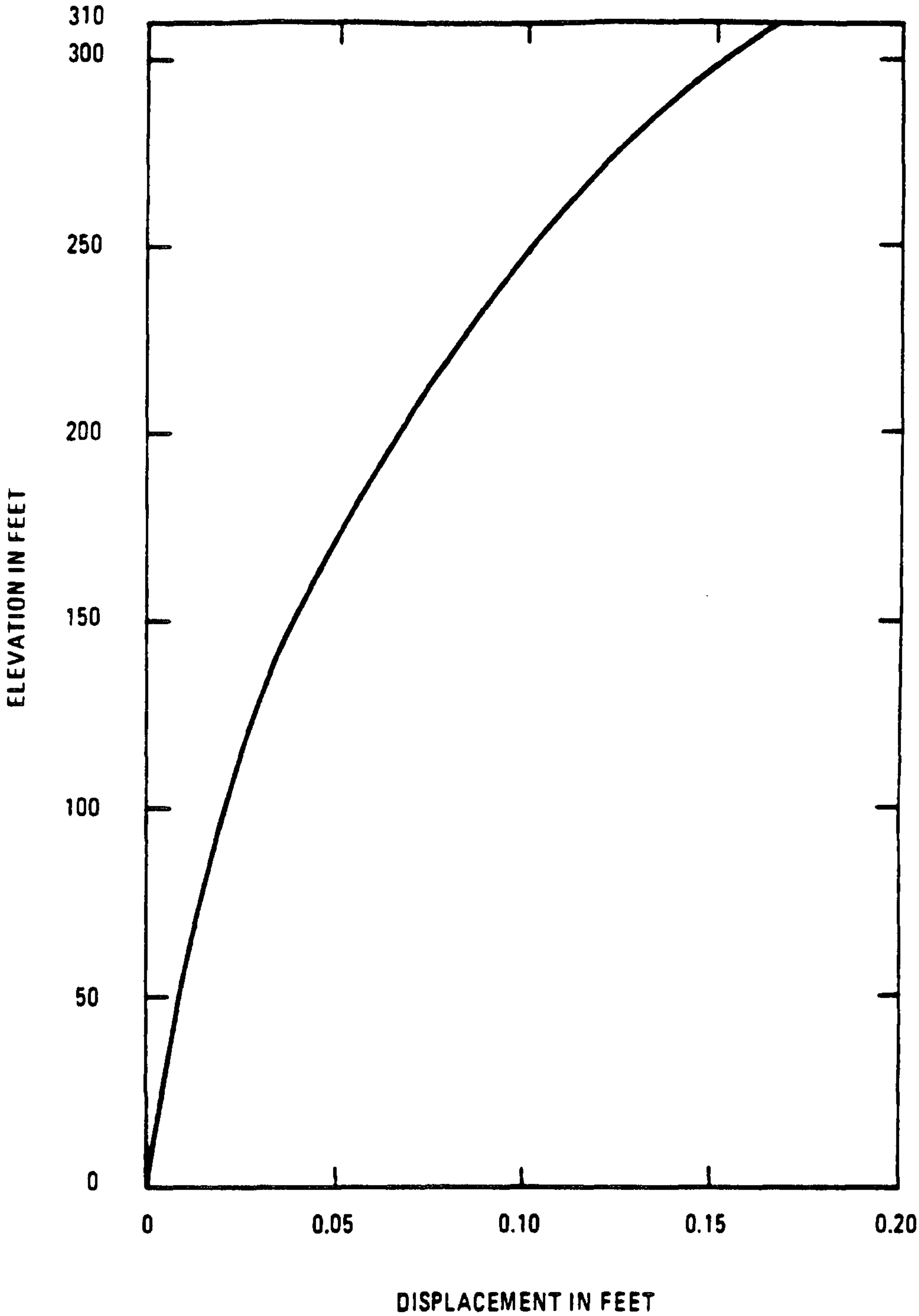
FIGURE 3.8-42



DRESDEN STATION
UNITS 2 & 3

CONTROL ROOM OBE MOMENT DIAGRAM

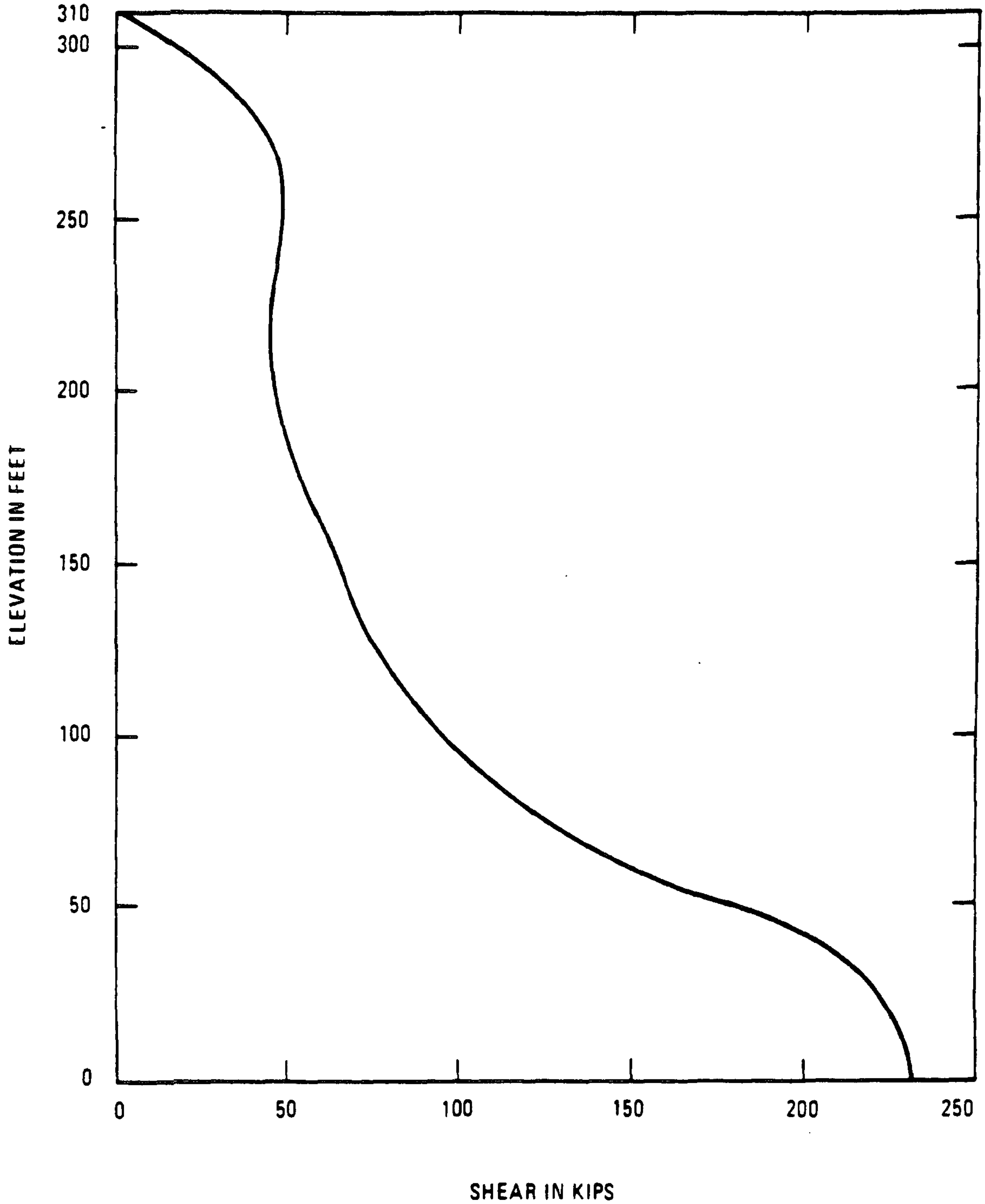
FIGURE 3.8-43



DRESDEN STATION
UNITS 2 & 3

CONCRETE CHIMNEY OBE DISPLACEMENT DIAGRAM

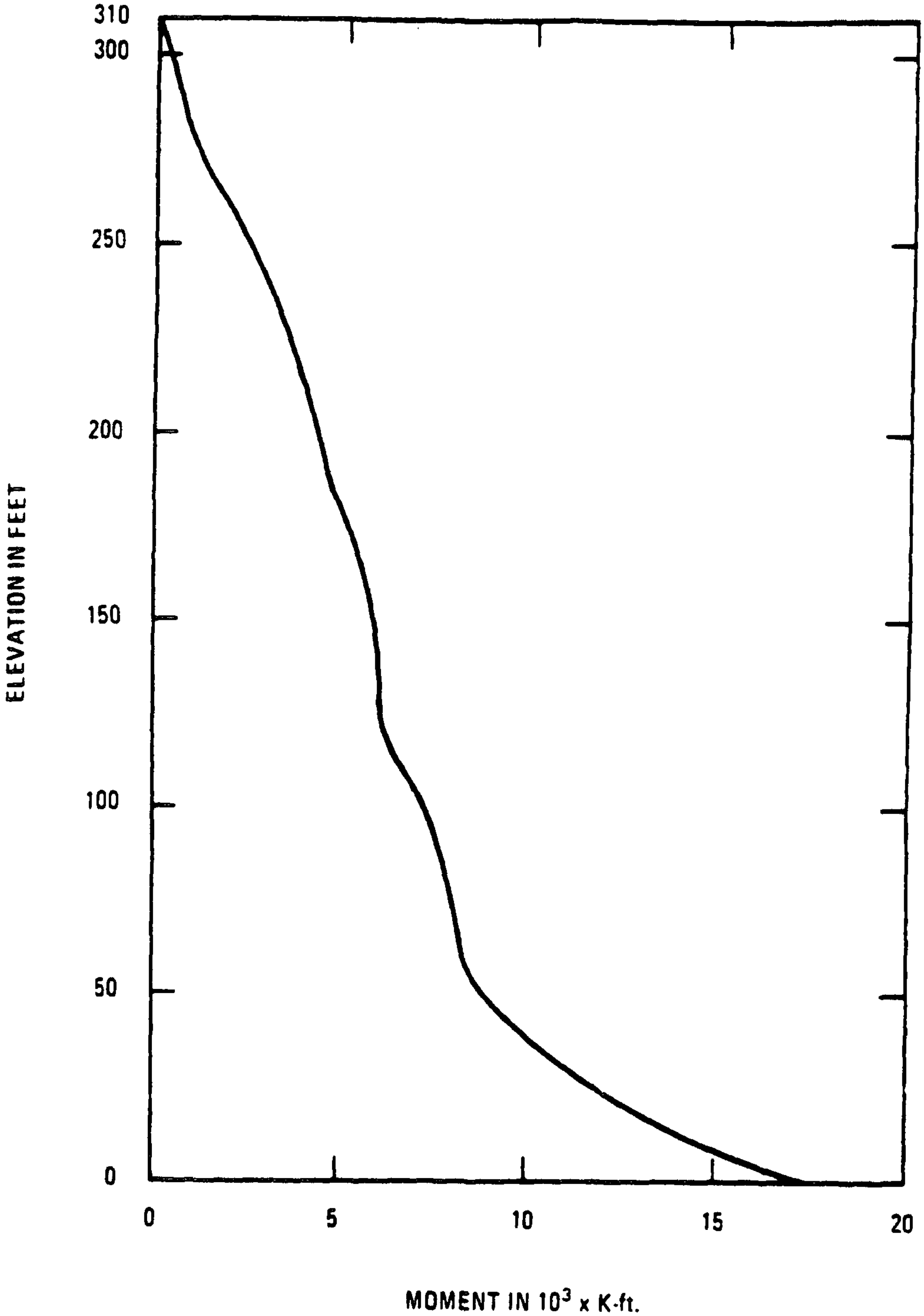
FIGURE 3.8-44



DRESDEN STATION
UNITS 2 & 3

CONCRETE CHIMNEY OBE SHEAR DIAGRAM

FIGURE 3.8-45

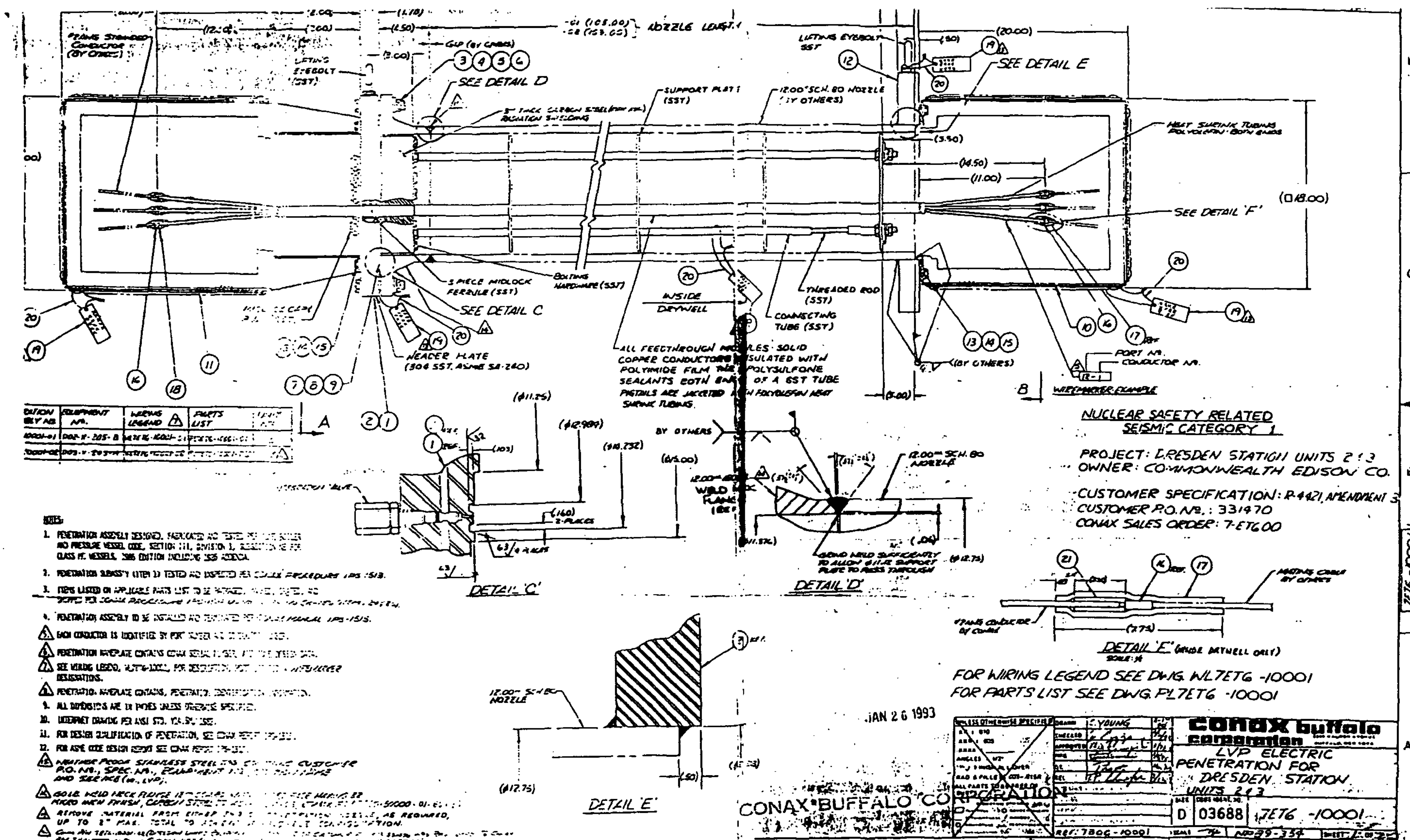


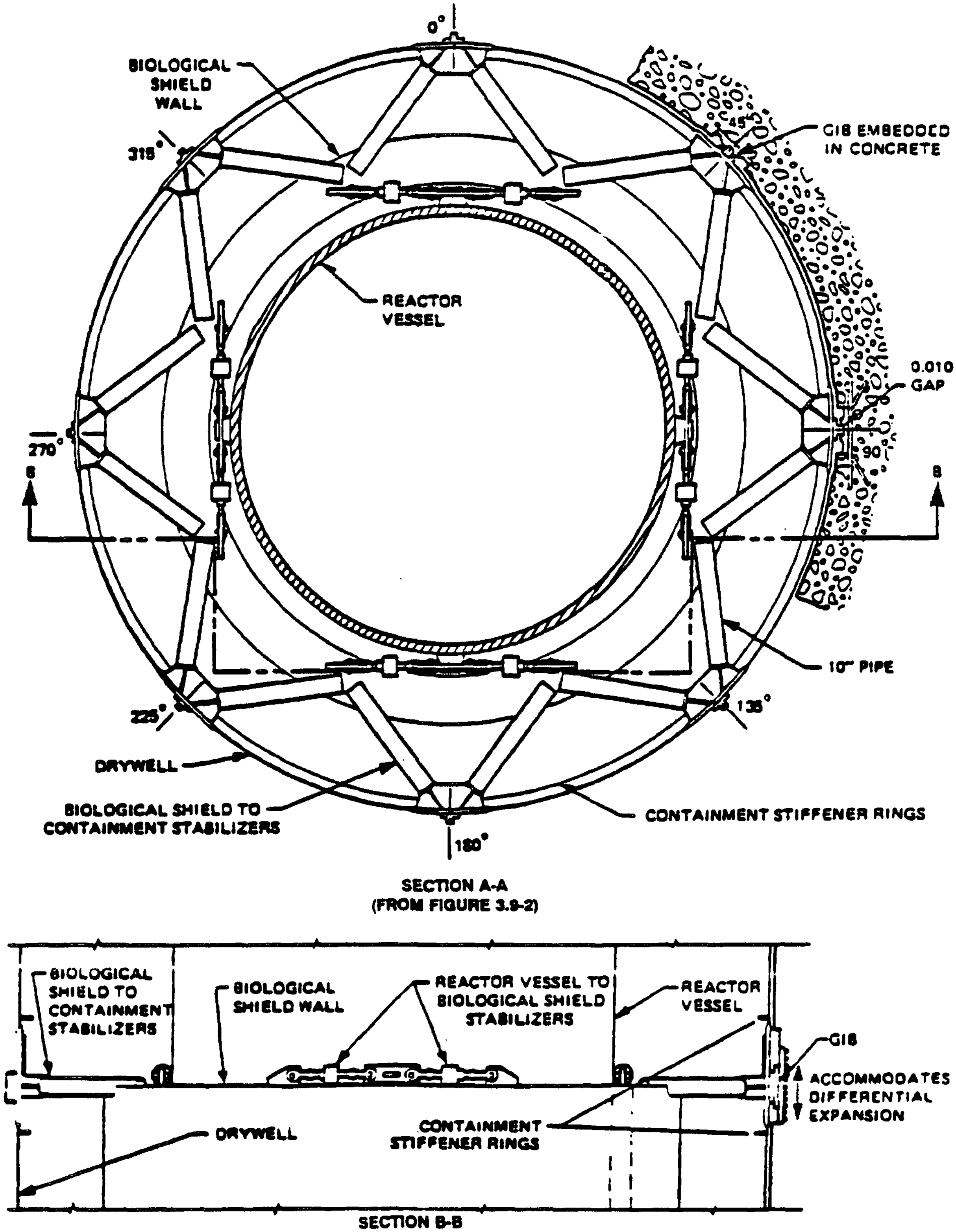
DRESDEN STATION
UNITS 2 & 3

CONCRETE CHIMNEY OBE MOMENT DIAGRAM

FIGURE 3.8-46

FIGURE 3.8-47

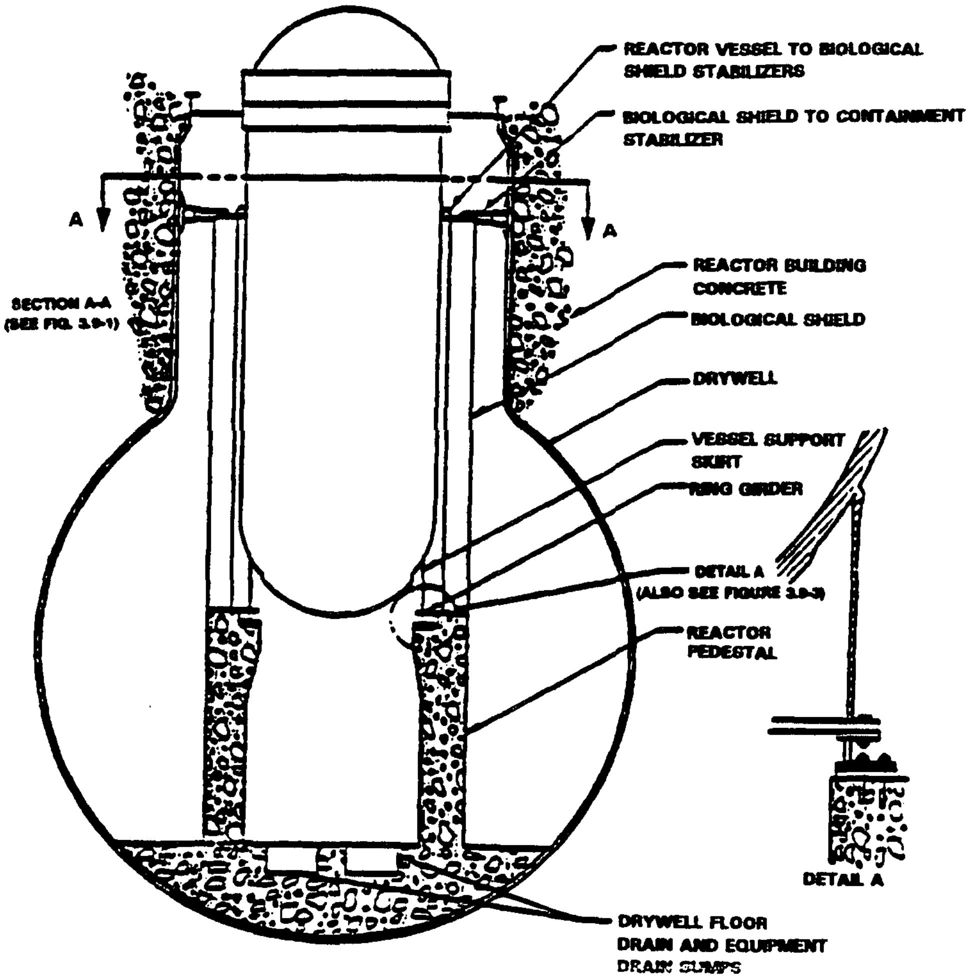

SEE DETAIL D
SEE DETAIL E
SEE DETAIL C
SEE DETAIL 'F'
SUPPORT PLATE (SST)
12.00" SCH. 80 NOZZLE (BY OTHERS)
HEAT SHRINK TUBING POLYOLEFIN BOTH ENDS
INSIDE DRYWELL
CONNECTING TUBE (SST)
THREADED ROD (SST)
HEADER PLATE (304 SST, ASME SA-240)
ALL FEEDTHROUGH MODULES SOLID COPPER CONDUCTORS INSULATED WITH POLYIMIDE FILM THE POLYSULFONE SEALANTS BOTH ENDS OF A SST TUBE PIGTAILS ARE JACKETED WITH POLYOLEFIN HEAT SHRINK TUBING.
WIREMARKER EXAMPLE
PORT NO.
CONDUCTOR NO.
NUCLEAR SAFETY RELATED
SEISMIC CATEGORY 1
PROJECT: DRESDEN STATION UNITS 2 & 3
OWNER: COMMONWEALTH EDISON CO.
CUSTOMER SPECIFICATION: R-4421, AMENDMENT 3
CUSTOMER P.O. NO.: 331470
CONAX SALES ORDER: 7-ET600
DETAIL 'C'
DETAIL 'D'
DETAIL 'E'
DETAIL 'F'
12.00" SCH. 80 NOZZLE
BY OTHERS
FOR WIRING LEGEND SEE DWG. WL7ET6 -10001
FOR PARTS LIST SEE DWG. PL7ET6 -10001
NOTES:
1. PENETRATION ASSEMBLY DESIGNED, FABRICATED AND TESTED PER THE ASME BOILER AND PRESSURE VESSEL CODE, SECTION III, DIVISION 1, SUBSECTION NE FOR CLASS MC VESSELS, 1986 EDITION INCLUDING 1986 ADDENDA.
2. PENETRATION SUBASSEMBLY (ITEM 1) TESTED AND INSPECTED PER CONAX PROCEDURE IPS-1513.
3. ITEMS LISTED ON APPLICABLE PARTS LIST TO BE PACKAGED, ...
4. PENETRATION ASSEMBLY TO BE INSTALLED AND TERMINATED PER CONAX MANUAL IPS-1515.
5. EACH CONDUCTOR IS IDENTIFIED BY PORT NUMBER AND CONDUCTOR NUMBER.
6. PENETRATION NAMEPLATE CONTAINS CONAX SERIAL NUMBER...
7. SEE WIRING LEGEND, WL7ET6-10001, FOR DESCRIPTION, PORT ... WIREMARKER DESIGNATIONS.
8. PENETRATION NAMEPLATE CONTAINS PENETRATION IDENTIFICATION ...
9. ALL DIMENSIONS ARE IN INCHES UNLESS OTHERWISE SPECIFIED.
10. INTERPRET DRAWING PER ANSI STD. Y14.5M-1982.
11. FOR DESIGN QUALIFICATION OF PENETRATION, SEE CONAX REPORT ...
12. FOR ASME CODE DESIGN REPORT SEE CONAX REPORT ...
JAN 26 1993
CONAX BUFFALO CORPORATION
conax buffalo corporation
LVP ELECTRIC PENETRATION FOR DRESDEN STATION UNITS 2 & 3
D 03688 7ET6 -10001
REF: 7806-10001



DRESDEN STATION
UNITS 2 & 3

REACTOR VESSEL STABILIZER

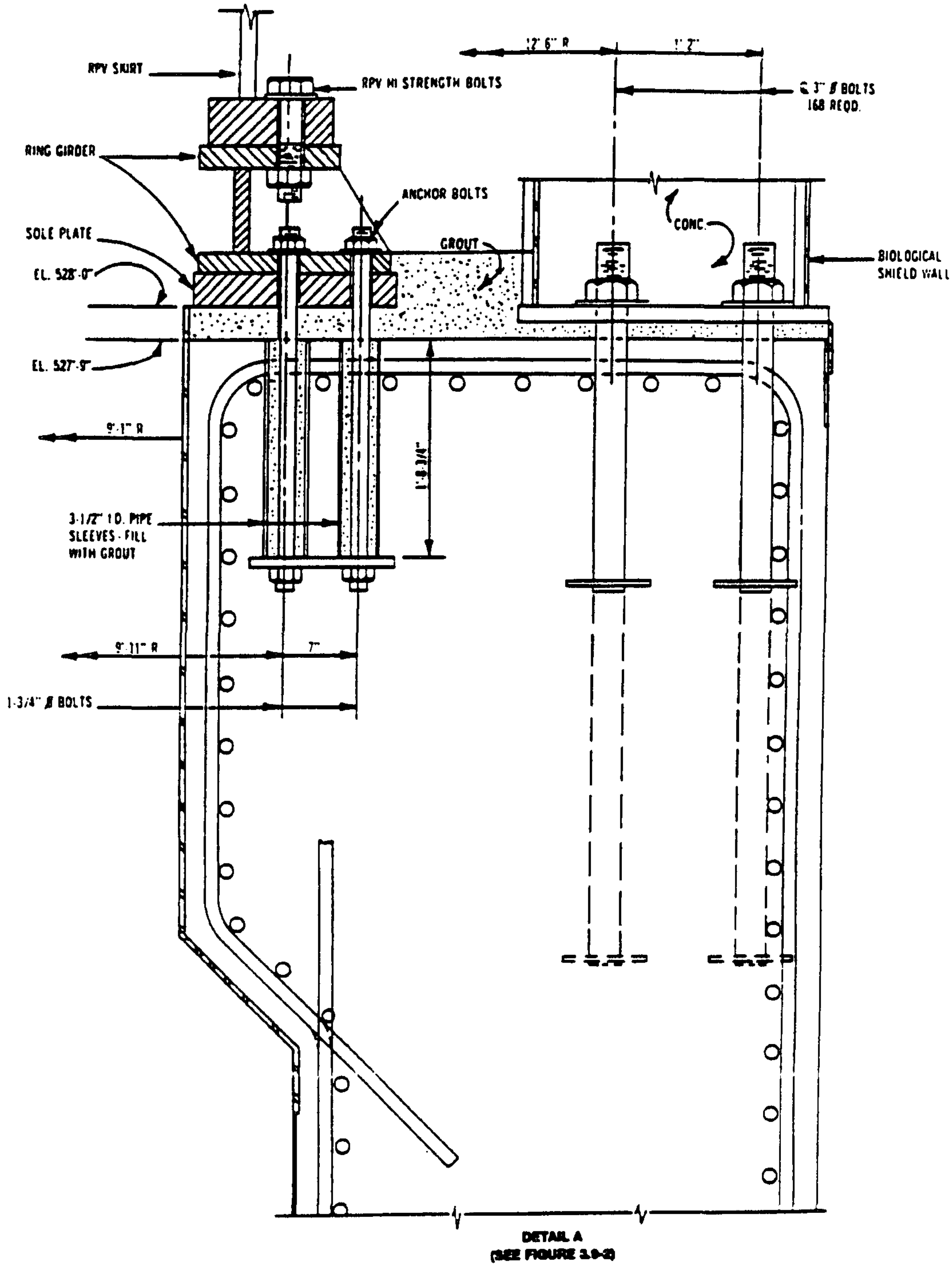
FIGURE 3.9-1



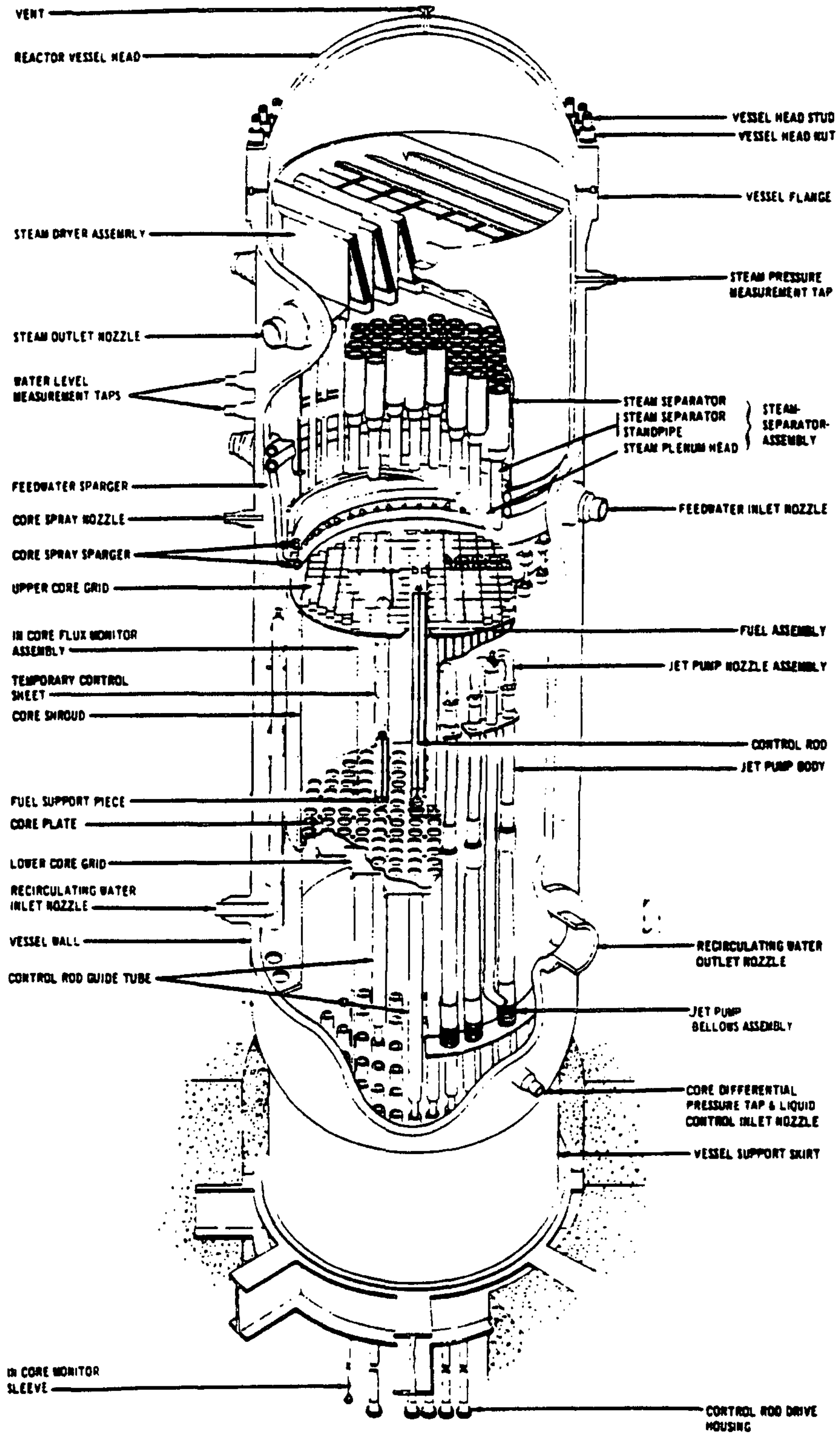
DRESDEN STATION
UNITS 2 & 3

REACTOR PEDESTAL

FIGURE 3.9-2



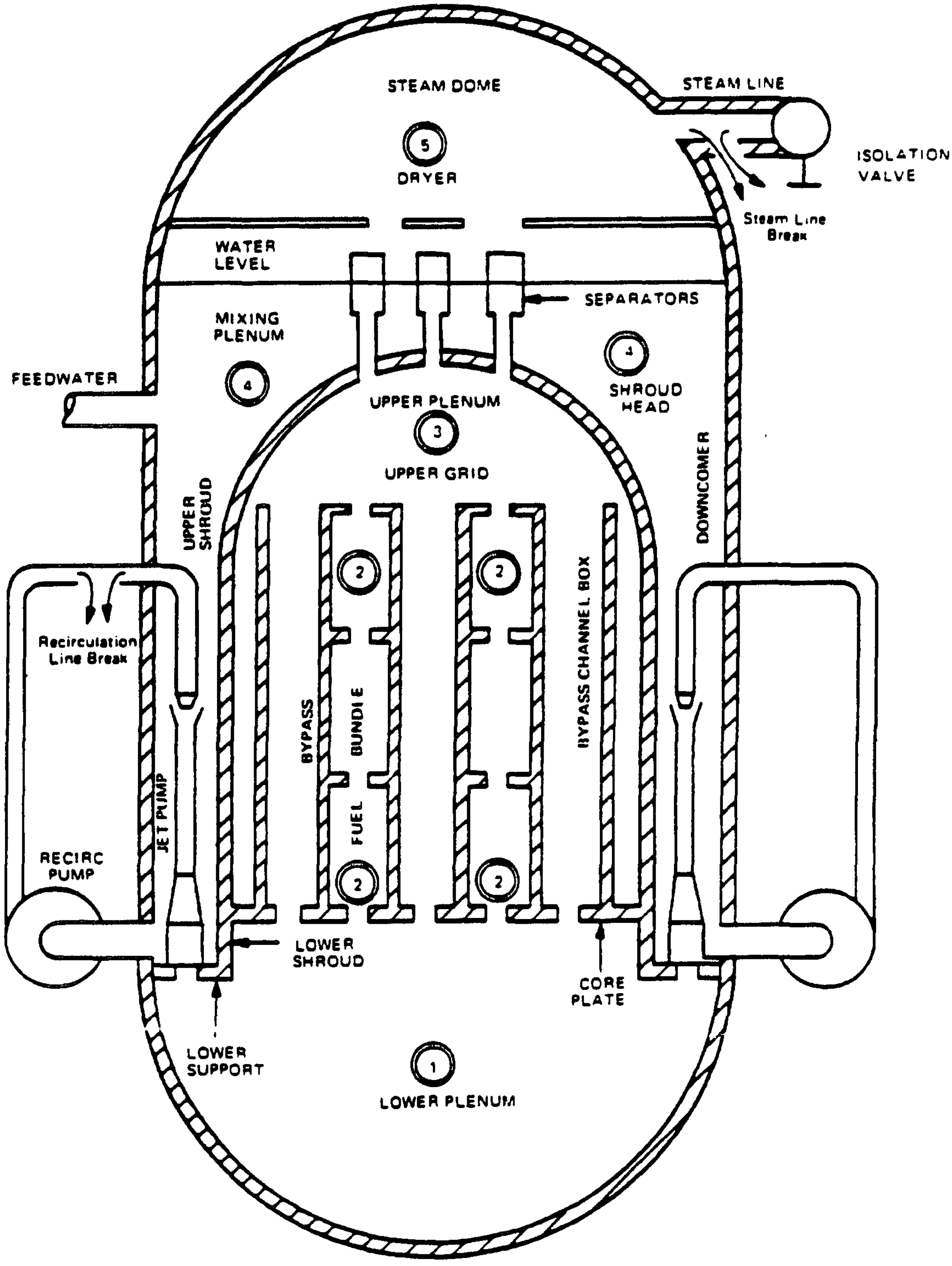
DRESDEN STATION UNITS 2 & 3
REACTOR VESSEL SUPPORT SKIRT ANCHORAGE DETAIL
FIGURE 3.9-3



DRESDEN STATION
UNITS 2 & 3

REACTOR VESSEL AND INTERNALS ISOMETRIC

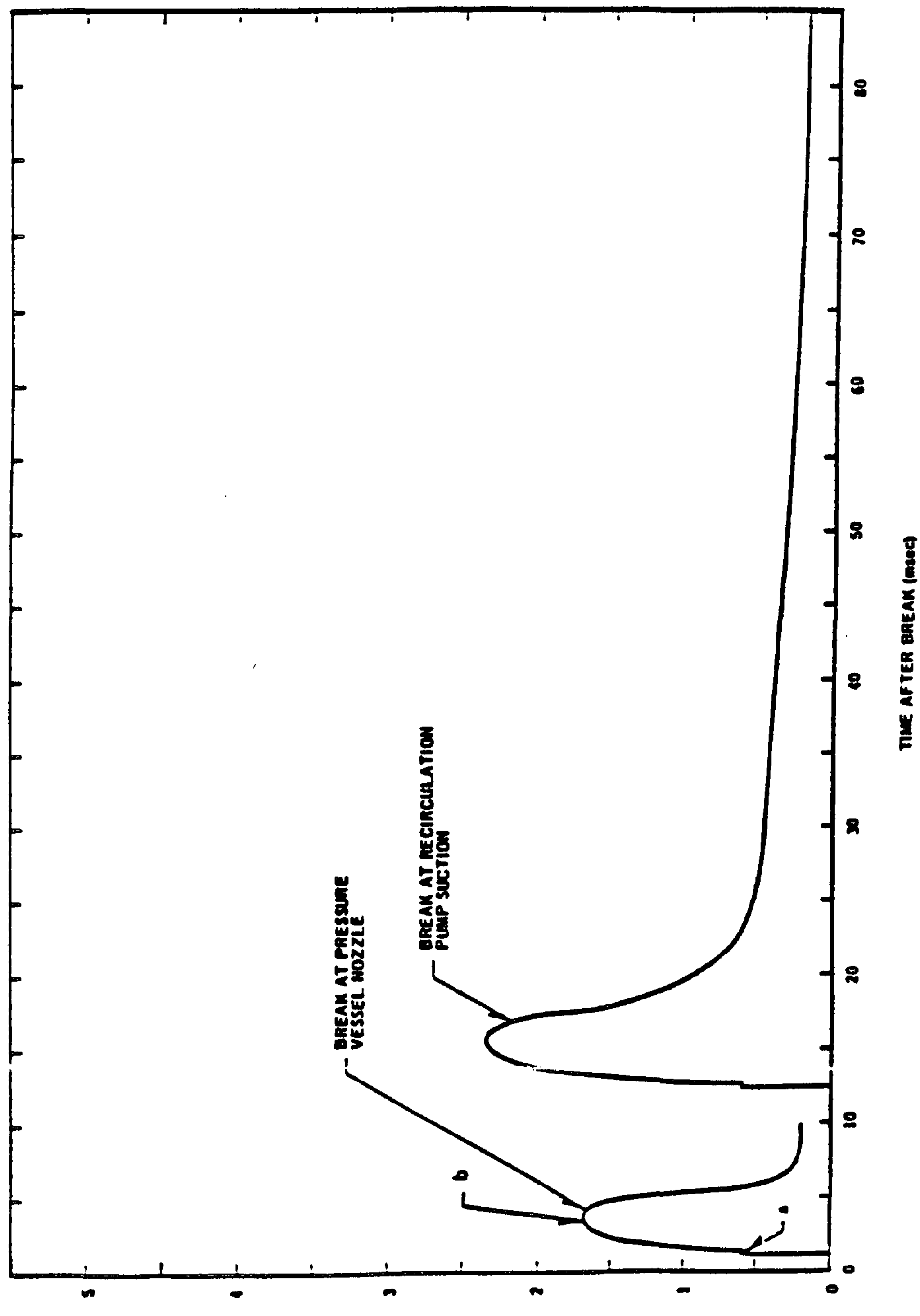
FIGURE 3.9-4



DRESDEN STATION
UNITS 2 & 3

BWR INTERNAL CONFIGURATION
WITH JET PUMP

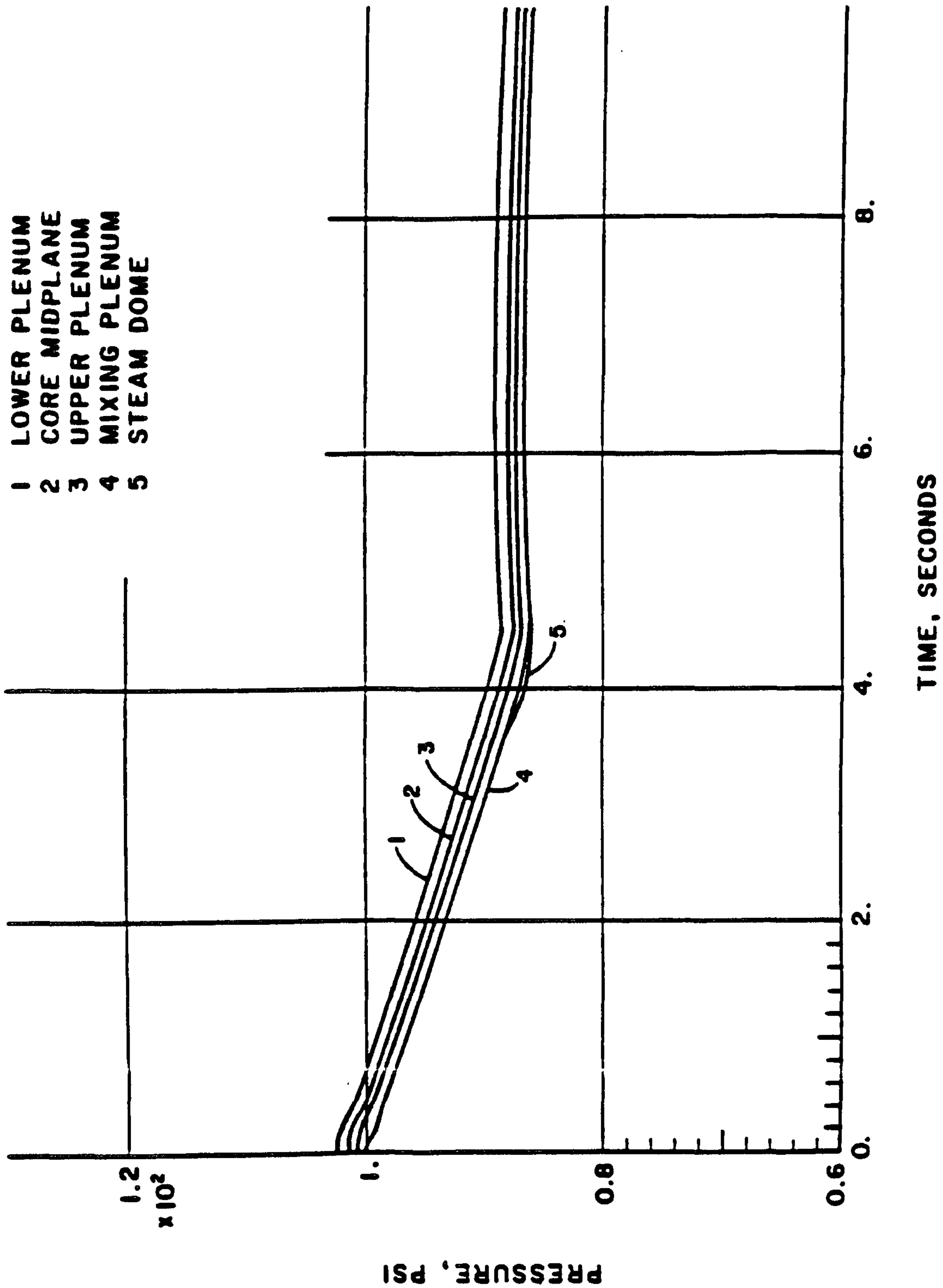
FIGURE 3.9-5



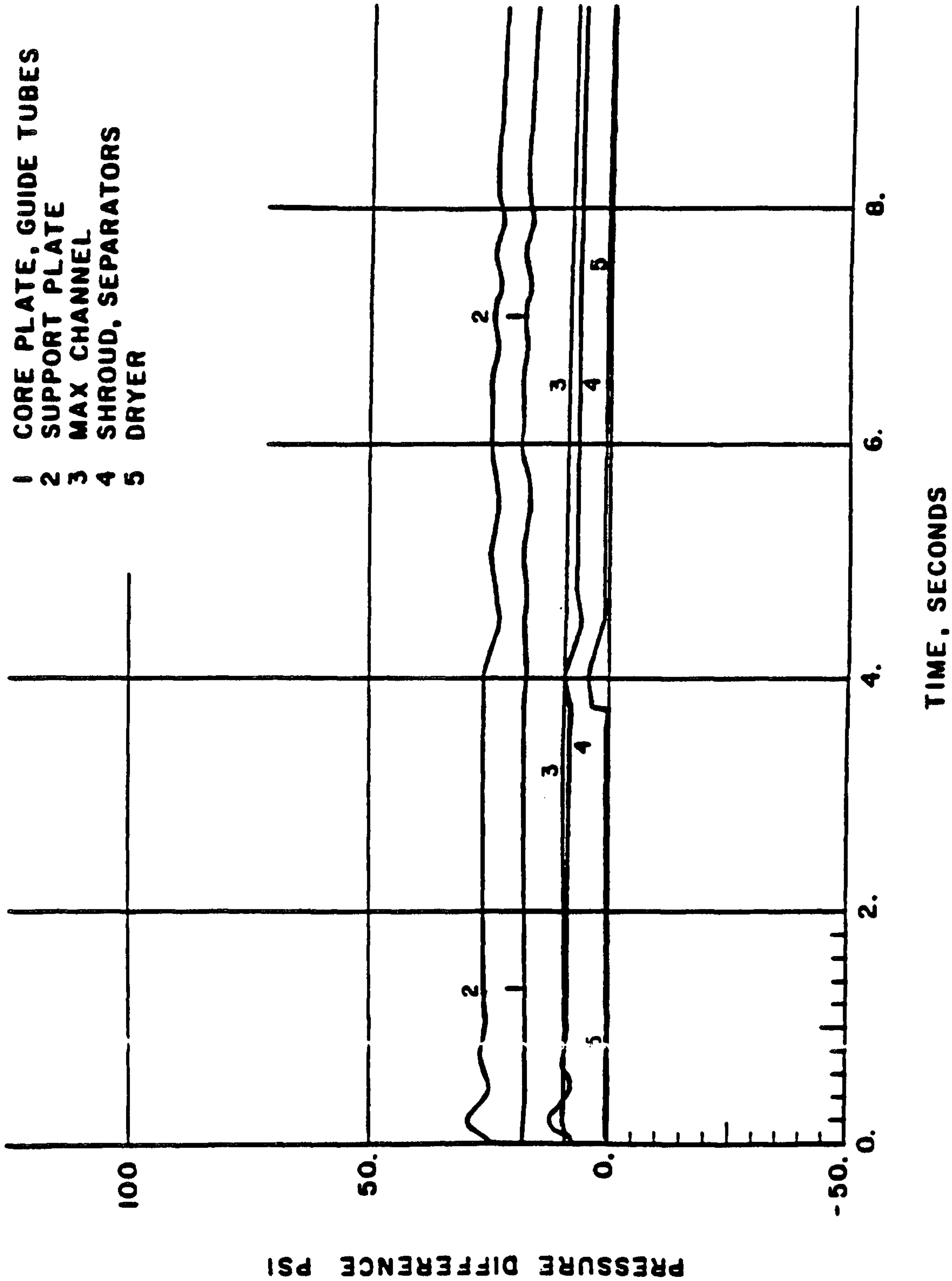
DRESDEN STATION
UNITS 2 & 3

REACTOR VESSEL SHROUD FORCE
FOLLOWING RECIRCULATION LINE BREAK

FIGURE 3.9-6



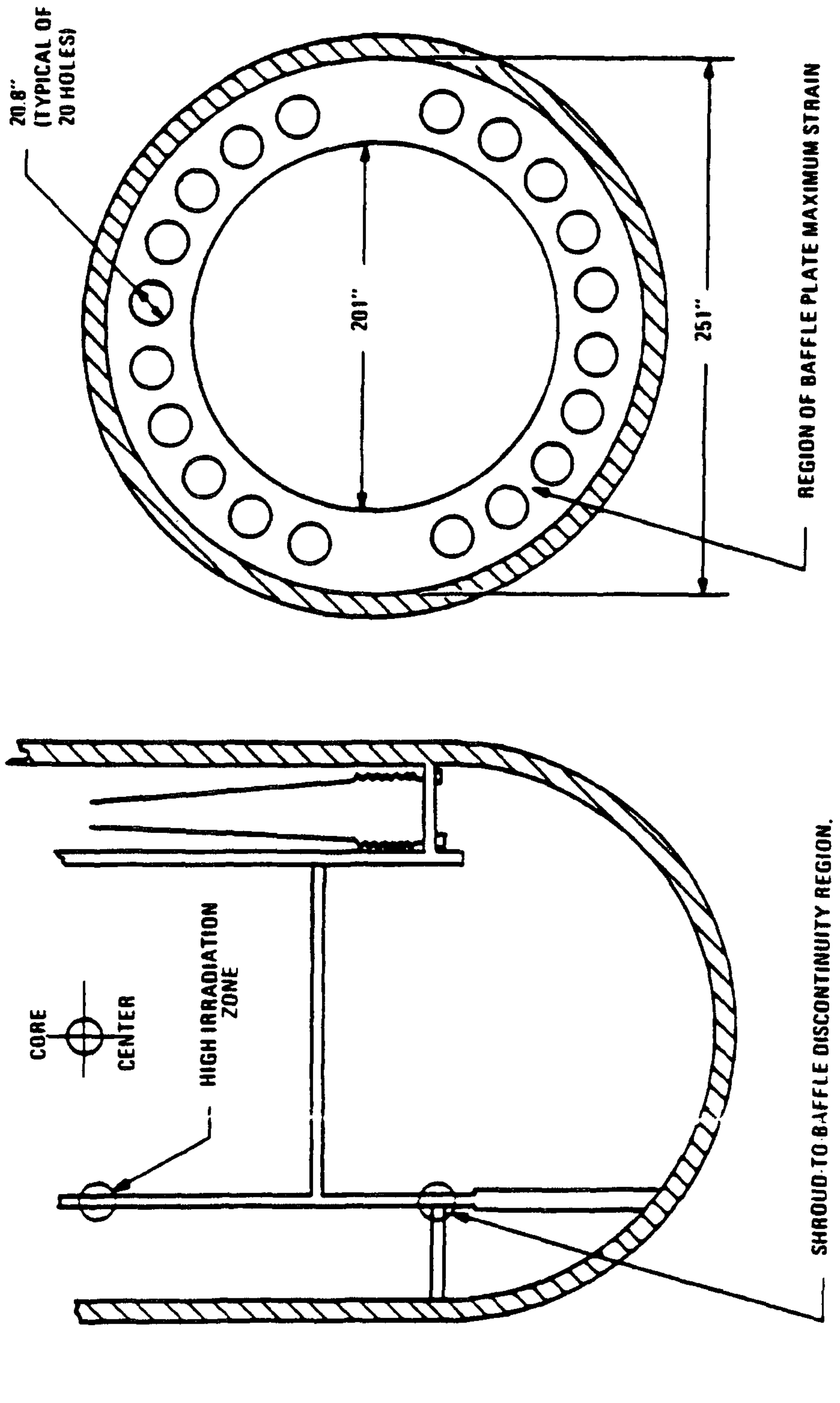
DRESDEN STATION UNITS 2 & 3
BLOWDOWN TRANSIENT REACTOR PRESSURE FROM 2527 MWt
FIGURE 3.9-7



DRESDEN STATION
UNITS 2 & 3

BLOWDOWN TRANSIENT REACTOR
DIFFERENTIAL PRESSURE FROM 2527 MWt

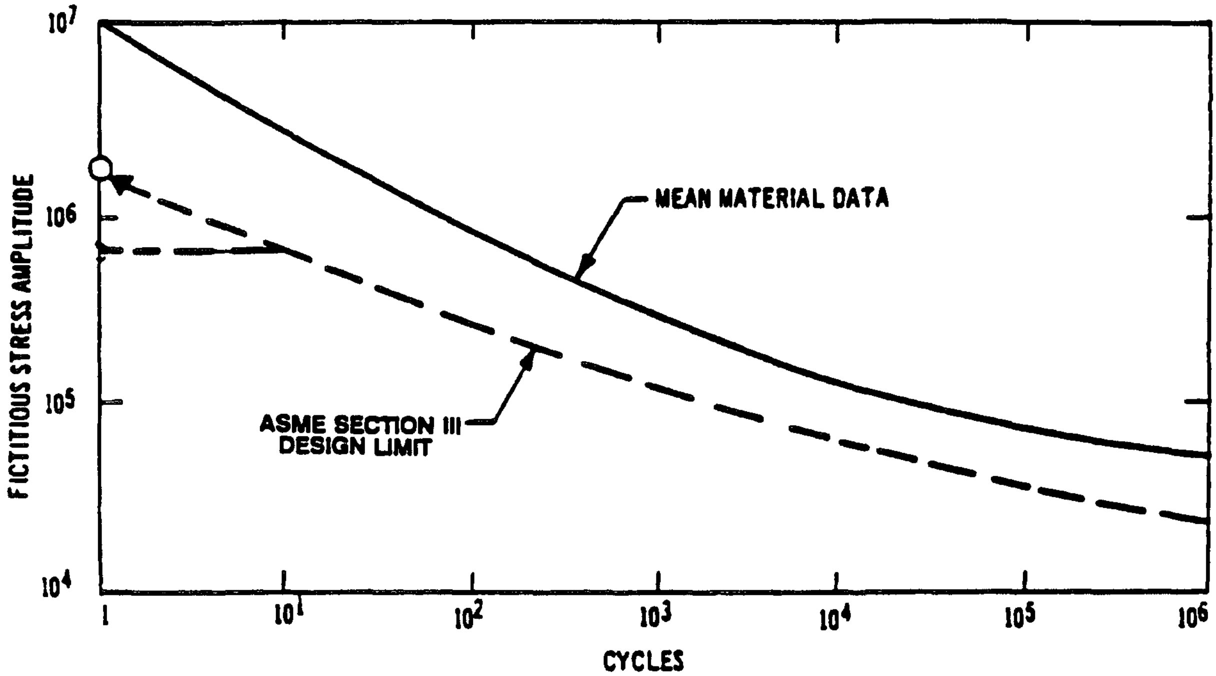
FIGURE 3.9-8



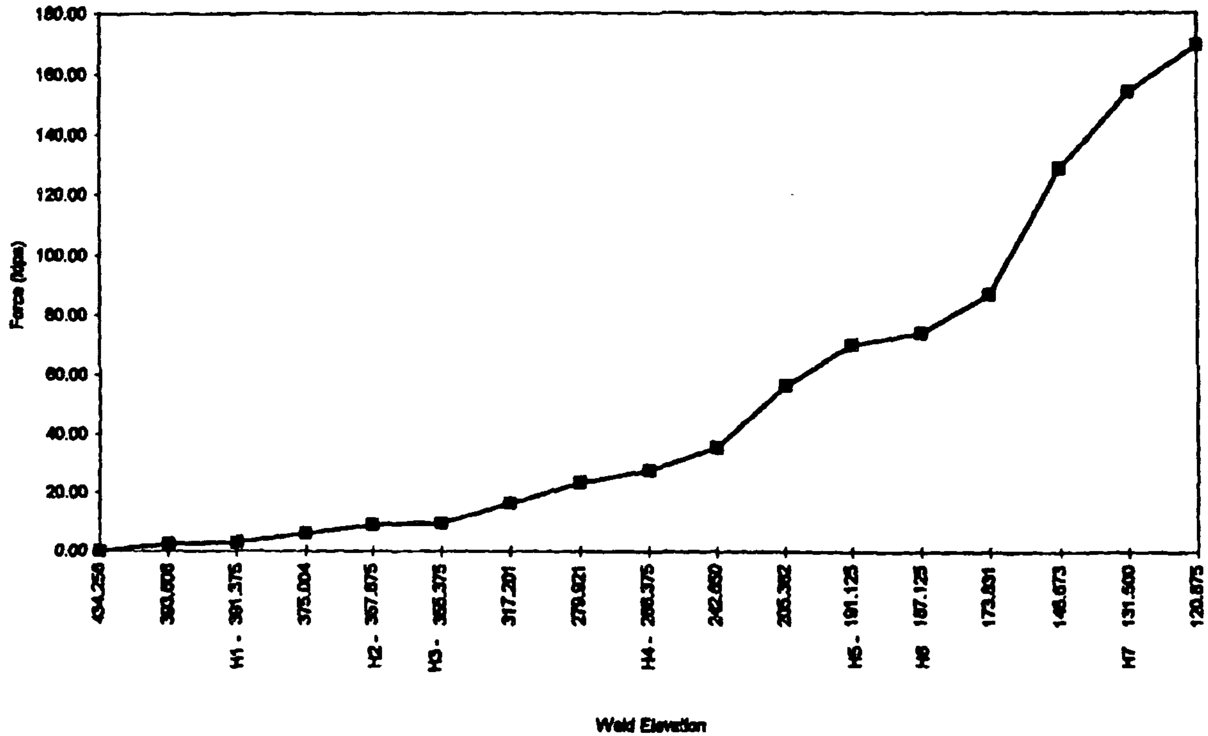
DRESDEN STATION
UNITS 2 & 3

REACTOR INTERNAL BAFFLE PLATE
LIGAMENT STRAIN STUDY

FIGURE 3.9-9



DRESDEN STATION UNITS 2 & 3
FATIGUE CURVE FOR INCONEL AND STAINLESS STEEL
FIGURE 3.9-10

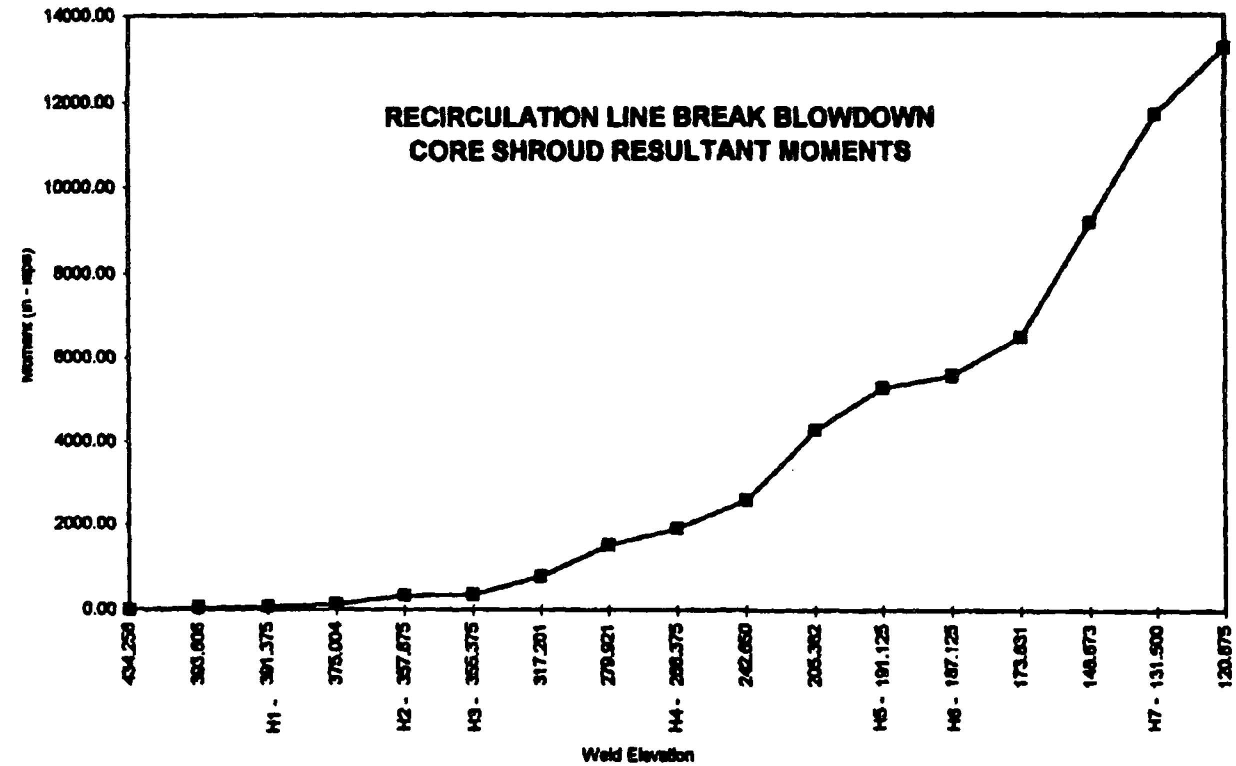


UFSAR REVISION 2, JUNE 1997

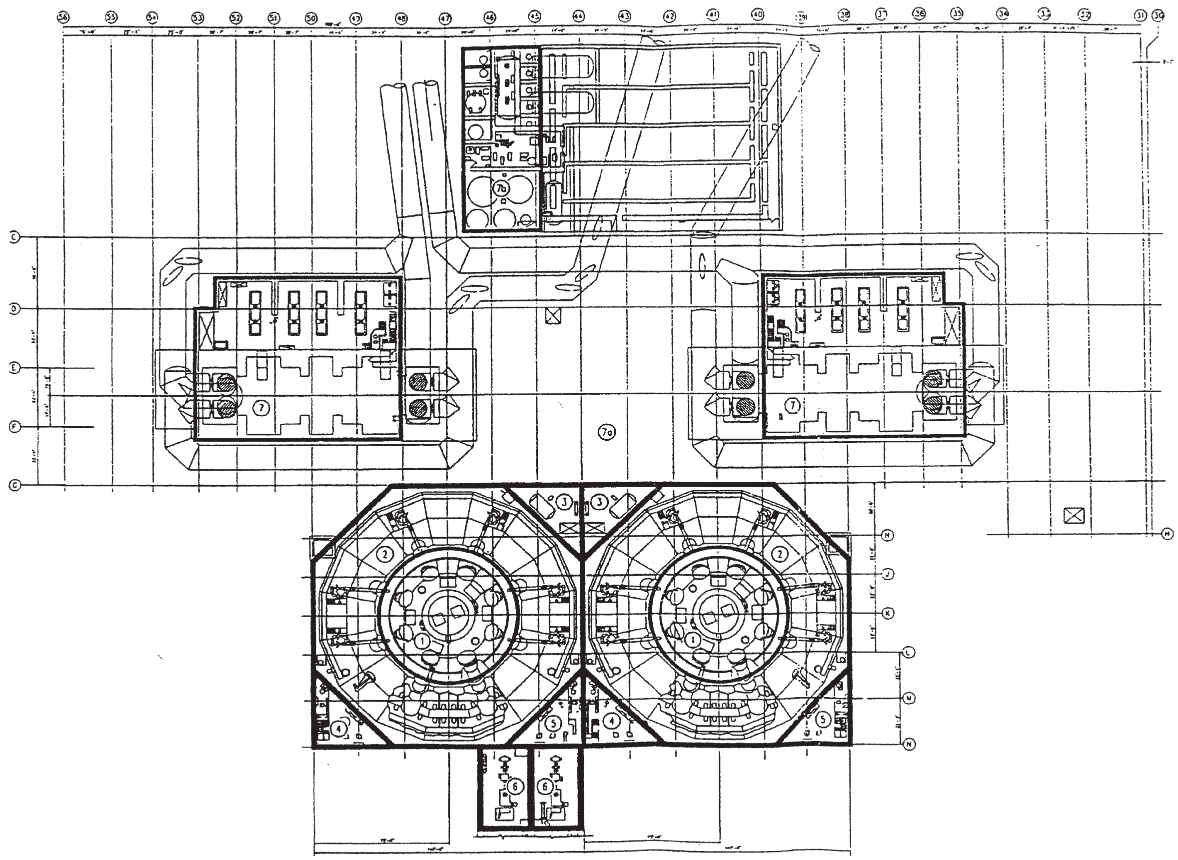
DRESDEN STATION
UNITS 2 & 3

RECIRCULATION LINE BREAK BLOWDOWN
CORE SHROUD RESULTANT FORES

FIGURE 3.9-11



UFSAR REVISION 2, JUNE 1997
DRESDEN STATION UNITS 2 & 3
RECIRCULATION LINE BREAK BLOWDOWN CORE SHROUD RESULTANT MOMENTS
FIGURE 3.9–12



BECHTEL
CHICAGO PROJECTS OFFICE

DRESDEN UNITS 2 AND 3
COMMONWEALTH EDISON CO.

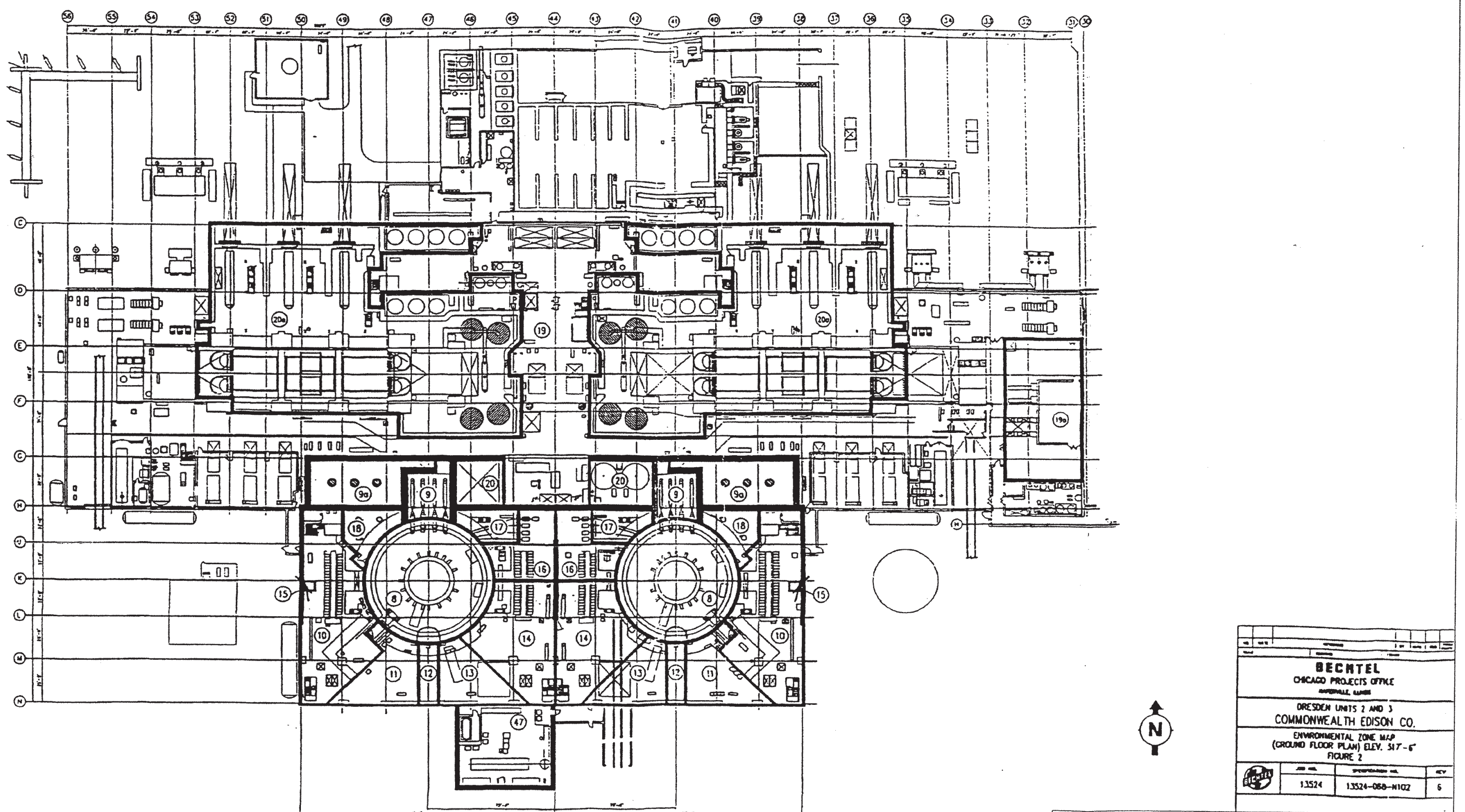
ENVIRONMENTAL ZONE MAP
(BASEMENT FLOOR PLAN) ELEV. 476'-6"
FIGURE 1

JOB NO.	SPECIFICATION NO.	REV
13524	13524-061-N102	6

DRESDEN STATION
UNITS 2 & 3

ENVIRONMENTAL ZONE MAP
BASEMENT FLOOR PLAN
ELEVATION 476'-6"

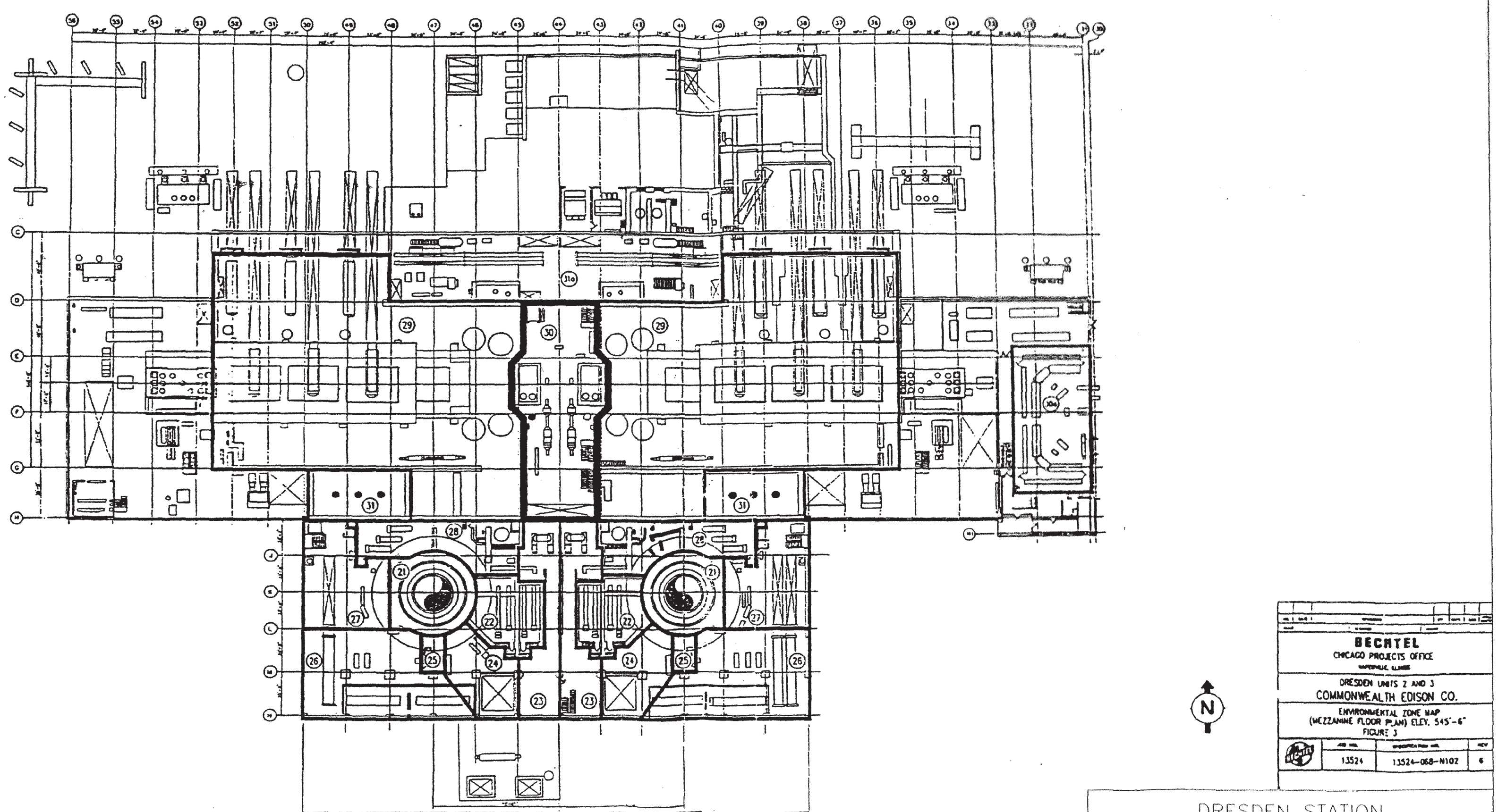
FIGURE 3.11-1



DRESDEN STATION
UNITS 2 & 3

ENVIRONMENTAL ZONE MAP
GROUND FLOOR PLAN
ELEVATION 517'-6"

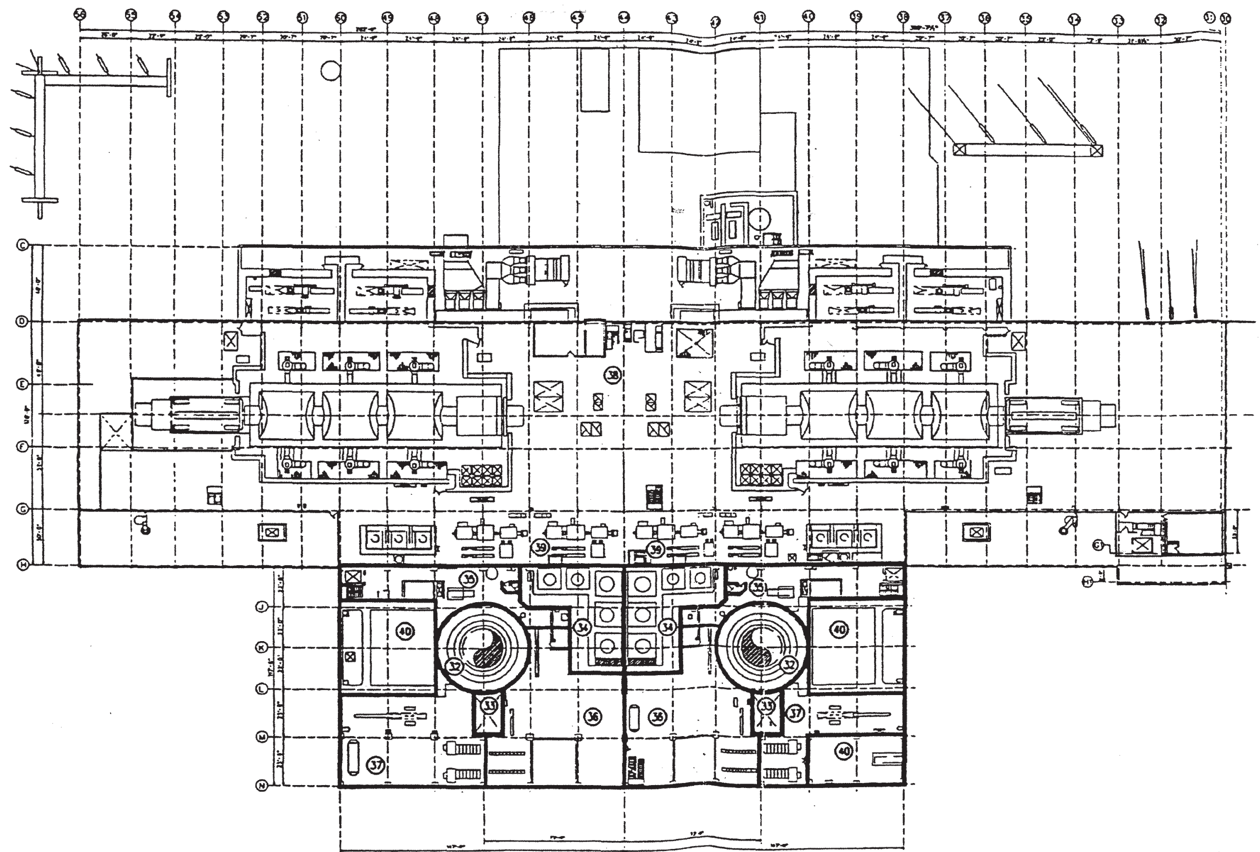
FIGURE 3.11-2



DRESDEN STATION
UNITS 2 & 3

ENVIRONMENTAL ZONE MAP
MEZZANINE FLOOR PLAN
ELEVATION 545'-6"

FIGURE 3.11-3



BECHTEL
CHICAGO PROJECTS OFFICE

DRESDEN UNITS 2 AND 3
COMMONWEALTH EDISON CO.

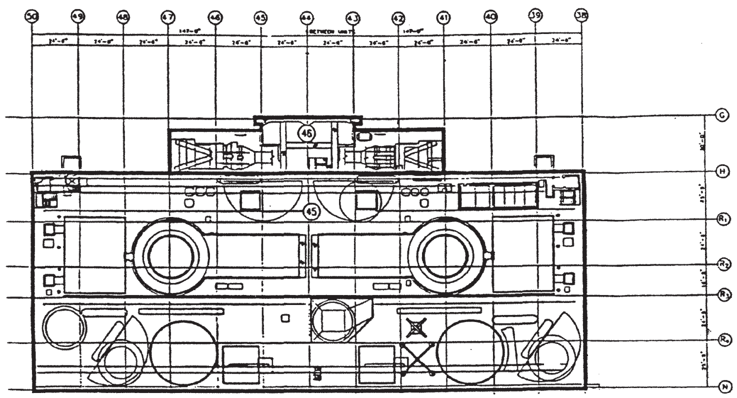
ENVIRONMENTAL ZONE MAP
MAIN FLOOR PLAN ELEV. 570'-0"
FIGURE 4

13524	13524-068-M182	6

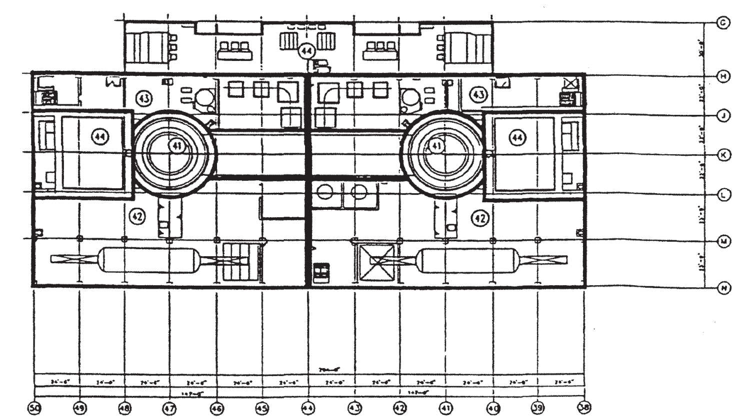
DRESDEN STATION
UNITS 2 & 3

ENVIRONMENTAL ZONE MAP
MAIN FLOOR PLAN
ELEVATION 570'-0"

FIGURE 3.11-4



FLOOR PLAN ELEVATION 613'-0"



FLOOR PLAN ELEVATION 589'-0"

BECHTEL
CHICAGO PROJECTS OFFICE

DRESDEN UNITS 2 AND 3
COMMONWEALTH EDISON CO.

ENVIRONMENTAL ZONE MAP
(REACTOR FLOOR PLAN) ELEV. 589'-0" & 613'-0"
FIGURE 5

13524	13524-068-N102	6

DRESDEN STATION
UNITS 2 & 3

ENVIRONMENTAL ZONE MAP
REACTOR FLOOR PLAN
ELEVATION 589'-0" & 613'-0"

FIGURE 3.11-5