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3.0 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS 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
B-305	Drywell Pipe Whip Protection Panel Locations & Details Unit 1
B-835	Drywell Pipe Whip Protection Panel Locations & Details Unit 2
M-4A	Environmental Zone Map & Design Requirements
M-5	General Arrangement Ground Floor Plan
M-6	General Arrangement Basement Floor Plan Units 1 and 2
M-8	General Arrangements Sections "A-A" & "B-B"
M-10	General Arrangements Sections "E-E" & "F-F"
M-464	Weight, Thermal, and Dynamic Isometric Main Steam (Inside Drywell)
M-465	Weight & Dynamic Data Iso. Core Spray Pump 1A & 1B Suction Unit 1
M-466	Weight, Thermal & Dynamic Iso. Core Spray Pump 1A & 1B Discharge Unit 1
M-467	Weight, Thermal & Dynamic Iso. Core Spray from Drywell to RX Unit 1
M-468	Weight, Thermal & Dynamic Iso. RHRS Pump Suction 1A, 1B, 1C & 1D Unit 1
M-469	Weight, Thermal & Dynamic Iso. RHRS Pump 1A & 1B Discharge Unit 1
M-470	Weight, Thermal & Dynamic Iso. RHRS Pump 1C & 1D Discharge Unit 1
M-471	Weight, Thermal & Dynamic Iso. RHR Service Water Pumps Suction Unit 1
M-472	Weight, Thermal & Dynamic Iso. RHRS Service Water Pump Discharge to Heat Exchanger 1A & 1B Unit 1
M-473	Weight, Thermal & Dynamic Iso. RHRS Heat Exchanger 1A & 1B Service Water Piping Unit 1
M-474	Weight, Thermal & Dynamic Iso. Reactor Building Cooling Water Piping Unit 1
M-475	Weight, Thermal & Dynamic Iso. HPCI Pump Suction & Discharge
M-476	Weight, Thermal & Dynamic Iso. HPCI Turbine Steam Supply & Exhaust Unit 1
M-477	Weight, Thermal & Dynamic Iso. Main Steam Piping Unit 1
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M-479	Weight, Thermal & Dynamic Iso. Pressure Suppression Piping, Sheet 2, Unit 1
M-480	Weight, Thermal & Dynamic Iso. Reactor Feed Piping Unit 1
M-483	Weight, Thermal & Dynamic Iso. Standby Gas Piping, Sheet 1, Unit 1
M-484	Weight, Thermal & Dynamic Iso. Standby Gas Piping, Sheet 2, Unit 1
M-486	Weight, Thermal & Dynamic Iso. Reactor Feed (Inside Drywell Unit 1
M-487	Weight, Thermal & Dynamic Iso. Core Spray Pump 2A & 2B Suction Unit 2
M-488	Weight, Thermal & Dynamic Iso. Core Spary Pump 2A & 2B Discharge
M-489	Weight, Thermal & Dynamic Iso. Core Spray from Drywell to RX Unit 2
M-490	Weight, Thermal & Dynamic Iso. RHRS Pump Suction 2A, 2B, 2C & 2D Unit 2
M-491	Weight, Thermal & Dynamic Iso. RHRS Pump 2A & 2B Discharge Unit 2

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3.0 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS
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3.0 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

This chapter describes the principal architectural and engineering design aspects of the Quad Cities Station. These include the structural and environmental design requirements imposed by consideration of natural phenomena and plant accidents. In addition, the chapter discusses, in Section 3.1, conformance of the plant design with general regulatory design criteria, and, in Section 3.2, systems for classifying plant structures, systems and components.

3.1 CONFORMANCE WITH NRC GENERAL DESIGN CRITERIA

The following section presents the proposed General Design Criteria (issued July 1967) which were used by the AEC as guidance in evaluating the original design of Quad Cities Station. Changes to the plant design are completed in accordance with the requirements of Exelon Generation Company 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 original plant design criteria. It has since been incorporated into this section of the UFSAR. [3.1.1]

This section of the report contains an evaluation of the design basis of the Quad Cities nuclear facility by means of the draft of the 70 proposed General Design Criteria for Nuclear Power Plant Construction Permits issued by the Atomic Energy Commission. [3.1.2]

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 Quad Cities 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 the Quad Cities station relative to each of the nine groups of proposed criteria. (The draft of criteria is 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 current 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 the Quad Cities station fully satisfies the intent of the criteria.

3.1.1 Group I - Overall Plant Requirements

The intent of the draft 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. [3.1.3]

It was concluded that the design of this plant is in conformance with the criteria of Group I 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 below for the individual criteria in this group.

3.1.1.1 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 used is required.

Response

The reactor facility's essential components and systems were 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 their compliance by those systems, components, and structures which are important to plant safety.

Systems or Components	Sections Which Identify Applicable Codes or Standards
Reactor coolant system	Applicable FSAR Sections: 3.4.3.1, 3.6.3, 3.6.3.2, 3.6.3.3, 3.6.3.5, 4.1, 4.2, 4.3.1, 4.4, and 12.1.3.6 Applicable UFSAR Sections: 3.2, 3.9.3, 3.9.5, 4.1, 4.2, 4.5, 5.2.1, 5.3.1, 5.3.3, 5.4.1, and 5.4.13
Containment system	Applicable FSAR Sections: 5.2.1, 5.2.2, 5.2.3.7, and 12.1.3.4 Applicable UFSAR Sections: 3.2, 3.8.2, and 6.2.1
Emergency core cooling systems	Applicable FSAR Sections: 6.2.3.2, 6.2.4.2, 6.2.5.2, and 12.1.3.3 Applicable UFSAR Sections: 3.2, 3.9.3, 3.9.6, and 6.3.2
Diesel Generator	Applicable FSAR Sections: 12.1.3.7 Applicable UFSAR Sections: 3.9.3
Radwaste	Applicable FSAR Sections: 1.2.6, 1.3.1, 1.3.10, 1.3.12, 1.5.4.9, 9.1, 9.2.1, 9.2.2, 9.3.1, 9.4.1, 9.5.1, and 12.3.1 Applicable UFSAR Sections: 1.2, 11.0, 11.2, 11.3, 11.4, and 12.3.2.1

3.1.1.2 Criterion 2 - Performance Standards

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, 2.4, 2.5, 6.2, and 11.5.3

Applicable UFSAR Chapter: 3

Additional Information

In 1977 the NRC initiated the Systematic Evaluation Program (SEP) to review the designs of older operating nuclear power plants (i.e., pre-GDC plants). Generic Letter 95-04[1] notified licensees of the final disposition of the SEP lessons-learned. The Generic Letter categorized tornado missiles as a "Category 3" issue – unresolved but covered by existing regulatory programs. As noted in Attachment 2 of the Generic Letter, the NRC determined the appropriate regulatory mechanism for resolving tornado missiles was the Individual Plant Examination of External Events (IPEEE) program. The NRC issued a Staff Evaluation Report for the Quad Cities IPEEE[2] on April 26, 2001 noting: "The staff concluded that the aspects of seismic events, fire, and high winds, floods and other (HFO) external events were adequately addressed." The report discusses Generic Safety Issue 156 (SEP) noting: "On the basis that no vulnerabilities associated with this issue were identified in the licensee's IPEEE submittal, the staff considers this issue resolved for Quad Cities."

3.1.1.3 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 throughout 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 10.6

Applicable UFSAR Sections: 6.2 and 9.5.1

3.1.1.4 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

Quad Cities Units 1 and 2 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 1 and 2. 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 1 and 2;

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

C. Operation or safe shutdown of either Unit 1 or 2 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 1 or 2 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, 6.2.3, 6.5, and 9.2.2

3.1.1.5 Criterion 5 - Records 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

Applicable UFSAR Chapter: 17

3.1.2 Group II - Protection by Multiple Fission Product Barriers

The intent of the draft 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 environs 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. [3.1.4]

It is concluded that design of this plant is in conformance with the Criteria of Group II 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.

3.1.2.1 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, 4.2, 4.3, 4.4, 4.6, 5.4.1, 5.4.13, 6.3, 7.6, 7.7, 15.2, 15.3, and 15.4

3.1.2.2 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.

In addition, Oscillating Power Range Monitors (OPRMs) have been installed to detect thermal-hydraulic instabilities that result in core power oscillations. If the instabilities grow to a point where the power oscillations could result in a condition exceeding the fuel safety limit, the OPRM automatically initiates a reactor scram via the reactor protection system.

Applicable FSAR Sections: 3.2, 3.3, 3.4, 3.5, 7.2, 7.3, 7.4, and 7.5

Applicable UFSAR Sections: 1.2, 4.2, 4.3, 4.4, 4.6, 7.6, and 7.7

3.1.2.3 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

Applicable FSAR Sections: 3.1 and 3.3

Applicable UFSAR Sections: 4.1 and 4.3.2

3.1.2.4 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, seismic loads as prescribed for the plant. Provisions are made to control or shutdown 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: 4.2 and 4.3

Applicable UFSAR Sections: 5.3 and 5.4

3.1.2.5 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 100.

Applicable FSAR Sections: 5.2, 5.3, and 6.2.4

Applicable UFSAR Sections: 4.2, 5.2, 6.2.1, 6.2.2, 6.2.3, 6.5, and 15.6

3.1.3 Group III - Nuclear and Radiation Controls

The intent of the draft 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. [3.1.5]

It is concluded that the design of this plant is in conformance with the criteria of Group III 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.

3.1.3.1 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, 3.3, 3.8.4, 6.4, 7.4, and 9.4

3.1.3.2 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.

Applicable FSAR Sections: 1.2.4, 1.3.6, 4.2, 7, 8, 11.2.3, and 11.3.3

Applicable UFSAR Sections: 1.2.1, 1.2.2, 4.3, 5.3, 7.2, 7.3, 7.4, 7.6, 7.7, 8.2, 8.3, 10.2, 10.3, 10.4, 11.5, and 12.3

3.1.3.3 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 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, 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: 1.2.1, 1.2.2, 7.4, 7.6, 7.7, and 9.3.5

3.1.3.4 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, 6.2, 6.5, 6.6, 7.3, 7.4, 7.5, 7.7, 7.8, and 7.9

Applicable UFSAR Sections: 1.2.1, 1.2.2, 4.6, 5.4, 6.3, 7.2, 7.3, 7.4, 7.6, 7.7, and 9.2

3.1.3.5 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 and 7.7

Applicable UFSAR Sections: 4.6, 5.4, 6, 6.3, 7.3, 9.2, and 9.3

3.1.3.6 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, 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 Sections: 5.2.5, 5.3, 7.3, and 7.6

3.1.3.7 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 to monitor significant parameters from specific plant process systems and specific areas, including the plant effluents to the site environs, 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, 6.2.3, 6.5, 11.2, 11.3, 11.5, and 12.3

3.1.3.8 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 to monitor significant parameters from specific plant process systems and specific areas, including the plant effluents to the site environs, and to provide alarms and signals for appropriate corrective actions.

Applicable FSAR Section: 7.6

Applicable UFSAR Section: 12.3

3.1.4 Group IV - Reliability and Testability of Protection Systems

The intent of the draft 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. [3.1.6]

It is concluded that the design of this plant is in conformance with the criteria of Group IV 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 individual criteria in this group.

3.1.4.1 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, 1.2.2, 7.2, 7.3, 7.4, 7.6, 7.7, 11.5, and 12.3

3.1.4.2 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 initiate 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, 1.2.2, 7.2, 7.3, 7.4, 7.6, 7.7, 11.5, and 12.3

3.1.4.3 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.

3.1.4.4 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 Sections: 7.2, 7.3, 7.6, and 7.7

3.1.4.5 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 systems 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, 1.2.2, 6.2, 7.2, 7.3, 7.6, 11.5, and 12.3

3.1.4.6 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 systems performance.

Applicable FSAR Sections: 1.2.5, 1.3.9, and 8

Applicable UFSAR Sections: 1.2.1, 1.2.2, and 8

3.1.4.7 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, 1.2.2, 7.2, 7.3, 7.6, 11.5, and 12.3

3.1.4.8 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 1 of the 2 channels, a situation insufficient to produce a reactor scram but readily available to perform its protective function upon another trip (either by failure or by exceeding the preset trip).

Applicable FSAR Sections: 1.2.4, 1.3.6, 7.4, 7.5, 7.6, and 7.7

Applicable UFSAR Sections: 1.2.1, 1.2.2, 7.2, 7.3, 7.6, 11.5, and 12.3

3.1.5 Group V - Reactivity Control

The intent of the draft 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. [3.1.7]

It is concluded that the design of this plant is in conformance with the Criteria of Group V 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 individual criteria in this group.

3.1.5.1 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 principle reactivity control systems. Control of reactivity is operationally provided by a combination of movable control rods, burnable neutron absorbers contained in the fuel, 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, 4.3, 4.6, and 9.3.5

3.1.5.2 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 damage limits.

Response

The plant design contains two independent and different principle reactivity control systems. Control of reactivity is operationally provided by a combination of movable control rods, burnable neutron absorbers contained in the fuel, 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 shut down system to cover emergencies in the operational reactivity control system discussed above. This standby system is designed to shut down the reactor in about two hours.

Applicable FSAR Sections: 1.2.1, 3.5.2, 3.5.3, and 6.7

Applicable UFSAR Sections: 1.2, 4.3, 4.6, and 9.3.5

3.1.5.3 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, 4.3, 4.6, 7.7, and 15.4

3.1.5.4 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, 4.3, 4.6, and 9.3.5

3.1.5.5 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 would not be exceeded for any reactivity transient resulting from a single equipment malfunction or operator error.

Applicable FSAR Sections: 3.5, 4.3.3, 6.7, and 7.3

Applicable UFSAR Sections: 4.6, 5.4.1, 7.7, 9.3.5, and 15.4

3.1.5.6 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 or (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 would 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: 1.2, 4.3, and 4.6

3.1.6 Group VI - Reactor Coolant Pressure Boundary

The intent of the draft 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. [3.1.8]

It is concluded that the design of this plant is in conformance with the Criteria of Group VI 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 individual criteria in this group.

3.1.6.1 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 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 Sections: 4.3.2, 4.6, and 15.4

3.1.6.2 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 loading, (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 USASI 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: 3.2, 5.2, 5.3, and 5.4

3.1.6.3 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 or 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 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 first draft of this criterion.

Applicable FSAR Sections: 4.2 and 4.3

Applicable UFSAR Sections: 5.2, 5.3, and 5.4

3.1.6.4 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 1560 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, 5.3, and 5.4

3.1.7 Group VII - Engineered Safety Features

The intent of the draft of the proposed criteria for this group is: [3.1.9]

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

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

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

D. To justify that each ESFs 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 VII 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 individual criteria in this group.

3.1.7.1 Criterion 37 - Engineered Safety Features (ESF) 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 and including the circumferential rupture of any pipe in that boundary assuming an unobstructed discharge) and variables exceed their operating limits, an extensive system of engineered safety features (ESF) limits the transient and the effects to levels well below those which are of public safety concern. These ESFs include 1) 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.); 2) those systems which offer additional protection against a reactivity excursion (reactor standby liquid control system, control rod velocity limiters, and control rod housing supports); 3) those systems which act to reduce the consequences of design basis accidents (main steam line flow restrictors, primary containment atmospheric control system); and 4) those systems which provide core standby and containment cooling in the event of a loss of normal cooling (core spray cooling system, core residual heat removal system (RHRS), core high pressure coolant injection (HPCI) system, automatic depressurization system, and the standby coolant supply system).

Applicable FSAR Sections: 1.2, 1.3, 5, 6, and 8

Applicable UFSAR Sections: 1.2, 4.6, 5.4.4, 6.0, 6.2, 6.3, 6.5, 7.3, 8.3, 9.2.8, 9.3.5, and 13.3

3.1.7.2 Criterion 38 - Reliability and Testing of ESF

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 the 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 engineered safety features 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, 5, 6, and 8

Applicable UFSAR Sections: 1.2, 4.6, 5.4.4, 6.0, 6.2, 6.3, 6.5, 7.3, 8.3, 9.2.8, and 9.3.5

3.1.7.3 Criterion 39 - Emergency Power for ESF

Alternate power systems shall be provided and designed with adequate independency, 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 engineered safety features functions under all postulated design basis accident conditions.

Applicable FSAR Sections: 1.2.5, 1.3.5, 1.3.9, 1.4, and 1.5.3

Applicable UFSAR Sections: 1.2.1.5, 1.2.2.5, 1.2.2.9, Table 1.2-3, 1.2.4.3, and 8.3

3.1.7.4 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 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: 5.2.3.8, 6.2.3.2, and 6.2.5.2a

Applicable UFSAR Sections: 3.5 and 6.3

3.1.7.5 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 functions. As a minimum, each engineered safety feature shall provide this required safety function assuming a failure of a single active component.

Response

The emergency core cooling systems (ECCS) are 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, 6, and 8

Applicable UFSAR Sections: 6.0, 6.2, 6.3, 6.5, and 8

3.1.7.6 Criterion 42 - ESF Component

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 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: 1.2.1d, 5.2.3.8, and 6

Applicable UFSAR Sections: 1.2.1, 6, and 6.2

3.1.7.7 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, 6.2.4.2, 6.2.5.2, 6.2.5.3, 6.2.5.4, and 6.3

Applicable UFSAR Sections: 1.2.1, 5.4, 6.0, 6.2, 6.3.2, 6.3.3, 6.3.4, and 9.2

3.1.7.8 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 emergency core cooling systems (ECCS) are 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, and 6.2.5.1

Applicable UFSAR Sections: 5.4, 6.1, 6.3, and 7.3

3.1.7.9 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 system, including reactor vessel internals and water injection nozzles.

Response

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

Applicable FSAR Sections: 6.2.3.4, 6.2.4.4, 6.2.5.1, and 6.2.5.4

Applicable UFSAR Sections: 4.6, 5.4, and 6.3

3.1.7.10 Criterion 46 - Testing of ECCS Components

Design provision 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: 6.2.3.4, 6.2.4.4, and 6.2.5.4

Applicable UFSAR Sections: 5.4 and 6.3

3.1.7.11 Criterion 47 - Testing of ECCS

A capability shall be provided to test periodically the delivery capability of the emergency core cooling system 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: 6.2.3.4, 6.2.4.4, and 6.2.5.4

Applicable UFSAR Section: 6.3

3.1.7.12 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: 6.2.3.4, 6.2.4.4, 6.2.5.4, and 8.3.1

Applicable UFSAR Section: 4.6, 5.4, 6.3.4, and 8.3.1

3.1.7.13 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 allowance for effects from metal-water or other chemical reactions beyond conditions that would occur with normal operation of emergency core cooling systems (ECCS). [3.1.10]

Applicable FSAR Sections: 1.2.3 and 5.2.3.4

Applicable UFSAR Sections: 1.2.1, 6.2.1, and 6.2.5

3.1.7.14 Criterion 50 - NDT Requirements 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 nil ductility transition (NDT) temperature. [3.1.11]

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 would be below the values stated in 10 CFR 100.

Applicable FSAR Section: 5.2.3.1

Applicable UFSAR Section: 3.8.2 and 6.2.1

3.1.7.15 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 would be below the values stated in 10 CFR 100.

Applicable FSAR Section: 1.2.3e, 5.2.2, 5.3, and 6.4

Applicable UFSAR Sections: 1.2.1, 6.2, and 5.4

3.1.7.16 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: 1.2.3b, 5.2.1, 5.2.2, 5.3.2, and 6.2.4.2

Applicable UFSAR Sections: 1.2.1, 6.2, and 6.3

3.1.7.17 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 7.7.2

Applicable UFSAR Sections: 6.2.1, 6.2.2, 6.2.3, and 7.3.2

3.1.7.18 Criterion 54 - Containment Leakage Rate Testing

Containment shall be designed so that 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 would be below the values stated in 10 CFR 100. 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: 1.2.3d, 5.2.4, and 5.3.4

Applicable UFSAR Sections: 1.2.1, 6.2.3, and 6.2.6

3.1.7.19 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 pressures which permit extrapolation of results to the design accident pressure conditions, using relationships established initially for comparative leakage at these two conditions.

Applicable FSAR Sections: 1.2.3d, 5.2.4.1, and 5.3.4

Applicable UFSAR Sections: 1.2 and 6.2.6

3.1.7.20 Criterion 56 - Provision for Testing of Penetrations

Provisions shall be made for testing penetrations which have resilient seals or expansion bellows to permit leaktightness 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

3.1.7.21 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

3.1.7.22 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 Section: 5.2.4, 6.2.4.4, and 6.2.4.1

Applicable UFSAR Sections: 6.2.1 and 6.2.2

3.1.7.23 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

3.1.7.24 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

3.1.7.25 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 Section: 5.2

Applicable UFSAR Section: 6.2.2 and 8.3

3.1.7.26 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 10.10

Applicable UFSAR Sections: 6.2.3, 6.5, and 9.4

3.1.7.27 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 10.10

Applicable UFSAR Sections: 6.2.3, 6.5, and 9.4

3.1.7.28 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 10.10

Applicable UFSAR Sections: 6.2.3 and 6.5

3.1.7.29 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, 7.6.2.4, and 10.10

Applicable UFSAR Sections: 6.2.3, 6.5, 8.3, and 11.5

3.1.8 Group VIII - Fuel and Waste Storage Systems

The intent of the draft 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. [3.1.12]

It is concluded that the design of this plant is in conformance with criteria of Group VIII 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 individual criteria in this group.

3.1.8.1 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 the Dry Cask Storage (DCS) system described in Section 9.1.2.4, 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, 1.2.2, and 9.1

3.1.8.2 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 the DCS system described in Section 9.1.2.4, 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 and 9.1

3.1.8.3 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 the DCS system described in Section 9.1.2.4, 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 Section: 9.1 and 12.3

3.1.8.4 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 the DCS system described in Section 9.1.2.4, 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 in designed and arranged to insure subcriticality in the storage pool.

Applicable FSAR Sections: 1.2.3, 1.3.4.2, 5.3, 9, and 10.1

Applicable UFSAR Sections: 1.2.1, 1.2.2, 6.2.3, 9.1, 11.2, 11.3, 11.4, 12.3, and 12.5

3.1.9 Group IX - Plant Effluents

The intent of the draft of the proposed criterion for this group is to establish the plant effluent release limits and to identify the means of controlling the releases within these guide limits. [3.1.13]

It is concluded that the design of this plant is in conformance with the criterion of Group IX 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 in this group.

3.1.9.1 Criterion 70 - Control of Releases 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 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 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. 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.

Refer to Section 9.1.2.4 for a description of spent fuel storage and handling using the DCS system and the Independent Spent Fuel Storage Installation (ISFSI).

Applicable FSAR Sections: 1.2.6, 1.3.11, 1.5.1, 1.5.2, 5.2.1, 5.2.2, 5.3, 7.6, 9, and 10.10

Applicable UFSAR Sections: 1.2.1, 1.2.2, 1.2.4, 6.2.1, 6.2.3, 6.5, 9.1, 9.4, 11.2, 11.3, 11.4, 11.5.4, 12.3, and 12.5

3.1.10 References

1. Generic Letter 95-04, Final Disposition of the Systematic Evaluation Program Lessons-Learned Issues, April 28, 1995.

2. Quad Cities Nuclear Power Plant – Review of Individual Plant Examination of External Events (IPEEE) Submittal, April 26, 2001.

3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

The plant structures and equipment are divided into two categories as related to safety. The categories are defined as: [3.2.1]

- Class I - Those structures and equipment of which a failure thereof could cause significant release of radioactivity (i.e. calculated off-site doses in excess of 10 CFR 100 or 10 CFR 50.67 as applicable) or are vital to a safe plant shutdown.

- 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. Unless specified otherwise in the FSAR, other structures and equipment not listed as Class I are Class II. The Class I systems and structures are as follows:

3.2.1 Class I - Structures

A. Reactor Building

B. Primary Containment Vessel

C. Control Room (including the auxiliary electrical equipment room and the cable spreading room)

D. 310-foot Concrete Chimney

E. Floor Drain Surge Tank Structure [3.2.2]

F. New Fuel Storage Vault [3.2.3]

3.2.2 Class I - Systems (mechanical)

A. Core Spray [3.2.4]

B. High Pressure Coolant Injection (HPCI)

C. Standby Liquid Control (SBLC)

D. Residual Heat Removal (RHR) including RHR Service Water piping. [3.2.5]

E. Standby Gas Treatment (SBGT) [3.2.6]

F. Control Room Emergency Ventilation [3.2.7]

G. Diesel-Generator Cooling Water and Fuel Oil Supply [3.2.8]

3.2.3 Class I - Nuclear Steam Supply Equipment

A. Reactor Vessel [3.2.9]

B. Reactor Vessel Supports

C. Control Rod Drive System components necessary for scram

D. Control Rod Drive Housing Supports

E. Primary Reactor Internals including:

1. Fuel Assemblies

2. Control Rods

3. Core Shroud

4. Core Support

5. Steam Separators (*)

6. Steam Dryer (*)

(*) These components are Class 1 Seismic but not safety-related.

F. Reactor Recirculating Water Subsystem

3.2.4 Class I - Systems (electrical)

A. Standby Diesel Generators (AC emergency power)

B. Station Batteries (DC power)

C. Essential buses and other electrical gear essential for the operation of Class I systems and equipment as listed herein.

D. Automatic Depressurization System [3.2.10]

3.2.5 Class I - Instrumentation and Controls

A. Reactor Protection System

B. Primary Containment Isolation System

C. Neutron Monitoring System

D. ECCS System Instrumentation

E. Control Rod Drive Instrumentation

F. Suppression Pool Temperature Monitoring System [3.2.11]

G. Containment Air Monitoring System [3.2.12]

H. Alternate Rod Insertion System [3.2.13]

3.2.6 Class I - Miscellaneous Category

A. Spent Fuel Storage facilities [3.2.14]

B. All piping systems connected to the RPV from the vessel, up to and including the outer isolation valve external to the drywell.

C. That portion of all other lines which penetrate or are attached to the primary containment, i.e., from the point of origin out to and including the outer isolation valve external to the primary containment.

There are a few cases where Class I items are located within Class II structures. These are:

A. Standby Diesel Generators 1 and 2 in Turbine Building	Drawing M-5
B. Diesel Generator Cooling Water Pumps in Turbine Building	Drawing M-6 and M-10
C. 4kV ECCS Switchgear in Turbine Building	Drawing M-5
D. RHR Service Water Pumps in Turbine Building	Drawing M-8
E. Batteries & Associated Equipment in Turbine Building	Drawing M-10
F. Control Room Emergency Ventilation in Turbine Building [3.2.15]	Drawing M-725
G. HPCI system in Turbine Building	Drawing M-6
H. Cables in Cable Tunnel in Turbine Building	Drawing M-5
I. Unit 1/2 Diesel in Diesel Building	Drawing M-5

Structural design of the buildings and the capabilities to withstand seismic events are discussed in greater detail in Section 3.8. However, to summarize, the location of the previously listed equipment has been investigated to assure that such areas of Class II structures will afford Class I protection to these components. Hence, although some Class I components are located within a Class II Building, the location of such components has been selected to assure that an equal degree of safety against structural failure is afforded as to Class I equipment located within the Class I reactor building. This conclusion can be seen from viewing the referenced figures for each component, noting the elevations and protective concrete structural walls and reviewing the methods of analysis.

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 assure: [3.2.16]

1. The integrity of the reactor coolant pressure boundary,

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

3. 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 10 CFR 50.67 as applicable).

Subsequent to Generic Letter 83-28, a reclassification of certain mechanical and electrical systems and components was undertaken 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 conform with the definition for safety-related equipment contained in the footnote of item 2.2 of 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 non safety-related. Detailed application of safety-related classification is identified in the station's work control system data base. The station's work control system data base complies with Generic Letter 83-28 for safety-related equipment classification identification.

The MEL includes the Requirements Summary Matrices (RSMs), which delineate the approach to system and components where a graded QA program was applied.

3.2.8 Industry Code Applicability to Reactor Coolant Pressure Boundary Components

Codes and standards applied to individual systems and components are contained in their specific UFSAR sections. [3.2.17]

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

A. USAS (ASA) B31.1, 1955 Edition, Code Cases N-7, N-9, and N-10
   1. USAS (ASA) B31.1.0, MSS-SP66
   2. USAS (ASA) B31.1.0, MSS-SP61

B. ASME Boiler & Pressure Vessel Code, Section 1, Summer 1965 Addenda

C. ASME Boiler & Pressure Vessel Code, Section III, Summer 1965 Edition, Summer 1965 Addenda

1. Paragraph N323
2. Paragraph N624.2 — N624.7
3. Paragraph N322
4. Paragraph N624
5. Paragraph N627
6. Paragraph N325

D. ASME Boiler & Pressure Vessel Code, Section VIII, 1965 Edition.

3.2.8.2 Reactor Recirculation Pumps

A. USAS (ASA) B31.1, 1955 Edition, Code Case N-7, N-9, and N-10

B. ASME Boiler & Pressure Vessel Code, Section III, Class C, 1965 Edition, Winter 1965 Addenda
   1. Paragraph N323.1
   2. Paragraph N624.2 - N624.7
   3. Paragraph N322
   4. Paragraph N624
   5. Paragraph N627
   6. Paragraph N325

C. ASME Boiler & Pressure Vessel Code, Section VIII, 1965 Edition

3.2.8.3 Main Steam Isolation, Safety and Relief Valves

A. USAS (ASA) B31.1, 1955 Edition

B. USAS (ASA) B31.1, 1967 Edition

C. USAS (ASA) B31, Code Cases N-2, N-7, N9, and N-10

D. ASME III (Target Rock SRV)

E. ASME III, 1980 Edition, Winter 1980 Addenda without Code Stamp (ERVs)

F. ASME III, 1995 Edition with 1996 Addenda without Code Stamp (ERVs)

3.2.8.4 Reactor Pressure Vessel

A. ASME Boiler & Pressure Vessel Code, Section III, 1965 Edition, Summer 1965 Addenda.
   1. Paragraph N-152.b; Summer 1967 Addenda
   2. Figure N-414; Winter 1966 Addenda
   3. Figure N-462.4.d; Summer 1967 Addenda
   4. Table N-525; Winter 1966 Addenda
   5. Paragraph N-626.5; Winter 1966 Addenda
   6. Paragraph N-627.7; Winter 1966 Addenda

B. ASME Code Cases
   1. 1332-1, Paragraph 5
   2. 1332-2
   3. 1335, Paragraph 4
   4. 1355.2, Paragraph 4
   5. 1336, Paragraph 1
   6. 1355
   7. 1441

3.2.8.5 Piping System

A. USAS (ASA) B31.1, 1967 Edition [3.2.18]

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. Where appropriate, the applicable UFSAR section which discusses this code or standard is listed.

SYSTEM	CODE
ACAD	ASME III, 1974 through summer 1976 Addendum; IEEE 279; 323-1974; 344-1975, 384 [3.2.19]
Cable new installations	IEEE 383-1974 (see Section 9.5.1)
CAM	IEEE 279, 323-1974, 344-1975, 384 [3.2.20]
CAM Pressure Retaining Components	ASME III, Division I, NA-4000, NC-2000, NC-4000, NE-2000, NE-5000, and NE-4000, 1974 through Summer 1976 Addendum (see Section 6.2.5.2)
Condensate Pit Level Alarms	IEEE 279 (see Section 3.4)
Containment	ASME III, 1965 through Winter 1965 Addendum, Class B (see Section 6.2.1)
Containment Penetrations	ASME III, Class B (see Section 3.8.2.1) [3.2.21]
Containment Penetration fitting design	ASME VIII (see Section 3.8.2.1.7) [3.2.22]
Control Rod Drive	ASME III [3.2.23]
Core Spray Piping	USAS B31.1 (see Table 6.3-4) [3.2.24]
Core Spray Pumps	ASME III, Class C (see Table 6.3-4) [3.2.25]
Core Spray Spargers & Nozzles	ASME III, 1965 (see Section 6.3.2.1.2)
Core Spray Vessel Nozzle	ASME SA 336, Code Case 1332 (see Section 6.3.2.1.2)
Drywell-Suppression Chamber Vacuum Bkr	IEEE-279 (see Section 6.2)
Electrical Distribution UV Relays	IEEE 279-1971, 323-1971, 384-1974, (see Section 8.3) [3.2.26]

Feedwater Piping	ASA B31.1 (see Section 10.3)
Fuel Pool Cooling/Cleanup Filters/Demin	ASME VIII (see Section 9.1.3)
HPCI Piping	USAS B31.1 & ASME Section 1 (see Section 6.3.2.3.2)
HPCI Pumps	ASME VIII (see Section 6.3.2.3.2)
Main Steam Piping	USAS B31.1 & ASME I (see Section 10.3)
Main Steam and Feedwater Welding	ASME IX (see Section 10.3)
Off-Gas Piping	ASA B31.1 (see Section 10.4.2.1)
Off-Gas Recombiner/Adsorber	ASME III, Subsection ND, Class 3 & ANSI B31.1-1971 (see Section 10.4.2.1)
Reactor Protection System	IEEE 279-1966 (see Section 7.2.1)
Refueling Bridge Rails	ASME II, ASME IX (see Section 9.1.4)
RHR Heat Exchanger Shell Side	ASME III, Class C Winter 1966 Addenda, N2113 (see Section 6.3 and Section 9.2.1)
RHR Heat Exchanger Tube Side [3.2.27]	ASME VIII (see Section 6.3 and Section 9.2.1)
Main RHR Pumps	ASME III, Class C; Hydraulic Institute Standards (see Table 5.4-5)
RWCU Filter Demineralizer Vessels & Pumps	ASME III, Class C (see Section 5.4.8)
RWCU Regenerative Heat Exchangers	ASME VIII (see Section 5.4.8)
Spent Fuel Racks	ASME III, 1980, Subsection NF; ASME Appendix XVII (see Section 9.1.2)
SPTMS	IEEE 279-1971, 323-1974, 344-1971 or 1975 (see Section 6.2)
SRVDL Vacuum Breakers	ASME III, Subsection NC, 1977 through Summer 1977 Addenda, Class 2 (see Section 6.2)
TIP Guide Tubes	ASME VIII (see Section 6.2)

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. [3.3-1]

Two major considerations result from wind and tornado loadings: first, the capability of the structure to withstand pressure forces generated (Section 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. [3.3-2]

3.3.1 Wind Loadings

3.3.1.1 Design Wind Velocity

As a minimum, all structures except those walls surrounding the control room HVAC system (Section 3.3.1.1.3) are designed to withstand a 110-mph wind load which is in excess of the Uniform Building Code requirements. The applicability of the 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. [3.3-3]

3.3.1.1.1 Reactor Building

This structure encloses the reactor, the primary containment, and most of the equipment associated with the safe shutdown 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. [3.3-4]

The entire structure encompasses what is known as the secondary containment, discussed in Section 6.2. The reinforced concrete portion of the building extends from the foundation, elevation 548 feet, to the reactor refueling floor at elevation 690.5 feet. The structural steel superstructure extends from elevation 690.5 feet to the roof level, elevation 737 feet. These two portions of the building have different wind resistance capacities.

The superstructure is capable of withstanding a load generated by a 300-mph wind. The resulting stresses in the steel frame at this wind load are equal to the yield stress. However, such stresses are not present in all the structural members such as those supporting the reactor building crane.

The resistance of the lower reinforced concrete portion and the steel siding are discussed in Sections 3.3.2.2.1 and 3.3.2.2.2, respectively.

3.3.1.1.2 310-Foot Concrete Chimney

The concrete chimney is designed for a wind velocity of 110 mph but is capable of withstanding substantially higher wind speeds. A wind speed of 217 mph is estimated to cause the chimney to fail at a point 60 feet above grade. At this wind speed, the reinforcing steel could theoretically rupture in tension and the chimney would then rotate about a point on the outer shell (on the opposite side from the failed steel), then fall to the ground. Since the wind direction is variable, it could fall toward any point on the compass. If the chimney fell toward the reactor building, about 3 feet of the chimney's top could conceivably land on the western-most edge of the Unit 2 reactor shield plug. The estimated impact force is far below that required to penetrate the plug. Should the concrete chimney break up with parts falling into the fuel pool, damage could occur to the fuel pool. However, such an occurrence would inflict less damage than if the fuel cask were to fall into the fuel pool. The consequences of dropping the fuel cask into the fuel pool are described in Section 15.7. [3.3-5]

The only Class I equipment potentially affected by a postulated failure of the chimney is the emergency electrical switch gear located in the turbine building adjacent to the reactor building. The 4 kV buses, 13-1, 14-1, 24-1, and 23-1, could be damaged if a part of the chimney penetrated the turbine building roof. These components are separated by approximately 25 to 40 feet and consequently, only one of the buses could be affected. Since the buses are redundant, emergency power would still be available from the unaffected bus. The 480-volt switchgear is set in a cell in the turbine building and is not vulnerable to vertically falling missiles such as those that may be originated by the falling chimney. [3.3-6]

The Unit 2 battery room is located in the northeast corner of the mezzanine floor of the turbine building. Although the room is protected from postulated chimney failure by the turbine building superstructure, a piece could penetrate the roof and strike the batteries. The redundancy of the Unit 2 battery to the Unit 1 battery compensates for this event.

3.3.1.1.3 Control Room HVAC System

Walls 182, 183, and 184 surrounding the control room Train A HVAC system can withstand a wind loading of 79 mph. Based on the ANSI 58.1 Code and a 50-year recurrence interval, the design wind load for these walls is 75 mph. The wind resistance capacity of these walls is not believed to be a safety concern because the walls have been designed to the basic building codes. [3.3-7]

Control room Train B HVAC system (the safety-related train) does not have exterior walls exposed to wind loadings.

3.3.1.1.4 Other Structures

The other station structures such as the turbine building and control room, the radwaste building, and the crib house are similar in design to the reactor building with respect to wind load resistance. [3.3-8]

The turbine building superstructure is similar to the reactor building superstructure, consisting of a structural steel frame-type design. All Class I components in the turbine building are located in levels below the superstructure, i.e., within a reinforced concrete structure with capabilities similar to the reactor building, with the exception of the Unit 2 alternate 125 Vdc battery (see Section 8.3). Any large equipment, such as the turbine building cranes located in the superstructure, is designed and supported to preclude failure that could damage any equipment related to the ECCS systems or cause any significant release of radioactivity.

The crib house, a low reinforced concrete structure located below grade, is highly resistant to wind loading or tornado-generated missile damage.

3.3.2 Tornado Loadings

3.3.2.1 Applicable Design Parameters

The following are the tornado design parameters: [3.3-9]

A. A maximum tangential velocity of 300 mph,

B. A translational velocity of 60 mph, and

C. A pressure drop of 3 lb/in^2 at the vortex within 3 seconds.

Figure 3.3-1 shows the relationship between the velocity component (tangential plus translational) and the pressure drop as a function of the distance from the center of the tornado.

Section 3.3.2.2.1 describes the resistance of plant structures to tornado generated surface pressure loadings and Section 3.3.2.2.2 describes the resistance of the structures to pressure drop effects. The effects of missiles generated by a tornado are discussed in Section 3.5.4.

3.3.2.2 Determination of Forces on Structures

3.3.2.2.1 Surface Pressure Effects

In the investigation of the effects of tornados on structures that house Class I equipment, a model of the tornado considering the relationship between the 300 mph tangential velocity, 60 mph transverse velocity, and the 3 lb/in^2 pressure drop was used. Figure 3.3-2 shows the resulting surface pressures (windward and leeward) for the combined effect of the three tornado parameters listed in Section 3.3.2.1. The exterior concrete walls of the reactor building, turbine building, and control room were investigated for these pressures and were found to be stressed within the ultimate strength allowables of the ACI code. Details of the resistance of these structures to tornado pressure loadings are described as follows. [3.3-10]

The lower portion of the reactor building is capable of withstanding tornado wind velocities of up to 500 mph without exceeding the 4000 lb/in^2 concrete stresses. The 500 mph-wind is equivalent to a 1000 lb/ft^2 loading. Before structural wall failure could occur, the materials have to reach at least a yield stress condition. Winds with velocities reaching 860 mph (or a load of 2980 lb/ft^2) would have to be present to create such a condition.
[3.3-11]

The control room is a self-contained Class I concrete structure which is encompassed by the service building at the south of the turbine building. It is a heavy-walled reinforced concrete structure with roof level at elevation 639 feet, the same as the turbine building operating floor level. The north wall is protected by the entire turbine building structure while the east, south and west walls are protected by the service building. The control room, due to its height, construction, and arrangement, provides tornado resistance similar to that of the reactor building. [3.3-12]

The radwaste building is a low, heavy-walled reinforced concrete structure with even greater tornado resistance than the reactor building because the shielding requirements dictate the design rather than structural live or dead loads.

3.3.2.2.2 Pressure Drop Effects

Comparison of the reactor building structure's capability to withstand the limited tornado data on pressure gradients, clearly demonstrates the safety factors inherent in the design. The lowest recorded air pressure during a tornado that was determined in the investigation of this subject occurred at St. Louis, Missouri, on May 27, 1966. The reading was 26.94 in.-Hg, which was 2.42 in.-Hg lower than recorded at a weather bureau office seven blocks away. [3.3-13]

The 2.42-in.-Hg pressure differential is equivalent to a loading of 170 lb/ft^2, considerably less than the capabilities discussed in Section 3.3.2.2.1. In addition, the 170 lb/ft^2 is only generated when an instantaneous pressure drop occurs. Since a tornado travels at a finite rate 40 mph reasonably, the maximum pressure drop occurs over a finite time, in this case about 1/2 second. Hence, the assumption of a probable maximum loading of 170 lb/ft^2 is a conservative approach.

Other special circumstances related to tornado wind hazards have also been considered. For example, depressurization rate could be of concern if not accounted for properly. As noted previously, for instantaneous pressure differential loadings, the reactor building concrete walls are capable of resisting much higher loadings than have ever been experienced. The reactor building superstructure has also been designed for this effect and has blow-off panels installed to alleviate any potential problems. The siding of the reactor building, which consists of 20-foot wide sections extending the full height of the superstructure, is designed to remain intact up to a wind velocity of 170 mph or 75 lb/ft^2 except for special sections. Those special sections are designed to blow off at a pressure of 70 lb/ft^2. Tests have been conducted to verify the performance of these blow-off panels. G.W. Reynolds' report "Venting and Building Practices as Practical Means of Reducing Damage from Tornado Low Pressures" [1], states that a vent area of 1 square foot per 1000 ft^3 of volume should reduce pressure differentials to a safe level. The blow-off sections result in nearly 5 times this required vent area.

This load relief capability through venting of the superstructure in no way impairs the safe shutdown of the reactor as the necessary equipment is all located in the concrete portion of the building.

Walls and slabs of reactor building compartments housing Class I equipment exposed to the interior of the superstructure have also been checked for their capacity to withstand a 3 lb/in^2 pressure drop. All areas checked have been found to be within ultimate strength allowable limits except the Unit 1 and 2 Battery Rooms and Diesel Generator Rooms. Under a 3 lb/in^2 pressure drop, the doors to these rooms would be subjected to a load of approximately 9000 pounds, which they are not capable of resisting. The doors would open outward away from critical equipment; thus, relieving the pressure in these rooms. [3.3-14]

Venting provided by the louvres installed in the rooms permits equalization of pressure. Therefore, any induced pressure differentials can be effectively alleviated. Thus, the diesel generator and battery rooms are not affected by the tornado. [3.3-15]

3.3.3 References

1. Reynolds, G.W., "Venting and Building Practices as Practical Means of Reducing Damage from Tornado Low Pressures," American Meteorological Society Bulletin No. 1, Volume 3g, January 1958.

3.4 WATER LEVEL (FLOOD) DESIGN

This section describes the Quad Cities flood protection features and major flood design evaluations. Section 3.4.1.1 summarizes the plant's design flood levels, basic structural acceptance criteria and response plans for external floods. Section 3.4.1.2 describes the potential sources of internal flooding and the design elements intended to mitigate the consequences of such an event. The evaluation methods for considering static and dynamic effects of flood loads are provided in Section 3.4.2, with results summaries for the drywell (Section 3.4.2.1), torus (Section 3.4.2.2) and radwaste building basement and high level activity waste tanks (Section 3.4.2.3).

3.4.1 Flood Protection

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

3.4.1.1 External Flood Protection Measures

Quad Cities' external flood control efforts are directed towards the prevention of damage resulting from the occurrence of the probable maximum flood (PMF) of the Mississippi River. The PMF, described in Section 2.4.3, produces a flood to elevation 589 feet at the Quad Cities site.

The initial structural design was based on a water level of elevation 590 feet, using a 33% increase in normal allowable stresses as permitted by code. This initial design flood elevation is below the plant grade of 594.5 feet and any mode of operation is, therefore, possible without additional protective measures. [3.4.1]

Evaluations for higher flood levels show that elevation 603 feet, which envelops the PMF elevation, is the maximum flood elevation which can assure that the plant can be shut down and maintained in a safe condition. The flood resistance of the plant to these higher floods is based on the following criteria:

For a flood occurring to the level of the plant grade elevation of 594.5 feet, any mode of operation is also possible, and no additional protective measures are needed to maintain structural integrity. Stresses in the structures would, in some cases exceed the 33% increased allowable stress used in the initial criteria. However, these higher stresses would not exceed the allowables corresponding to .85 f'_c (f'_c = concrete compressive strength) and .90f_y (f_y = reinforcing steel yield strength). This more liberal criteria is justified for a flood occurrence of this magnitude. For a flood of any elevation from 594.5 feet up to the surrounding ground elevation of 603 feet, the plant can and will be maintained in a safe condition by flooding the plant buildings to match the river elevation. A flood of this magnitude would provide sufficient time to enable shutdown procedures to take place and flooding of the structures to be initiated. Under this condition no heat removal nor related electrical power supply systems are needed.

The emergency protection measures available to minimize the impact of an external flood event are described below. [3.4.2]

The flood stage levels on the Mississippi River are predicted several weeks in advance by the U.S. Army Corps of Engineers. In the highly unlikely event that a maximum probable flood is predicted, steps to shutdown and cool the plant will be initiated a minimum of three days prior to the predicted time at which the water will go above plant grade elevation of 594.5 feet. This will reduce the decay heat from the reactor to a level which can be removed by natural circulation cooling between the reactor and the reactor cavities and storage pools.

Once the reactor is shutdown and cooled down, the drywell and reactor vessel heads will be removed. With the penetration hatches in the drywell bellows seal structures open, each unit's torus will be completely filled and the drywells will be filled to the level corresponding to the level attained when the torus is full. The hatches in the bellows seal structure will then be closed and the reactor cavities and dryer-separator pools will be filled to the level of the spent fuel pool. Finally, the spent fuel pool gates will be removed permitting free circulation of water through the storage pools, reactor cavities and dryer-separator pools.

The supply of water for this operation will be the two 350,000 gallon storage tanks, the 100,000 gallon clean demineralized water storage tank and river water as required. When the river level reaches plant grade, the plant doors will be opened to allow water levels to equalize forces on the building structures.

The basic procedure described above will be accomplished as follows. The torus will be completely filled and the drywell will be filled to the level attained when the torus is full using a six inch fire hose connecting the Fire Water Supply System (FWSS) to the RHR system. This connection will use the existing disconnect between the Drywell Spray isolation valves 1(2)-1001-23A and 1(2)-1001-26A. The river water from the FWSS will be routed into the RHR system and into the torus. It will take less than 12 hours to add the 1,000,000-gallons of water required to fill the torus completely. With the torus completely filled, the water level in the drywell will be just below the recirculation pump motors. [3.4.3]

After the torus is filled, the two 4700 gal/min core spray pumps will be started and the contents of the two 350,000-gallon storage tanks and the 100,000-gallon clean demineralized water storage tanks will be pumped into the reactors filling the reactor cavities and dryer-separator pools to approximately 25 percent full. River water from the FWSS, through the RHR system or Core Spray system, will then be used to complete filling the pools to normal level. This operation will take approximately 3 hours.

The gates between the fuel storage pools and the reactor cavities will then be opened allowing free circulation of water through the storage pools, reactor cavities and dryer-separator pools.

Independently powered portable pumping equipment available onsite will be set up above the projected flood elevation to supply the 200 gallon per minute makeup water required in the storage pools due to the evaporative cooling loss. A 250-gallon fuel supply is available on the site to supply this equipment. Prior to flooding the drywell, all electrical equipment will be de-energized. The dose rate at the property fence line will be 0.01 mr/hr based on the above evaporation rate.

Electrical equipment below elevation 608 feet will be racked out of service after the pumping operations are completed and prior to flood stage exceeding grade level. Plant doors will be opened to allow water levels to equalize on building walls.

A flood of this type would be expected to recede down to grade level in about 8 days based on information received from the U.S. Army Corps of Engineers at the Rock Island, Illinois Arsenal.

Plant cleanup operations will be started when the water level reached grade level. The plant lower levels will be de-watered, motors dried out and cleaned, electrical beakers dried out and cleaned up and electrical system checkouts made. The electrical system checks will be made in the same manner as performed from the preoperational test program to assure that all systems function as designed.

Generalized steps for securing the station in the unlikely event of a flood above the level of the historic flood of record is described below. [3.4-3a]

These steps will be initiated a minimum of three days prior to the predicted arrival of a flood of elevation 594.5 feet or greater, and both units will be shut down and decay heat will be removed using the normal procedures as specified for the RHR system.

Steps

1. Shut down both units (normal procedures)
2. Remove decay heat (normal procedures)
3. Install RHR system 6" Firehoses crosstie to Fire Water Supply System (FWSS)
4. Fill both tori with water through 6" Firehoses to RHR system
5. Remove shield plugs, drywell heads and reactor vessel head
6. Set up the portable pumping equipment
7. Fill radwaste tanks with fire system water
8. Place portable makeup demineralizers in service to fill condensate storage tanks
9. Fill reactor cavities and separator-dryer pools using core spray system and RHR system
10. Remove gates between storage pools
11. Rack out all main breakers for equipment below elevation 608 feet
12. Open plant doors

3.4.1.2 Internal Flood Protection Measures

Internal flood control efforts at Quad Cities are directed towards the protection of the Class I plant structures from internal sources that can conceivably introduce large amounts of water into below grade areas. 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.

3.4.1.2.1 Protection of the Condensate Pump Room and Residual Heat Removal Service Water Pump Rooms

On June 7, 1972, the failure of a rubber expansion joint caused flooding of the condensate and service water pump room[1]. As a result, water-tight Class I vaults with water-tight Class I bulkhead doors have been constructed to isolate the RHR service water pumps and diesel generator cooling water pumps from all other equipment in the condensate pump rooms and to protect the pumps from being flooded by a failure of either the condensate, condensate transfer or clean demineralized water systems. [3.4.4]

Basically two design modifications have been made as a result of the flood of June 7, 1972: [3.4.5]

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 RHR service water pumps and the diesel generator cooling water pumps.

B. The RHR service water pumps and the diesel generator cooling water pumps have been enclosed in three separate watertight vaults in each unit. These vaults are designed to keep the sources of flood water in the condensate pump room from flooding the RHR service water pumps and the diesel generator cooling water pumps.

The design details of these two modifications are discussed in the following sections:

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 the 586-foot elevation:

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

   The ventilation closures have been analyzed and found acceptable to withstand a head of 30 feet of water in addition to a 0.16 g (design basis earthquake (DBE) load value) vertical acceleration as applied to the mass of water. The plate and structural members providing these closures 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 a watertight door. The watertight door has been designed to withstand a head of 30 feet of water plus the effects of a 0.24 g (DBE load value) horizontal seismic load. Also included is the sloshing effect of the 30 feet head of water considered using 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, RTV 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 with the addition of 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 0.24 g horizontal and a 0.16 g vertical seismic occurrence. This includes sloshing as in paragraph B. The construction of the existing wall is more than adequate to contain 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 flooding of the condenser area. The following switches are installed:

	Level Above Floor	Function [3.4.6]
1.	1'0" (1 switch)	Alarm, condenser pit low water level
2.	3'0" (1 switch)	Alarm, condenser pit high water level
3.	5'0" (2 redundant switch pairs)	Alarm condenser pit hi-hi 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 the flooding to continue due to operation of the pumps. [3.4.7]

Should the switches at level (1) and (2) fail or the operator fail to trip the circulating water pumps on alarm at level (2), the actuation of both level switches in a pair at level (3) shall automatically trip the circulating water pumps of the affected unit and activate an alarm in the control room. If, however, only one level (3) switch of a pair of switches is activated, the alarm will come up, but the pump will not be automatically tripped. These redundant level switch pairs at level (3) are designed and installed to IEEE 279, "Criteria for Nuclear Power Plant Protection Systems."

3.4.1.2.1.2 Isolation of the RHR Service Water Pumps and the Diesel Generator Cooling Water Pumps from Flood Water

Each unit at Quad Cities Station has four RHR service water pumps and one diesel generator cooling water pump with an additional diesel generator cooling water pump located in Unit 1. Each unit has the above pumps in three separate and isolated vaults. The pumps are separated in the following manner, using Unit 1 as an example: RHR service water pump 1-1001-65A is in one vault; RHR service water pumps 1-1001-65B and 1-1001-65C and diesel generator cooling water pump 1/2-3903 are in a second vault; RHR service water pump 1-1001-65D and diesel generator cooling water pump 1-3903 are in a third vault. Unit 2 pumps are separated similarly except there is no duplicate 1/2-3903 diesel generator cooling water pump in Unit 2, and it is RHR service water pump 2-1001-65A (not 2-1001-65D) that shares a cubicle with the diesel generator cooling water pump. [3.4.8]

Each of the vaults was constructed by building a new flood protection wall around each of the three RHR service water pump areas in each unit. These new walls are keyed into the existing walls and slabs and anchored with drilled in anchors. In the case of the areas for the 1D and the 2A RHR service water pumps, new ceilings at the 572.5 feet elevation were also constructed. Ceilings at 572.5 feet existed over the other pumps.

The walls were designed and constructed to withstand a head of 30 feet of water outside the wall in addition to the combined horizontal effect of a 0.24g (DBE earthquake loading) seismic occurrence, vertical effect of a 0.16g (DBE earthquake loading) seismic occurrence, and a sloshing effect of the water per TID-7024 Nuclear Reactors and Earthquakes. The ceilings mentioned above were designed and constructed to withstand the maximum flood level in addition to the vertical effect of 0.16g seismic occurrence. The horizontal effect of 0.24g seismic occurrence was found to be negligible when applied to the ceilings.

The walls and ceilings 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 is expected to occur which would cause flooding of the RHR service water pump vaults.

Access into the RHR service water pump vaults is by means of watertight, steel, bulkhead doors. These doors are designed and constructed to the same flood and seismic criteria as the walls.

Other than fire protection piping, pipe penetrations through the vault walls are sealed in two different manners. Where the system is flexible enough to allow an anchor at the wall, the pipe is welded to a plate which is then welded to the pipe sleeve to form a watertight seal. Where the system is not flexible enough to anchor the pipe at the wall, a pipe sleeve is installed in the wall and the pipe is routed through it. Seals are designed to withstand a head of 40 feet of water. Fire protection piping is grouted at the wall penetration.

Electrical cables supplying power to the RHR service water pumps and the diesel generator cooling water pumps are sealed in the vault walls using a fitting which consists of a box frame which is cast into the vault wall.

Small conduit penetrations are sealed in a manner similar to the pipes discussed above.

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

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

The watertight doors on the RHR service water pump vaults are equipped with a control room alarm system. The alarm actuates from a limit switch on any one of the six doors (three per unit), and indicates that a vault door is open. [3.4.10]

Each of the RHR service water vaults contains a sump pump which collects any floor and equipment leakage inside the vault and pumps the water past three discharge check valves (one inside the vault, remaining two outside) to the service water discharge line. The two check valves outside the vaults are checked each operating cycle to assure their integrity. The previous floor drain design had a pipe penetrating into each vault which was normally capped and required manual action to drain a vault floor. Previous equipment drains that led to the turbine building equipment drain sump have been permanently capped. [3.4.11]

3.4.1.2.2 Protection of the Emergency Core Cooling System

Certain provisions have been made to detect the presence and location of an emergency core cooling system (ECCS) leak. These provisions are described in Section 6.3.2. [3.4.12]

Submarine doors have been added to each of the following ECCS pump compartments: the RHR rooms, the HPCI rooms and core spray rooms at elevation 554 feet of the reactor building. These doors will prevent water in the torus room area from leaking into the pump compartment and thus will assure the availability of the ECCS in the event of a passive failure.

Ball valves are installed in the floor drain lines of the reactor building corner rooms. The ball valves are normally closed to protect the ECCS equipment in the corner rooms by preventing water in the torus area or from one of the other corner rooms from flowing into the corner rooms through the drain lines. The ball valves may be manually opened from outside of the corner rooms to drain the rooms via the floor drains into the reactor building sumps. [3.4.13]

The internal flood protection measures described in Sections 3.4.1.2.1 and 3.4.1.2.2 establish that a single failure in a non-Class I system will not preclude safe shutdown of the affected unit. In order to preclude safe shutdown of the affected units, such flooding would have to disable the core cooling pumps located in five completely separate areas of each unit; i.e., RHR corner rooms A and B; core spray corner rooms A and B; and the HPCI pump room. However, each of these rooms is protected from adjacent areas by watertight doors and walls. Therefore, no single failure will cause flooding to more than one of these areas and no single failure of non-Class I systems will prevent safe shutdown of the affected units. [3.4.14]

3.4.1.2.3 Protection of the Drywell and the Torus

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 via the main vent lines and the downcomers into the torus. Without pumping, the sumps would overflow to the drywell floor, and once the level reached the main vent lines and then the downcomers, it 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. [3.4.15]

A reliable means of measuring primary containment water level during a post-LOCA core cooling contingency operation is provided for with indication on 90(X)-4 panel (LI 1(2)-1640-21).

3.4.1.2.4 Protection of the Electrical Cable Tunnel

Water level alarms are present in the electrical cable tunnel such that water level in the tunnel can be detected. [3.4.16]

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 designed to withstand the additional effects due to seismic loads when it is already in a flooded condition. Two conditions were considered critical in evaluating the drywell capability under seismic loading. The first is when the drywell is flooded to the knuckle which is the transition point between the spherical section and cylindrical section of the drywell (elevation 629.9 feet). The second condition is when the drywell is flooded to the normal pool level (elevation 689.5 feet). The results of the evaluations are contained in Section 3.8, and show that in both cases the critical buckling stresses are at the embedment of the drywell at elevation 576 feet, and that the stresses in the drywell under the combination of a flooded post-accident condition and either an operating basis [3.4.17] earthquake (OBE) or a design basis earthquake (DBE) remain below critical buckling stresses. See Table 3.8-1 for the summary of the stresses in the concrete at the embedment of the drywell at elevation 576 feet.

3.4.2.2 Torus

The torus and its supports were originally designed for a basic seismic acceleration of 0.3 g compared to the DBE value of 0.24g as assigned to this site. It was subjected to this high acceleration value in combination with the effects of the drywell being flooded to elevation 689.5 feet. See Section 3.8 for the discussion of the results of the evaluation of the torus due to a flooded drywell and a seismic load. [3.4.18]

It should be noted that the torus seismic evaluation has been revised during the Mark I program; see Section 3.8.

3.4.2.3 Radwaste Building Basement and High Level Activity Waste Tanks

The radwaste building is designed to withstand the uplift forces and pressures on the slab at elevation 572.8 feet and the walls below grade due to hydrostatic pressures of water at elevation 590 feet. Allowable stresses were increased by 33% over normal allowable stresses. The exterior walls below grade are coated with a waterproof coating and floor drains will carry off any seepage to a sump located in the basement of the radwaste building. [3.4.19]

In addition, if the largest tank in the radwaste basement (collector tank) failed, only 22,000 gallons of water would be released. This release would result in a basement water level of less than four feet. This level would not be capable of raising any of the basement tanks since all tanks are supported by legs about three feet above floor level.

The protection provided against flooding and uplift of the high activity waste tanks in the event of site flooding caused by heavy rains or a rise in the ground water table is as follows: [3.4.20]

A. The threshold elevation of the lowest opening into the radwaste building from the outside is 2.5 feet above grade.

B. The radwaste building concrete below grade has been sealed with a waterproofing agent.

C. Redundant sump pumps in the radwaste basement are provided to remove any water that would breach the above barriers. These pumps have a capacity of 50 gal/min each.

In addition, the time that exists between the flooding of the site above plant grade and subsequent flooding of the high activity waste tank rooms will permit the tanks to be filled with clean water to prevent uplift. [3.4.21]

3.4.3 References

1. Quad Cities Station Special Report No. 3A, Condensate Pump Room Modifications (Permanent Flood Protection of the RHR Service Water Pumps and Diesel Generator Cooling Water Pumps), November 7, 1972.

3.5 MISSILE PROTECTION

This section describes the missile protection of applicable station structures, equipment or 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 is provided below in terms of the missile sources such as internally generated missiles, turbine missiles, missiles generated by natural phenomena, or events near the site.

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 should occur, redundant equipment will remain available to perform the safety function.

Electrical equipment and wiring for primary containment isolation systems (PCISs), high pressure coolant injection (HPCI), low pressure coolant injection (LPCI), core spray (CS), and 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 will be assured. Separation requirements apply to control power and motive power for all systems concerned. Arrangement and/or protective barriers have been erected such that no locally generated force or missile can destroy both redundant PCIS, HPCI, LPCI, CS, and ADS functions. In the absence of confirming analysis to support less stringent requirements, the following criteria were generally followed for initial plant design considerations: [3.5.1]

A. In rooms or compartments having rotating heavy machinery (such as the main turbine generator and 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 safety systems redundant to each other and located in a missile prone zone such as discussed previously 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. The pipe whip restraints provided on high-energy lines outside the containment are described in Section 3.6.1.

3.5.2 Internally Generated Missiles

Missile protection is given special consideration under assumed accident conditions. The following summarizes the pertinent design considerations. [3.5.2]

The driving force for potential missiles within the containment comes 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 greater kinetic energy.

Consideration was 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 because they are 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 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 preceding conservative design criteria it was found that no small missiles (e.g., thermowells and small valve components) 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 calculation method used to determine the energy required to penetrate the containment shell is 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:

$$\frac{E}{D} = U(0.344T^2 + 0.032T)$$

where:

E = Critical kinetic energy required for penetration, ft-lbs

D = Diameter of missile, inches

U = Ultimate tensile strength, psi

T = Plate thickness, inches

This equation has been plotted for the various thicknesses of the drywell shell and is shown in Figure 3.5-1.

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

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 insufficient distance available between the stem and drywell to achieve the energy necessary to penetrate the 0.75-inch thick containment. Also, the bonnet would be deflected by obstructions, hangers, or uneven failure of the bolting.

It has been shown in experiments conducted by CB & I (Reference 3.5.6-1) that safety margins exist in the containment shell under missile type loadings. The CB&I experiments are discussed in Section 3.6.

Small missiles do not achieve a high enough velocity to attain an energy level sufficient to penetrate sound containment shell material. [3.5.3]

Since the missile load is a limited displacement load, and the tests indicated the containment vessel could withstand the postulated displacement without impairment, the vessel will withstand the combined strain effects of D + P + H + R + T + E without impairment. These symbols are defined as follows: [3.5.4]

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 [3.5.5]

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

T = Thermal load on containment, reactor vessel, and internals due to LOCA

E = Design earthquake load, peak ground accelerations are 0.12 g horizontal and 0.08 g vertical

Hence, it was concluded that maintenance of the containment integrity during missile loadings would be assured. [3.5.6]

Where possible, consideration was given to achieving missile protection through basic plant component arrangement such that, if failure should occur, the direction of flight of the missile is away from the containment vessel. In addition to the care with which equipment is oriented with regard to missiles, special care was taken in component arrangements to see that equipment associated with engineered safety systems, such as the core spray and the containment spray, were segregated in such a manner that the failure of one would not cause the failure of the other, or that the failure of any component which would bring about the need for these engineered safeguard systems would not render the safeguard system inoperable. Additionally, the control rod drive mechanisms are located in a concrete vault that provides protection from potential missiles. The suppression chamber has essentially no source of internal or external missile generation and the vent pipes connecting it to the drywell are protected by the jet deflectors. [3.5.7]

3.5.3 Turbine Missiles

In the unlikely event of a turbine disc breakup, there are no components associated with a safe shutdown of the plant that could be damaged by potential missiles originating at the turbine. [3.5.8]

Since the turbine missiles would fly away circumferentially, and since the control room is located on a line coincident with the turbine centerline, it is unlikely that the control room would be affected. Furthermore, its enclosure consists of approximately 2 - 3 feet of concrete. [3.5.9]

The diesel generator rooms and battery rooms, are remotely situated from the turbine-generator, and are at other floor elevations and would also be unlikely to be affected by missiles. They are redundant and separated. Some of the electrical buses are adjacent to the turbine-generator and are located on the same floor or a floor just above the main turbine floor. Section 3.5.1 addresses the physical separation and protective barrier criteria which provide added assurance for the survivability of the redundant electrical equipment of the PCIS, HPCI, LPCI, CS, and ADS. As described in Section 3.5.1, and in the unlikely event of a turbine-originated missile striking the electrical gear, only one of the redundant supply systems would be affected.

The replacement HP rotor consists of an integral rotor, without shrunk-on wheels. The new HP turbine rotor is not considered a source for potential missile generation, and therefore, an HP turbine rotor missile probability analysis is not required.

All original low-pressure turbine rotors were replaced with a design that was less susceptible to stress corrosion cracking. The original rotor design had “shrunk-on” rotors, which were replaced with rotor discs that used a “welded-on” design to minimize stress corrosion cracking. However, the replacement rotors continued to be susceptible to stress corrosion cracking in the area of the blade-to-wheel attachment, requiring greater inspection frequency. The susceptibility to stress corrosion cracking is due to the impact of high moisture content steam in a high stress area of a high strength material. The low-pressure rotors were once again replaced with rotors designed to address the problems at the wheel attachment to reduce the required inspection frequency. The design for the current rotors have lower stresses and are able to use lower yield strength alloys that are not susceptible to stress corrosion cracking.

The generator rotor is encased in the heavy stator, and any failure of the rotor would not penetrate the casing.

The normal rotating speed of the turbine is 1800 rpm. The total calculated rotational energy contained in the high-pressure spindle rotating at 1800 rpm, is approximately 2.5×10^8 ft-lbs. If approximately one-fourth of this energy is available for each quarter section in a postulated fracture, each piece would have an energy of 6.25×10^7 ft-lbs available to it. If no energy were consumed in the fracture and in passing through the casing, the theoretical maximum separation velocity corresponds to about 350 ft/s.

The potential trajectory of any missiles produced by failure of the turbine rotor or blades consists of the following:

A. The piece leaving its point of origin would, assuming sudden release, be tangential to the rotating element. The probability of direction is equal throughout 360°.

B. Pieces leaving in a direction horizontally or in the downward 180° included angle would follow approximately a straight-line path unless deflected by objects encountered in the flight path.

C. Missile pieces leaving in the upward 180° included direction would follow a parabolic path unless otherwise deflected.

D. A piece going straight up would, of course, follow a straight line path unless deflected.

For each potential target point, at the same elevation as the turbine shaft, there are two angular directions that the missile could leave its point of origin. At 45° there is only one target point and this would correspond to the maximum range.

Specifically, the turbine-generator at Quad Cities is located at elevation 639 feet. The refueling floor level of the reactor building is at elevation 690.5 feet, and the centerline of the turbine-generator is separated by 125 feet from the centerline of the reactor vessel.

If a potential missile originating at the turbine rotor were released, the following directions can be considered of no consequence:

A. A missile headed (projected) in any direction in the downward 180° included angle, since the approximately 12-foot thick turbine foundation shields the plant areas sideways and approximately down to 60? below horizontal. In the downward direction the pieces would penetrate the condenser and probably end up on the basement floor. The turbine foundation further serves to shield the main steam and feedwater isolation valves at the primary containment penetrations. A potential missile could rupture a steam or feedwater line, but only on the turbine side of the isolation valves.

B. A missile projected in all directions above the turbine main floor and to the west of the plant, including the straight up direction, since these are directed away from the reactor building.

C. A missile projected in all directions toward the reactor building up to approximately 35° above horizontal. Missiles travelling in this direction would strike the concrete wall separating the reactor and turbine building. This wall varies in thickness from1.5 – 4 feet depending upon shielding requirement, etc. Beyond this wall is the 4 – 6 feet thick concrete cylinder that surrounds the primary containment vessel. Because of the double barrier, it is believed that no potential missile could penetrate these two barriers. In addition, some noncritical equipment is located in the intervening space which would serve to intercept the missiles.

   This leaves the included angle from about 35° above the horizontal to 90°; i.e., straight up, toward the reactor building as an area for missiles to travel and land on the refueling floor of the reactor building. It is impossible to estimate the energy of the pieces consumed in either the rotor fracture or by passing through the heavy steel double casing of the high pressure turbine. Furthermore, some energy is consumed by a potential missile traveling through the turbine building roof, reactor building superstructure, etc. It appears that there exists an area approximately 5° in included angle toward the reactor building which could result in a missile travelling in a direction which would land the pieces on the 40-foot diameter shield plug over the primary containment vessel at the refueling floor. This plug is approximately 6 feet thick and is in place at all times during plant operation.

It is calculated that this plug could accommodate the force imposed by a 30,000 pound piece dropped from a 60-foot height. This would correspond to a height of approximately 110 feet above the turbine room floor. The calculations developed for this case are based on fracturing of the concrete plug and allowing the reinforcing steel to yield. No allowance has been made for energy consumed in the plug by elastic or plastic deformation which would take the reinforcing steel beyond the yield point, and which would still permit the plug to remain intact. In this event, it is likely that pieces of concrete would spall from the bottom of the plug and strike the top of the drywell. These pieces would not penetrate the steel cap of the drywell.

Based on the preceding information, the shield plug could safely absorb 1,800,00 ft-lbs. with no potential hazard to the primary containment. Undoubtedly, the shield plug could absorb considerably more than this without damaging the primary containment, but it is difficult to assess what this would be. It is believed that the plug would first deform until it rested on the drywell cover. The cover would then bend and would probably deform until it contacted the reactor pressure vessel head. It is likely that the drywell would not tear prior to contact with the reactor vessel. At this point the movement would cease since the vessel is supported on concrete, which in turn would transmit further downward forces into the building substructure.

If a missile were generated, it is likely that even small ones could damage the secondary containment, since the superstructure is an insulated metal panel with a thickness determined by resistance to weather and service conditions. The primary containment, however, is surrounded by 4 – 6 feet of concrete, plus an additional concrete building which tends to protect it. It is, therefore, believed that no potential missile originating at the turbine would penetrate the primary containment from the side. This could only damage the primary containment if a massive rotor piece were directed upward in a very specific direction permitting it to come to rest on the shield plug covering the drywell at the refueling floor. Even in this case it would have to be dropped from a distance in excess of 60 feet above the reactor refueling floor elevation.

3.5.4 Missiles Generated by Natural Phenomena

A major consideration of wind loads is the effect of missiles generated by a tornado. The reactor building walls have been analyzed to determine this effect. [3.5.11]

Two types of missiles have been considered:

1. A utility pole 35 feet 0 inches long with a butt diameter of 13 inches and a unit weight of 50 lb/ft^3 having a velocity of 150 mph; and

2. A 1-ton mass with a velocity of 100 mph with a contact area of 25 square feet.

The walls were analyzed for the effect of these missiles and the analysis was based on ultimate stresses. The utility poles were considered to have perpendicular incidence at the 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 Standard Review Plan, Paragraph 3.5.3, the FSAR tornado missiles will not penetrate or cause scabbing of the exterior building walls.

In addition, reactor building hatch covers are designed such that they will not lift and act as heavy missiles during venting due to differential pressure caused by tornados.

3.5.5 Missiles Generated by Events Near the Site

Explosive materials on or near the plant site, including hydrogen that is stored as part of the hydrogen water chemistry program, are covered in Section 2.2.

3.5.6 References

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

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 and secondary containment, and all essential equipment inside or outside primary containment, including components of the reactor coolant pressure boundary, have been adequately protected against the effects of blowdown jet and 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 Primary Containment

3.6.1.1 High Energy Piping

In December 1972, the AEC issued letters to the licensees of all operating nuclear power plants, including Quad Cities Units 1 and 2, requiring reviews of the effects of piping failures outside of the primary containment structure. The letter applicable to Quad Cities[1] referenced General Design Criterion 4 of 10 CFR 50 Appendix A, as well as the "previous version" of these criteria, as the bases 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. Commonwealth Edison Company (CECo) responded to this request with Special Report No. 12 (Reference 2). [3.6.1]

Modification M04-1(2)-91-027A&B was performed to the RWCU system which replaced non-safety related sections of piping that were susceptible to Intergranular Stress Corrosion Cracking (IGSCC). Due to the pipe size and pipe routing changes, calculation EMD-068129 was performed to supplement the HELB analysis for the Reactor Water Cleanup piping (Reference 3).

The following paragraphs summarize the CECo report and the NRC's evaluation of the report. [3.6.2]

3.6.1.1.1 Criteria

A summary of the criteria and requirements included in the AEC letter of December 18, 1972,[1] is set forth in the following subsections:

3.6.1.1.1.1 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 and 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 included pipe whip, structural (including the effects of jet impingement), and environmental.

The criteria used to determine the design basis piping break locations in the piping systems are as follows: [3.6.3]

A. ASME Section III Code Class I 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 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 exceeds 2.0 S_{M} for ferritic steel, and 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 and based on all normal, upset, and testing plant conditions exceeds 0.1; and

4. At intermediate locations in addition to those determined by the previously listed locations, 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 Code 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 Code, Section III, or the USA Standard Code for Pressure Piping, ANSI B31.1.0-1967; and

3. Intermediate locations in addition to the previously listed locations, 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 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 from 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 1/2 the pipe diameter in length and 1/2 the wall thickness in width (defined as "critical crack size"). [3.6.4]

3.6.1.1.2 High Energy Systems

For pipe breaks outside primary containment, the following systems were analyzed:

A. Main steam (MS),

B. Feedwater (FW),

C. High pressure coolant injection (HPCI),

D. Reactor water cleanup (RWCU), and

E. Reactor core isolation cooling (RCIC).

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.

The following systems were considered but not analyzed due to physical separation or absence of impact on equipment important to safety:

A. Extraction steam to heaters A, B, C, or D,

B. Heater drain from heaters C or D,

C. Condensate booster,

D. Moisture separator drain, and

E. Control rod drive hydraulic system.

3.6.1.1.3 Areas or Systems Affected by High Energy Line Breaks

An evaluation was conducted, as documented in Special Report No. 12, and supplemental calculation EMD-068129 for the RWCU piping, of the effects of high energy line breaks (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 and cool down the reactor and maintain cold shutdown conditions.

3.6.1.1.3.1 Systems and Components

A. Control and instrument cables and raceways,

B. Electrical distribution system,

C. Emergency ac power supply (diesels),

D. Heating and ventilation systems (needed for long-term occupancy to maintain the reactor in a safe shutdown condition),

E. Reactor control systems and associated instrumentation,

F. Cooling and service water systems, and

G. Emergency core cooling system (ECCS) components.

3.6.1.1.3.2 Structures

A. Primary containment and torus,

B. Main steam tunnel,

C. Control room,

D. Diesel generator rooms,

E. Auxiliary equipment rooms,

F. RCIC rooms,

G. HPCI rooms,

H. Reactor building, and

I. Battery rooms.

3.6.1.1.4 Specific Areas of Concern

Commonwealth Edison Company Special Report No. 12 (Reference 2), 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 the following specific areas of concern 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 1 and 2. Subjects covered include compartment pressurization, pipe whip, compartment flooding, environmental effects, and control room habitability. As a result of the performance of Modification M04-1(2)-91-027A&B: "The Replacement of IGSCC Susceptible Piping, Valves and Regenerative Heat Exchangers for the RWCU System," calculation EMD-068129 was subsequently performed to supplement the HELB analysis (Reference 3). GE SIL No. 604 discussed the possibility that, for certain plants, the Reactor Water Clean-up (RWCU) System pipe break analysis was non-conservative, and that under certain conditions, the system isolation logic might not isolate the postulated break. This condition was evaluated for the Quad Cities units and resulted in the RWCU Automatic Isolation Modification.

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 previously, were considered for pipes in the main steam tunnel, outside primary containment, and the turbine building. The compartment pressurization calculations included pressure plus impingement forces. [3.6.5]

Each of the facility's steam tunnels is divided into two 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, simultaneous rupture of one MS line and two adjacent FW lines was assumed. Main steam isolation valve 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 could easily withstand this transient. For Unit 1, 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 2, were those which could produce such damage. For Unit 1 only, CECo committed to provide improved support for the blowout panels by:

A. Installing an additional W36 x 135 beam on top of the existing W24 on the east-west wall of the tunnel, 11 feet east of column row "G";

B. Replacing the existing 6B x 12 east-west beams supporting the panels with five W21 x 55 beams; and

C. Upgrading the chains restraining the blowout panels.

All of the high energy line break modifications have been completed as documented in a November 17, 1977 letter from M. S. Turbak (Commonwealth Edison) to Karl L. Goller (NRC).

For Unit 2, the blowout panels are somewhat removed from the areas of potential pipe ruptures. The pressure transient resulting from the above calculation would not lead to damage to the safety related cabling (as in Unit 1); thus, no modification was proposed.

In the reactor building, the consequences of HELBs and cracks in the HPCI, RWCU, and RCIC systems were evaluated. Damage to the torus, a steel-walled steam suppression chamber used as a heat sink in several modes of operation of the ECCS, could occur as a result of certain HPCI and RCIC pipe longitudinal breaks identified in Tables 13 and 15, respectively, of Reference 2. Commonwealth Edison Company provided pipe restraints at the critical break points to reduce the impingement loads. These restraints consisted of U-shaped plates covering the break points and anchored to the nearest structure. [3.6.6]

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 MS and FW system pipe breaks in the steam tunnel, for HPCI, RWCU, and RCIC system line breaks in the reactor building, and for MS and FW system pipe breaks in the turbine building. This evaluation included the double-ended rupture of the largest pipe in the MS and FW 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 MS 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 MS system pipes indicated in Table 16, of Reference 2. For Unit 1, the modifications proposed to mitigate the consequences of the pressure transient would also serve to mitigate the consequences of the pipe whip transient. For Unit 2, CECo committed to add an additional north-south beam (east of the main steam lines) approximately 15 feet west of column row "H" to protect against pipe whip.

In the reactor building, whipping HPCI or RCIC piping, resulting from circumferential line breaks indicated in Table 17 and 19, respectively, of Reference 2, could damage the torus (described previously). Other whipping HPCI piping, identified in Table 17 of Reference 2, could damage the core spray valve, MO-1-1402-4A, and other smaller HPCI piping. The U-shaped restraints (described in paragraph 3.6.1.1.4.1) installed at the critical break points would serve to mitigate the consequences of the pipe whip transient.

All of the high energy line break modifications have been completed as documented in a November 17, 1977 letter from M. S. Turbak (Commonwealth Edison) to Karl L. Goller (NRC).

Postulated pipe whip calculations for HELBs in the 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, which is covered in Special Report No. 12. Although the main steam isolation valves 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. All electrical cables required for safe shutdown enter the reactor building in an area unrelated to the steam tunnel at an above-grade elevation when there is no flood potential. The effects of flooding from loss of piping integrity for other than the main steam or feedwater lines, and from 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 the HELB. Adverse temperature, pressure, and humidity were the parameters which were used in the evaluation of safety-related equipment.

The NRC reviewed CECo's assessment of the consequences of environmental effects on safety-related electrical equipment and found that 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 control rod drive (CRD) hydraulic system was investigated in detail following the 1980 Browns Ferry 3 failure to scram event, as required in NUREG 0803. 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 4.6.4.2. [3.6.7]

3.6.1.3 Instrument Line Break Outside Primary Containment

Following a review of the FSAR, the AEC raised several questions concerning instrument lines originating at the reactor coolant pressure boundary and terminating outside primary containment. Since the main issues addressed were secondary containment integrity and potential offsite radiological consequences, this subject is covered in Sections 6.2.3 and 15.6.2, respectively.

3.6.2 Postulated Piping Failures in Fluid Systems Inside Primary Containment

This section contains design bases and evaluations related to two categories of high energy piping located within the drywell. The first category includes only the reactor recirculation (RR) system. The second category includes all other large reactor coolant pressure boundary (RCPB) lines, such as main steam and feedwater. As described further in the two subsections covering these categories, very different approaches were taken in consideration of mechanistic pipe break effects, that is, pipe whip, jet impingement, and jet thrust reactions on broken lines. Most of the differences resulted from the time frame of original design development, during the transition period between consideration of a LOCA as a hypothetical event, postulated primarily for sizing of containments and ECCS, and treatment of all line breaks in a primarily mechanistic manner, as required by 1971 General Design Criterion number 4. The latter requirements had a particularly significant impact on boiling water reactor (BWR) MK I containment design considerations.

3.6.2.1 Recirculation Loop Piping System

The recirculation lines within the primary containment are provided with a system of pipe restraints designed to limit excessive motion associated with pipe split or circumferential break. The design utilizes a number of supports and limit stops which permit thermal expansion of the pipe. Both types of breaks, the circumferential break or the longitudinal split are considered in the support and limit stop arrangement. [3.6.8]

3.6.2.1.1 Design Bases

The design bases for the system of pipe restraints on the recirculation loop piping system are as follows: [3.6.9]

A. The pipe ruptures are assumed to occur anywhere in the system and are assumed to be instantaneous circumferential guillotine breaks or longitudinal pipe splits. (Note exception in item G).

B. The maximum distance between restraints is no greater than that distance between restraint bracket and the containment drywell shell plate. (Note exception in item G).

C. The position of restraint brackets limits excessive motion to assure that a rupture in the recirculation system does not result in cascading pipe and system failures so that safe shutdown of the reactor is possible.

D. The pipe restraints are arranged so as not to interfere with the normal operation of the system, including earthquake motions.

E. The allowable stresses for the restraint brackets and support steel are 150% of the AISC code allowable for the material used. The restraint ring is designed to limit bending stresses within the ultimate strength of the material and to limit the tensile stress to 90% of yield. The recirculation pump restraint cables (wire rope) are limited to 90% of their breaking strength.

F. The design loads for the restraint system are the product of the reactor vessel operating pressure times the pipe flow area. The bases for establishing the design loads for the restraint system as stated are as follows:

   1. The pipe rupture would occur at normal operating pressure. [3.6.10]

   2. Pressure drop through pipe fittings, elbows, valves, pumps, jet-pump throttling, and nozzle orifice drops was not considered. This is conservative.

   3. Pressure decay of the system was not considered in the static analysis of the pipe restraint system. This is conservative.

   4. The free flow area of the pipe was used. This is conservative.

   5. Thrust load factors for the blowdown of this system were not considered. This is not conservative, but is more than offset by the conservative items above.

   6. The pipe rupture would occur in a slow predictable manner, forcing the relatively flexible piping system against the restraint bracket in an equivalent predictable manner.

G. One restraint (HER# 0200-G-106) has been removed from the suction piping of the Unit 1 system utilizing the relaxation in arbitrary intermediate pipe rupture requirements from NRC Generic Letter 87-11 and the pipe break location criteria from BTP MEB 3-1 (GL 87-11 attachment).

3.6.2.1.2 Design Evaluation

The recirculation pipe restraint system consists of a thick steel ring surrounding the pipe and anchored to the pedestal or shield wall by a steel bracket and structural steel support members. After a pipe rupture, the pipe (if moving radially outward) would bear against the restraint ring causing deformation. The restraint ring will stretch and deform around the pipe taking the configuration of the pipe. The theoretical bending stress in the restraint ring is the force times the radius of the ring based on an elastic analysis. [3.6.11]

The bending stress in the ring is self-relieving beyond the yield stress of the material and this stress is limited to the ultimate strength of the material. The tensile and sheer stresses are limited to 90% of yield strength of this material.

The bending stress in the ring structure of pipe restraints at the location where the ring is attached to the support base has been investigated. [3.6.12]

The design of the restraint bracket calls for the outside face of the base supporting plate to be tangent to the inside face of the restraint ring. This alignment tends to minimize the effect of bending at this joint. Under design load conditions, however, there will be a bending stress in the ring at this joint. This stress could be somewhat in excess of the normally defined yield stress of 1.2% elongation of the material when a constant moment is considered. However, this is a self-limiting stress in that the bending of the ring to match the pipe shape reduces the moment arm. This reduction results in a lesser moment than when calculated as fixed input load system. Even assuming that the bending moment is constant over a ring length of 10 inches, the maximum bending strain in the joint is approximately 3% elongation. For this material (A36) the minimum strain elongation capability is approximately 20%. Thus, even without accounting for the moment reduction characteristics built into the system, it can be seen that a substantial margin of safety against this type of failure is available.

On the basis of the assumptions described above, for the worst case (the most highly loaded restraint ring) our examination of combined stresses has shown that the strain is less than 4%.

The pipe restraints on the recirculation loop have the following average safety margins calculated by the allowable stress (0.9 F_y) divided by the average calculated tensile and shear stress: [3.6.13]

Line O.D.	Stress
28 inch	1.55 ksi
22 inch	1.43 ksi
12 inch	3.10 ksi

Plastic deformation of the piping at the restraint was considered in this analysis. If the pipe rupture was of a nature to load the pipe beyond the elastic limit of the pipe, then a plastic hinge would form in the pipe at the restraint. The length of pipe from postulated break to the restraint was maintained less than the distance from the containment vessel shell to this plastic hinge, and therefore plastic deformation of the pipe was of no concern.

The pipe analysis and effects of a subsequent pipe rupture in an adjacent system or the same system is not considered due to the following:

A. The restraints limit the extent of damage to the ruptured system within the immediate area,

B. Pressure loads in the system are immediately decreased, even with a postulated 12 inch pipe rupture, and

C. The effect and impact on adjacent piping systems is nominal.

3.6.2.2 High Energy Piping Inside Primary Containment Other Than Recirculation Loop

With the exception of containment penetrations, addressed in Section 3.6.2.2.1 below, the GE approach to pipe break considerations for systems other than reactor recirculation was prevention of pipe break interactions. This was accomplished by a combination of minimization of the probability of occurrence of a pipe break through use of high quality material and extensive non-destructive examination, and minimization of potential interaction effects by proper physical arrangement of systems piping and equipment, as addressed in Section 3.6.2.2.2 below. The AEC did not fully accept this approach and mandated further risk reduction measures as described in Section 3.6.2.2.3 below. [3.6.14]

3.6.2.2.1 Containment Penetration Protection

In the design of the primary containment and the components therein, special consideration has been given to mitigation of pipe whip and jet force effects under the assumed accident condition of a double-ended rupture of a MS, FW, or other large lines in order to maintain containment penetration integrity. [3.6.15]

All large pipes that penetrate the containment are designed so that, they have anchors or limit stops located outside of the containment to limit the movement of the pipe if necessary. 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.

The space between the containment vessel and the concrete is controlled so that, in areas which are backed up by concrete and are subjected to jet forces, the integrity of the containment will not be violated. Concrete backing is not available for the vent openings to the suppression chamber and jet deflectors are put across these openings for jet protection.

Pipes which penetrate the containment shell, and which are capable of exerting a reaction force due to line thermal expansion or containment movement which cannot be restrained by the containment shell are provided with bellows expansion seals, appropriate guards, limit stops, or anchors as required to maintain stresses with allowable design limits. These design features are utilized to assure integrity of the penetration during plant operation and during accident conditions.

Pipes which penetrate the containment where the reactive forces can be restrained by the containment shell are provided with full strength attachment welds between the pipe and the containment shell. These penetrations are designed for long-term integrity without the use of a bellows seal.

3.6.2.2.2 Physical Arrangement

Special care is taken in component arrangements to see that equipment associated with engineered safety systems such as the core spray, low pressure coolant injection (LPCI) and the containment spray are segregated in such a manner that the failure of one cannot cause the failure of the other, or that the failure of any component which would bring about the need for these engineered safety systems will not negate adequate core cooling. Core spray lines enter the upper cylindrical portion of the drywell, separated from the recirculation lines which enter the lower spherical portion of the drywell, which separates these lines by about 24 feet. Also, each LPCI loop injects coolant through separate portions of the recirculation system. In addition, each containment spray header and support structure is designed to withstand a load equivalent to the jet forces associated with a break of the largest pipe within the drywell. With the exception of the incore monitoring detectors, sensors associated with the reactor protection system, including the drywell pressure detectors, are located external to the drywell and concrete structure, and are thus protected; however, the incore monitoring detectors are physically separated in the drywell, as are the sensing lines to the aforementioned other instruments. The control rod drive mechanisms are located in a concrete vault that provides protection. [3.6.16]

3.6.2.2.3 Containment Shell Pipe Whip Protection

3.6.2.2.3.1 Design Bases

During the original design of Quad Cities, explicit consideration of pipe whip impact on the drywell shell was limited to the reactor recirculation system. During FSAR review, however, the AEC initiated a requirement for consideration of lines other than recirculation. The responses, and succeeding AEC questions, eventually led to the AEC's mandating of additional drywell shell protection measures at Quad Cities similar to those provided in similar, contemporaneous BWR plants, such as Vermont Yankee. The following paragraph is a paraphrase of the AEC comment which mandated provision of these measures at Quad Cities. [3.6.17]

A high degree of confidence is required that, in the event of a pipe rupture that leads to pipe whip inside primary containment, the integrity of the containment barrier shall be assured such that an uncontrolled release of fission product is prevented. An acceptable level of protection is accomplished by the installation of suitable energy-absorbing structures within certain portions of the drywell.

The following criteria were used to determine the extent and location of application of energy-absorbing material: [3.6.18]

A. The piping systems having sufficient energy to rupture the containment pressure boundary are the MS, FW, HPCI, and RHR lines.

B. These large pipes are postulated to rupture at circumferential welds at changes in fluid flow direction, and areas of high stress.

C. The pipe whip movements are limited to directions normal to this ruptured plane and the jet reaction loads, causing pipe rotation at the next nearest point of system directional change or point of system rigidity (fixity, elbow, nozzle, etc.).

D. The ruptured component is assumed to move in an arc of $\pm$ 5° around a plastic hinge at the postulated point of rigidity.

E. The drywell areas requiring protection are those areas where the ruptured components are postulated to contact the drywell pressure boundary.

3.6.2.2.3.2 Design Evaluation

Calculations have shown 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 enroute 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 Chicago Bridge and Iron (CB&I) have shown that for an impact area of 1.08 ft^2 the containment shell would deform to the point of contacting the concrete (radial deformation of approximately 2-3/4 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. [3.6.19]

The 2.75 inches is based upon full collapse of the gap filler material and the available gap at ambient temperature. Both assumptions are very conservative. More realistically, the maximum gap that could occur is about 2 inches. Thus, based upon deflection, at least a 1.5 safety factor exists. In addition, the test deflection of greater than 3 inches is applicable for a load application condition of 1.09 square feet. Any jet force with sufficient magnitude to deflect the shell would occur over a larger area and hence a greater deflection could be tolerated and a significantly greater safety factor would result. For example, a main steam line break would result in the load being applied over a minimum area equal to the pipe area which is 1.755 square feet. This area is 60% greater than the application area used in the test, and in actuality, the jet impingement area would be substantially greater depending upon the distance from the break to the wall. It is therefore concluded that ample safety margin exists to assure containment integrity under jet force loading conditions. [3.6.20]

Refer to Section 3.5.1.2 for the combined loading conditions, including missile and jet loads, which the containment vessel will withstand.

An energy-absorbing system was installed at Quad Cities in accordance with the directive from the AEC. The energy-absorbing system absorbs the initial impact of the pipe section and distributes the force over a portion of the primary containment shell and containment shield wall concrete. The material used for this protection is special siding with 1/4-inch steel plate spot-welded to each face. This composite honeycomb material has been referred to as "Tornado Siding." This siding is capable of absorbing approximately 1 x 10^6 ft-lbs of kinetic energy/ft^2. [3.6.21]

The siding is attached to additional weld pads to the steel containment pressure vessel in 24 x 24-inch panels. The use of small panels permits the material to follow the contour of the vessel. The material, does not restrict access to piping welds or component welds for inservice inspection. The energy-absorbing system has a negligible effect on the free containment volume and consequently has no effect on the accident analysis. Installation details are shown on P&ID B-305 and B-835.

3.6.2.3 Interaction with Other Structures and Components

This section contains discussions of other pipe break interactions inside primary containment potentially involving all high energy piping systems. Specific sections cover the reactor vessel, the reactor shield wall (surrounding the reactor vessel) and the containment shield wall (surrounding the drywell). [3.6.22]

3.6.2.3.1 Reactor Vessel

Jet reaction forces on the reactor vessel were analyzed, with the reactor vessel 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 two 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 vessel would not be moved by jet reaction forces sufficient to cause rupture of other connected pipes. [3.6.22a]

3.6.2.3.2 Reactor Shield Wall

The reactor shield wall consists of a 24-foot diameter circular cylinder attached to the vessel support pedestal and extending upward approximately 45 feet. This cylinder forms the outer shell of the annulus; the inner shell is formed by the vessel wall and support skirt. The pedestal forms the base of the annulus with the top open to the drywell. The reactor shield wall is 27 inches thick and is constructed from 27-inch vertical I-beam columns, tied together by horizontal WF beams and 1/4-inch steel plates. These plates are welded to the column flanges, both inside and outside, thereby forming a double-walled shell. This shell is filled with concrete to provide shielding capability. The pipes leaving the vessel at elevations below the top of the reactor shield wall penetrate the shield. The penetrations in the vicinity of the core utilize removable shield plugs which fit around the pipe. The plugs are provided in order to allow access to the pipe welds for purposes of inservice inspections. These removable plugs are covered by two 9-inch thick steel plates attached to the shield wall by two vertical hinges, each 1- 1/2 inch in diameter, with both halves locked in place by a 1-inch diameter locking pin. [3.6.23]

This configuration was conservatively analyzed to determine the capability of the shield wall to withstand pressures generated in the annulus. The criteria utilized to estimate beginning of wall failure are that:

A. Only the two 1/4-inch plates, acting as a thin cylindrical shell, would resist the pressure forces, taking no credit for beam or concrete strength; that failure commences when shear stress at the fillet weld joining the plates to the column flanges reaches 90% shear yield.

B. Shear yield is only 1.5 times code allowable which would result in a failure commencement at approximately 24 ksi whereas shear ultimate is in the order of 40 ksi.

C. The pressure differential is a constant load, whereas the pressure differential would decrease as the drywell pressurized. For these assumptions, the beginning of wall failure would occur when the pressure in the annulus reaches 46 psi. This differential pressure is from the annulus across the shield wall to the drywell space.

The inner annulus wall is the vessel and its skirt. The skirt would withstand an external differential pressure of 150 psi before the onset of skirt buckling could commence. Hence, the shield wall is the more critical component.

Estimates of pressures that could be generated in the annulus were also made. The differential pressure is primarily a function of the break area and the annulus vent area. These parameters have been investigated parametrically to determine differential pressure as a function of the break to vent area ratio. The analyses assumed that 100% of the energy released through a given break size would enter the annulus. The largest pipe, the 28-inch recirculation pipe, has a cross-sectional area of 3.65 ft^2. Such size is equivalent to a 38 psi annulus pressure, but no pipe of this size is considered as being a credible break size within the annulus. The minimum wall thickness for the various piping systems occurs at the safe-end joint to the piping. All other sections from this joint back to the vessel have thicker wall sections and, therefore, have lower stresses. The largest line which has the safe end located in the annulus is the 4-inch jet pump instrument line nozzle. For all larger lines, the double-ended line break would result in the flow being directed into the drywell volume and not into the annulus. The rupture of this 4-inch line would result in a pressure differential across the reactor shield wall of 1 psid, which is considerably less than the capability of the shield wall.

The effect of power uprate to 2957 MWt on the biological shield annulus pressure has been determined to be less than 1.2 psi. This increase in pressure is well within the biological shield wall structural capability.

The effects of this 4-inch rupture in terms of missile generation were also considered. For Quad Cities, the 1 psid would not generate a missile because removable plugs are retained by hinged and locked doors as described. Forces would not be high enough to fail these attachments. Also, the effects of the shear drag forces created by a jet which would result from an open ended break of larger lines within the shield wall penetration was considered. The 12-inch jet pump inlet is typical of these larger lines where the analyses showed that, depending upon such factors as roughness coefficient, degree of two-phase flow and break size, a shear drag force of 1 to 8 kips would be created on the removable plug which would tend to move the plugs out against the doors. This 8 kips load, plus the additional 12 kips jet force load, is within the capability of the locking bars and hinges so that no missiles would be generated in this manner.

An analysis was also performed of the effects of jet forces resulting from a double-ended line break of the 4-inch line, assuming the jet forces from the break were to impinge directly on the removable plug, the resulting load would be 11 kips, which is less than the aforementioned restraining capability.

It is therefore concluded that the possible jet forces which could result from leaks inside the reactor shield wall would not lead to fracture of the shield with unacceptable consequences.

3.6.2.3.3 Drywell/Containment Shield Wall Gap

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 will withstand jet forces which may occur upon failure of any system piping. Where concrete is not available, such as at the vent openings, barriers are placed for jet protection. [3.6.24]

An evaluation was performed of the structural aspects and loads associated with steam or two-phase fluid mixtures which could enter the gap between the containment and containment shield wall. The mechanical and jet effects of this condition would not be of any great consequence because pressure within the containment would always remain greater than the pressure in the gap. The jet fluid escaping the pipe break pressurizes the containment, causing it to expand against the compressible material within the gap. This condition would limit the extent of pressurized volume in the gap caused by the jet and would restrict the distribution of the load. The loading condition of this jet on the containment could produce local yielding inward as the jet pushes against the plate. The containment shield wall has been specifically designed to withstand a 662 kip jet load from anywhere within the drywell. This capability is greater than the maximum jet load expected from any pipe in the drywell. [3.6.25]

All large pipes which penetrate the containment were designed so that they have anchors or limit stops located outside the containment to limit the movement of the pipe. These stops were 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 on the containment were taken as 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 have been provided with a system of pipe supports designed to limit excessive motion associated with a pipe slit or circumferential break. Vent discharge headers and piping were designed to withstand the jet reaction force caused by flow discharge into the suppression pool. [3.6.26]

3.6.3 References

1. Letter from A. Giambusso (AEC) to Byron Lee (CECo), dated December 18, 1972, "Effects of a Piping System Break Outside Containment," including errata issued on January 16, 1973.

2 "Analysis of Effects of Pipe Break Outside the Primary Containment," Quad Cities Station Units 1 and 2, Special Report No. 12, Revision 1, February 1975 (including Appendix E to the Report which was submitted on November 17, 1977).

3 Calculation EMD-068129, "Reactor Water Cleanup System Pipe Whip/Jet Impingement Loads," Quad Cities Unit 1 and 2, Revision 0.

3.7 SEISMIC DESIGN

At the time that the purchase specifications for the Quad Cities equipment were written, the seismic requirements were specified to be equivalent to the maximum ground motion accelerations. Initial seismic analyses unique to Quad Cities were performed using the Golden Gate Park spectra from the San Francisco earthquake of 1957. Dresden analyses were also used to obtain loads for the Quad Cities designs; the Dresden analyses were based on the El Centro earthquake of 1940. Subsequent to these initial analyses, a re-evaluation was performed using the Housner earthquake. [3.7.1]

The specific details of the application of these three input spectra and the methods used to perform the seismic evaluations are provided in the following sections.

3.7.1 Seismic Input

The seismic design criteria of structures and equipment were based upon the recommendations of seismologist Perry Byerly. Engineering consultants, John Blume and Associates, reviewed the seismology, geology, and other site data and recommended the criteria provided in Volume II, Appendix F, of the Quad Cities Plant Design Analysis Report (PDAR). [3.7.2]

The Class I structures, piping, and equipment (with the exception of the drywell) were initially seismically analyzed with the earthquake input corresponding to the Golden Gate Park south 80° east (S80E) component of the 1957 San Francisco earthquake normalized to 0.12g at the base of the reactor building (hereafter referred to as the Golden Gate Park earthquake). For structures and equipment analyzed using the response spectrum method, smoothed curves such as those shown in Figure 3.7-1 were used. For structures and equipment analyzed using the time history method, the actual Golden Gate Park record was normalized to 0.12g and used in the analysis; the unsmoothed response spectrum of this time history is shown in Figure 3.7-2. [3.7.3]

The Quad Cities drywell was evaluated using the results of the Dresden drywell analysis. The seismic input for the Dresden plant was the north-south component of the 1940 El Centro earthquake, normalized to 0.10g. Figure 3.7-3 shows the response spectrum for the El Centro earthquake which was used at Dresden. [3.7.4]

The seismic input ground motion considered in the re-evaluation of the structures, piping and equipment corresponds to the Housner response spectrum normalized to 0.12g. As shown in Figure 3.7-2, the Housner spectrum is higher than the Golden Gate Park spectrum for periods greater than 0.265 seconds. A similar comparison was made for the El Centro and Housner spectra. [3.7.5]

All Class I structures, piping, and equipment were re-evaluated to ensure that the results of the original analyses, based on Golden Gate Park and El Centro inputs, adequately enveloped those of the Housner input. For systems analyzed to the Golden Gate Park spectrum with fundamental periods less than 0.265 seconds, the spectral accelerations from the original analysis were higher than the corresponding spectral accelerations from the Housner event; therefore, the original analyses were acceptable. For those structures where the Housner event may control, reanalysis using the Housner input was performed.

The following Class I structures, piping, and equipment whose periods fall in the range where the Housner spectral accelerations exceed those of the Golden Gate Park earthquake were reanalyzed using the Housner spectrum and were evaluated using the higher seismic loads obtained from either of the two analyses. [3.7.6]

A. Reactor-turbine building system,

B. Main chimney,

C. Primary steam lines, and

D. Feedwater lines.

Details of these and other evaluations are provided in Sections 3.7.2. and 3.7.3.

Design response spectra for building-attached piping, equipment, and components were developed using the original response spectra, generated for specific damping levels, based on the Golden Gate Park earthquake. The peaks of the time history-derived spectra were broadened by 15%. The spectral accelerations for periods above 0.265 seconds were then manually adjusted for the higher input accelerations from the Housner earthquake. [3.7.7]

Design response spectra for additional damping levels were generated using synthetic time history generation methods. These spectra supplemented the original building-generated spectra described in the preceding paragraph. The complete set of final design spectra are used in the current qualification of Category I components.

All seismic input response spectra and time histories discussed above correspond to the horizontal loading condition. As previously discussed, the horizontal operating basis earthquake (OBE) load is based on a peak ground acceleration of 0.12g. In addition, a simultaneously acting vertical acceleration of 0.08g, constant for all periods, is applied to the structure and the resulting stresses added directly to the horizontal stresses. [3.7.8]

Results of these OBE analyses, combined with operating stresses, were compared against applicable codes without the usual increase in allowables for short-term loading to ensure compliance. An additional analysis was performed to ensure a safe shutdown during or after a seismic event of twice the OBE magnitude discussed, i.e., a horizontal ground acceleration of 0.24g and a vertical acceleration of 0.16g.

3.7.2 Seismic System Analysis

This section provides an overview of the methods used to perform seismic system analyses at Quad Cities. The subsections in Section 3.7.2.1 provide modeling and analysis details for the major Class I structures and mechanical systems at Quad Cities. Section 3.7.2.2 provides general information for the evaluation of Class II structures and systems.

3.7.2.1 Class I Structures and Systems

The overall responsibility for the adequacy of seismic design was maintained by GE with support from their consultants. The design basis for the Quad Cities seismic criteria was developed for GE by John A. Blume and Associates; Keith, Feibusch Associates, Engineers; and J. Sexton. The consultants completed the design analysis consisting of an investigation of both the flexural dynamic response and the rocking dynamic response of the subject buildings based upon the Sargent & Lundy (S&L) design drawings. Using the Golden Gate Park and Housner earthquake criteria, they established plant design envelopes of maximum acceleration, displacement, shear, and overturning moment versus height and transmitted this to the architect/engineer (S&L) for use in final seismic designs for all Class I structures. In addition, these consultants established floor response spectra for the Class I structures used by the piping designers (S&L and GE). All analyses were transmitted to the responsible organization via written data analysis reports. [3.7.9]

The design criteria for all seismic data were documented by the responsible organization in the form of "earthquake analysis data" reports. These reports were then transmitted to the design groups. These reports are in the form of descriptive data analysis, seismic curves, piping isometrics, and computer data listings. General Electric Company, through design reviews and drawing releases, monitored the seismic design to assure that all procedures were followed.

As previously stated, one of two methods of analysis of Class I structures and equipment founded directly on soil are considered acceptable; either a response spectra or a direct time-history method of dynamic response. The damping values used in the seismic analyses are listed in Table 3.7-1. The damping values in Table 3.7-1 are used in the OBE analyses but are also conservatively used in the design basis earthquake (DBE) analyses. Sufficient modes are considered to assure participation of all modes having periods that result in significant magnification. [3.7.10]

For seismically-analyzed systems, only the drywell and reactor pressure vessel (RPV) equipment are vulnerable to relative displacement of the supports. The drywell and RPV were analyzed dynamically to simultaneously account for both inertia forces and support displacements. 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 repairs as well as other RPV internals computer repair designs. The rebaselined model was modified to incorporate the core shroud repair hardware and was used to perform the seismic analysis for the repair design. The Golden Gate and Housner 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 Q2R13 and Unit 1 during Q1R14. (See Sections 3.7.2.1.4 and 3.9.3.1.1 for a description of the drywell and RPV analysis, respectively.) [3.7.11]

Class I structures or equipment founded or supported on other than the ground are analyzed on the basis of their natural frequencies and the predominant frequency of the supporting structure. [3.7.12]

3.7.2.1.1 Reactor-Turbine Building

The reactor building is a reinforced concrete structure from the rock foundation to elevation 690.5 feet and a steel superstructure configuration from elevation 690.5 feet to the roof elevation 737 feet. The turbine building is similar: a reinforced concrete section to elevation 639 feet and a steel superstructure to the roof at elevation 700 feet. The two buildings are connected at the reactor operating floor elevation 639 feet and near the steel-framed roof at elevation 690.5 feet. [3.7.13]

3.7.2.1.1.1 Modeling

The mathematical models depicting this total building system that were used in determination of the dynamic responses are shown on Figure 3.7-4 and 3.7-5 for the north-south and east-west earthquake directional inputs, respectively. The mathematical model of both the turbine and reactor buildings is such that interaction effects of the buildings are also accounted for directly in the computations. Note that the north-south direction model uses only a building connection at elevation 639 feet whereas the east-west direction uses connections at elevations 639 feet and 700 feet. The diesel generator room housing the Unit 1/2 (swing) diesel generator is included in this analysis because the room is rigidly connected to the reactor building east wall through base and roof slabs and two vertical end walls. Since the buildings are founded directly on rock, they are assumed to act with the ground motion at the foundation level. [3.7.14]

These models are equivalent systems of the buildings (including the new fuel storage vault), representing masses interconnected by a system of weightless springs. The masses are lumped at each floor level, at points of intersection of diagonal members in the braced areas, and at the roof diaphragm elevation. Each mass represents the combined mass of the structure and equipment for the specific area represented. Stiffness values are determined from the moments of inertia and effective shear areas by cutting a horizontal section through the building between each mass point and utilizing a modulus of elasticity of 3×10^6 and 30×10^6 psi for concrete and steel, respectively.

The design modulus of 3×10^6 psi is in accordance with the American Concrete Institute (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. 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 significant. Also, the percent 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. [3.7.15]

3.7.2.1.1.2 Analysis

The parameters discussed in Section 3.7.2.1.1.1 were utilized as inputs to a computer program that was designed to solve the dynamic response of structures subjected to arbitrary ground motions. The computer program used in these analyses has been discussed in detail in the Dresden Station Units 2 and 3, AEC Dockets 50-237 and 50-249.

The computer program uses input data in the form of the actual structure inertias, areas, and shear areas and accounts for effects of axial and shear deformations in the calculation of the stiffness matrix. The program then computes natural periods of vibration and mode shapes of the mathematical model by solving for the eigenvalues and eigenvectors utilizing stiffness and mass matrices. Using generalized coordinate and mass matrices, in conjunction with the input ground motion and damping factor for all modes, the generalized displacement response of the structure is computed. [3.7.16]

The Duhamel integral is integrated by a step-by-step solution using a computing interval of 0.005 seconds. Response of each mass for each individual mode at each increment of time is retained in the core storage, and the total response for each time increment is obtained by adding this data together for each mode at each particular instant. This results in an exact combination of mode participation without the necessity of using approximate methods of modal participation.

The results of these computations provide a time-history of displacements and from this, for the total modes considered, a time-history of inertia forces is determined. Once the time-histories of displacement and inertia forces are established, time-histories of shears, moments, and accelerations are computed.

For the original analysis, the input motion used was the first 5 seconds of motion from the time-history of ground accelerations of the Golden Gate Park earthquake. A structural damping level of 5% was used.

The first five modes of system vibration in each direction were used in the analysis. The sixth mode has a period approaching that of a rigid system; therefore, the influence of the sixth and higher modes has a negligible participation in the overall response and was not used. Table 3.7-2 summarizes the modal periods in seconds obtained for both directions of response.

3.7.2.1.1.3 Development of Response Spectra

As previously discussed in Section 3.7.1, the reactor-turbine building was originally analyzed using the Golden Gate Park time-history input but was reanalyzed to consider the Housner time-history input. The design of the reactor-turbine building was performed using the enveloped results of the two analyses. Floor response spectra were developed using the following method. [3.7.17]

The earthquake response of the concrete portion of the reactor building is dominated almost entirely by one mode having a period of vibration of about 0.15 seconds. Because of this fact, the revised floor response spectra were constructed by combining three curves. First, in the period range between 0.0 and about 0.3 seconds, the Golden Gate Park spectrum controls over the Housner spectrum. It can be concluded that, in this period range, the floor response spectra generated from the Golden Gate Park input also control.

Second, for the periods of vibration in the long period range (greater than about one or two seconds), the floor spectra tend to approach the ground spectrum. In this long period range, the accelerations for the Housner spectrum are greater than those of the Golden Gate Park spectrum. In this period range, the floor response spectra generated using the Golden Gate Park input are adjusted to account for the higher accelerations of the Housner Spectrum. [3.7.18]

Finally, in the period range between the point where the previously calculated Golden Gate Park spectrum controls, and where the floor spectra are very nearly represented by the Housner ground spectrum, a transitional curve was established. This transitional curve was constructed by multiplying the ordinates of the Housner spectrum by a magnification factor that was calculated as a function of the ratio of the period under consideration to the period of the input floor motion. For simplicity of analysis, this magnification factor was conservatively calculated by assuming a sine wave acceleration input. Thus the magnification factor presented in standard textbooks on vibrations (for an example, see Figure 5-14 of "Engineering Vibrations," by Jacobsen and Ayre, 1958) can be used as the conservative amplification factor for the transitional portion of the floor response spectra.

These three curves constitute the floor response spectra in the applicable period ranges, and because of the one-mode nature of the concrete portion of the building, these same spectra combinations produce more conservative floor spectra than would be obtained from a multi-mass time-history method using the time history associated with the Housner spectrum. Note that a multi-mass time-history method was used to obtain the results for the Golden Gate Park earthquake.

For the design of equipment and components in the rigid range, the peak accelerations at the appropriate reactor-turbine building elevation can be used. These peak accelerations are graphically depicted in Figures 3.7-6 and 3.7-7. [3.7.19]

3.7.2.1.1.4 Soil-Structure Interaction Considerations

The Quad Cities reactor building is founded on rock. Details of the foundation and its preparation can be found in Reference 1. The earthquake motions imparted to the building through the rock and to the soil overburden through the rock will cause each to vibrate in a particular fashion. Because of the dissimilarity in the properties of the two, a dynamic interaction effect will arise between the building and the soil, causing an effective increase in the static soil pressure on the substructure walls. This increase in pressure has been incorporated for design purposes by the following methods. [3.7.20]

For dry soil, an effective increase in the normal lateral earth pressure was calculated to approximate the effects due to an earthquake. The method utilizes Coulomb's theory for calculating lateral earth pressures but includes the effect of the horizontal acceleration on the weight of the failure wedge of soil behind the wall.

For saturated soil, an effective increase in pore water pressure was calculated by methods utilized for determining the hydrodynamic pressure on dams and added to the results calculated by the above procedure utilizing the buoyant soil weight.

Normal design practice for buildings of this type is to consider a 1,000 lb/ft^2 surcharge around the building for construction loads. The effective increase in lateral earth pressure for the surcharge load was greater than the seismic design pressures calculated, hence, no special considerations for the seismic loadings were necessary.

Random rock voids of up to 20 feet across were assumed under the reactor building. The magnitude of these incompletely-filled cavities in the foundation bedrock were not considered to be sufficient to allow any settling of Class I structures and equipment. However, four bench marks were established on the exterior walls of the Units 1 and 2 reactor building, one each on the concrete pedestals supporting the two drywells, and one on the 310-foot concrete ventilating chimney. Readings were taken during the construction period to monitor settlement and to confirm the original assumption. [3.7.21]

3.7.2.1.2 Control Room

The control room was dynamically analyzed in a manner essentially identical to that of the reactor-turbine building discussed in Section 3.7.2.1.1. One difference in these analyses is that for the north-south direction response, the control room acts as a rigid part of the reactor-turbine building, and, therefore, the results in this direction are based on the reactor-turbine building analyses. The dynamic response developed for the east-west direction is for the control room uncoupled from the reactor-turbine building. [3.7.22]

The other primary analytical difference in the analyses is that three modes of response were considered for the control room in the east-west direction. These first three modes have natural periods of 0.065, 0.027, and 0.017 seconds. Response due to higher modes with shorter periods was neglected, since the participation of such modes has a negligible effect on overall response.

The peak accelerations for the control room are graphically depicted in Figure 3.7-8. These accelerations are used for the design of rigid equipment and components as discussed for the reactor-turbine building in Section 3.7.2.1.1.3.

3.7.2.1.3 Concrete Chimney

The 310-foot concrete chimney serves as the gaseous waste discharge point for both Units 1 and 2. The structure is conical, with a base internal diameter of 22 feet 7 1/2 inches and a top diameter of 11 feet. Thickness varies with the height from 33 to 7 inches. The foundation is filled with sand, and fixity exists about 8 feet below grade. Consideration of the base as fixed results in a shorter period for the stack than if the base were allowed to rotate, and thus the corresponding response acceleration will be higher. Therefore, base rocking and translation were conservatively neglected. [3.7.23]

A dynamic analysis was performed for this structure in the same manner as discussed for the original analysis of the reactor-turbine building, i.e., the chimney was analyzed using the Golden Gate Park earthquake input. The mathematical model utilized treated the chimney as a cantilever structure composed of 33 mass points connected by weightless elastic columns. Seven modes of vibration were considered in the analyses with a 5% damping value assigned to all modes. The periods are 1.381, 0.372, 0.158, 0.089, 0.058, 0.042, and 0.033 seconds, respectively, for these first seven modes. Influence of higher modes is considered negligible to the overall response, and, therefore, higher modes were not utilized in the calculation. Numerical results of this analysis are discussed in Section 3.8.4.3.

3.7.2.1.4 Drywell

John A. Blume and Associates reviewed the Quad Cities drywell with respect to the Dresden drywell including considerations of site geology, input earthquakes, and building arrangements. The review resulted in using the Dresden drywell seismic envelopes of shear, moment, and displacement as design values for Quad Cities. [3.7.24]

The primary factors in the selection of the Dresden drywell analyses as being applicable to Quad Cities are that, below elevation 652 feet 8 inches the shear values for the empty condition are essentially constant and independent of elevation, and the Dresden reactor-turbine building displacements are greater than the corresponding Quad Cities displacements. The drywell seismic inertia contribution is small in comparison to the effects of seismic building movements as indicated from the nearly constant shear values at elevations below 652 feet 8 inches. This indicated that the critical load source is displacement of the reactor-turbine building. The Dresden displacements are 120 and 70 mils in the north-south and east-west direction at the shear lug elevation, whereas the corresponding Quad Cities displacements are 65 and 55 mils. Hence, use of the Dresden design parameters results in a more conservative design than if a detailed drywell analysis for Quad Cities had been used.

The mathematical model of the drywell structure, which includes the reactor-turbine building interaction effects, is depicted on Figures 3.7-9 and 3.7-10.

The drywell shell is connected to the skirt plate by a full penetration weld. The skirt plate is 5/8-inch thick and is anchored into the foundation concrete by twenty-four 2 1/4-inch diameter bolts. On this basis, the drywell was considered fixed at the base. [3.7.25]

A 2-inch gap between the drywell and concrete containment shield wall was ensured by cementing preformed polyurethane foam elastic sheets to the steel drywell and all joints and around the sleeves for the drywell penetrations. The material is compressible, therefore, the drywell is considered to be free except at elevations 652 feet 8 inches and 565 feet 10 inches.

The seismic shear is transferred to the foundation concrete by the skirt plate bearing against the concrete; and therefore, there is no need to rely on friction between the bottom of the drywell shell and the supporting foundation concrete to maintain a stable condition.

The drywell was designed for the effects of seismic loads in a flooded condition. Two conditions were considered critical in evaluating the drywell capability under seismic loading. The first is when the drywell is flooded to the knuckle (elevation 629.9 feet). The second condition is when the drywell is flooded to the normal pool level (elevation 689.5 feet). [3.7.26]

Five percent damping was used in the analysis for all six modes considered. Periods in seconds for both an empty and flooded condition for each direction of applied force is summarized in Table 3.7-3. The difference in mass between the empty and flooded drywell is small; therefore, there is no variation in modal periods for the first five modes. Additional modes appear in the flooded model after the fifth mode. [3.7.27]

3.7.2.1.5 Suppression Chamber

The suppression chamber (torus) is a torus-shaped steel vessel having an inside diameter of 30 feet and a major diameter of 109 feet. It is supported vertically by 32 columns, 16 inner and 16 outer, with saddle supports between each pair of columns. In addition, 26 snubber-type supports located at the ring header provide both vertical and lateral support. Additional lateral stability is provided by four pinned, embedded anchorage assemblies located 90° apart and identified as seismic supports. These supports transmit seismic loads from the torus to the concrete foundation. Dynamically, the torus is a complete system in itself; the vents, headers, and downcomers are separated from the torus by means of bellows which provide no support. [3.7.28]

3.7.2.1.5.1 Original Seismic Analysis

In the original analysis, the torus was idealized as a single degree of freedom system. Its spring constant was determined from the calculated shear deformations of the pins and bottom plates of the four seismic supports. By comparison, the upper plates are rigid in shear, and all plates and pins were considered rigid in bending. The columns contribute a negligible amount of resistance compared to the stiffness of the seismic supports.

The original analysis considered both the flooded and normal conditions. Using the calculated stiffness and mass, the fundamental period of vibration of the torus was determined for the two cases. These periods are quite short in relation to the significant frequencies of the response spectra provided on Figure 3.7-1; therefore, the ground horizontal acceleration of 0.12g was used to determine the stresses induced by the operating basis seismic event.

The original analysis did not include the 16 saddle supports which were added subsequent to the original analysis. [3.7.29]

The mass of the header was assumed to be negligibly small in comparison with the mass of the torus and therefore had no influence on the response of the torus. In considering the amplification factor of 1.5 for the design of the header, it was recognized that the smaller the mass ratio (header to torus), the larger the header response. It was also recognized that the closer the period of vibration of the header is to the period of vibration of one of the torus modes, the larger will be the magnification of response. However, those effects were greatly mitigated by the fact that the resonant mode of the torus is the fourth ovaling mode and therefore transmits very little energy to the header. This fact, coupled with the low stress levels in the header, fully justified the use of the 1.5 magnification factor used in the header analysis. [3.7.30]

The ringheader analysis used a three-dimensional stiffness matrix which included the effects of flexural and torsional shear and axial deformations. Using this stiffness matrix and a mass matrix calculated for the mathematical model, the natural periods of vibration and corresponding mode shapes were determined. [3.7.31]

Since the chamber is very rigid compared to the ring header, the ground response spectrum for 0.5% damping was applicable to the header. Hence, the header was analyzed using a uniform equivalent static coefficient corresponding to the ground spectral acceleration at the header natural period multiplied by a factor of 1.5.

The 1.5 factor provided a 50% margin to account for the effects of higher header modes and the minor magnification that could be produced from torus and header interaction. The fourth ovaling mode of the 30-foot cross section plate is the primary contributing mode to interaction, and the additional 50% margin on the equivalent static coefficient accounted adequately for this effect.

The resulting horizontal coefficient used was 0.4g, which was combined with a simultaneous vertical acceleration of 0.08g.

Vents and downcomers are isolated from the torus dynamically and are attached to the drywell. Downcomer design is controlled by jet forces which are much greater than seismic loads. Vents with their bellows are designed to take movements due to temperature, pressure, and earthquake loading conditions. The bellows have the capability of taking axial and lateral movements resulting from these conditions. [3.7.32]

3.7.2.1.5.2 Mark I Program

After the original seismic analysis described in Section 3.7.2.1.5 was performed, hydrodynamic loads associated with the Mark I Program were identified. These new loads necessitated a re-evaluation of the torus and associated subsystems. Seismic analyses of the torus, performed as part of the Mark I Program are discussed in Section 3.8.2. [3.7.33]

3.7.2.1.6 Reactor Vessel Support System

In order to ascertain the adequacy of the reactor vessel support structure, a dynamic analysis was performed to determine shear, moments, displacements, and forces on the supporting structure. The system configuration which was analyzed essentially consists of a fixed base at the pedestal, connected through the support structure to the vessel. The vessel is connected to the shield wall by stabilizers and the shield wall is interconnected to the drywell by a truss arrangement. The upper supports are located at elevation 653 feet approximately. The pedestal is a cylinder carrying the reactor, reactor shield wall, floor framing, piping, and equipment with these vertical loads being transferred through bearing to the drywell base and to the rock foundation. Lateral load due to seismic action is transferred through the pedestal base keyway in the drywell interior concrete. [3.7.34]

The amount of frictional force available to resist horizontal shear is directly proportional to the normal pressure (proof load) between the RPV skirt flange and top flange of the ring girder. The frictional force, both in its total amount and its coefficient of sliding friction, is independent of the areas in contact, so long as the total pressure remains the same. The friction-type connection of the RPV skirt flange to the ring girder, in which some of the bolts lose a part of their clamping force (proof load) due to applied tension during an earthquake, suffer no overall loss of frictional shear resistance. The bolt tension produced by the moment is coupled with a compensating compressive force on the other side of the axis of bending. [3.7.35]

As part of the core shroud repair program, new horizontal seismic analyses of Quad Cities Units 1 and 2 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 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. Both Unit 1 and Unit 2 have the core shroud repair hardware installed. [3.7.36]

The seismic analyses were performed using the time history method. The input motions included the S80E component of the 1957 Golden Gate Park earthquake record and a synthetic time history record enveloping the Housner spectra. Both time histories were normalized to a peak ground motion of 0.24g. Bounding Design Basis Earthquake (DBE) and Operating 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 re-evaluate the adequacy of the reactor pressure vessel support system.

3.7.2.2 Class II Structures and Systems

For Class II structures, normal design practices for the design of power plants in the State of Illinois have been utilized. As a minimum, however, seismic inputs were in accordance with the Uniform Building Code (UBC) recommendations for Zone I and the allowable stresses were in accordance with the UBC. [3.7.37]

Detailed information related to the site geology and seismic probabilities which demonstrate the conservatism inherent in the Class II design criteria and methods can be found in the previously referenced PDAR, Appendix F. Examples of Class II design practice are the analysis of the radwaste and crib house structures.

The crib house walls were investigated for lateral earth effects due to both the OBE and DBE. The details of this investigation are discussed in Section 3.8.6. [3.7.38]

3.7.3 Seismic Subsystem Analysis

This section provides an overview of the methods used to perform seismic subsystem analyses at Quad Cities. Subsections 3.7.3.1 through 3.7.3.3 provide modeling and analysis details for piping systems, equipment, components, and minor structures at Quad Cities.

3.7.3.1 Piping

Originally, Class I piping systems were analyzed by either dynamic modal computer analysis or by the use of force-deflection curves, as discussed more fully in Section 3.9.3. Modal analysis was performed for piping larger than 8 inches in diameter, and force-deflection curves were applied to piping 8 inches and smaller in diameter. [3.7.39]

Subsequent to the original analyses, two piping-related programs, which involved reanalysis of Class I piping systems, were completed. The Mark I Program (described in Sections 3.8, 3.9 and 6.2) included re-evaluation of piping attached to the suppression chamber and the safety relief valve discharge piping which were impacted by Mark I hydrodynamic loads. [3.7.40]

In addition, the release of IE Bulletin 79-14 (described in Section 3.9.3.1) prompted the re-evaluation of safety-related piping larger than 2 in. in diameter.

Most of the piping affected by either the 79-14 or Mark I Program was reanalyzed by rigorous computer methods, which is described in Section 3.9.3.

3.7.3.2 Equipment

Generally, Class I equipment is seismically qualified by testing or analysis. The qualification of Class I instrumentation and electrical equipment is addressed in Section 3.10. The qualification of Class I mechanical equipment is addressed in Section 3.9. [3.7.41]

3.7.3.3 Other Structural Elements

This section describes methods used to analyze and evaluate other structural elements not included in the system analysis discussions in Section 3.7.2. Among these structural elements are small buildings, building elements within larger structures, and miscellaneous structures such as retaining walls and tunnels.

3.7.3.3.1 Fuel Pool

The seismic analyses of the fuel pool space were included in the overall reactor-turbine building analyses discussed in Section 3.7.2.1.1. The analysis models for the reactor-turbine building, shown in Figures 3.7-4 and 3.7-5, included the mass of the fuel pool water and the fuel pool concrete in mass points 4 and 5. [3.7.42]

3.7.3.3.2 Class I Tunnels

The Class I tunnels are supported on concrete down to the rock strata. These tunnels are a part of the underground concrete structure and act with the reactor and turbine buildings as a unit. Any underground Class I pipe or cable carried in these tunnels are evaluated using applicable spectra from the reactor-turbine building. [3.7.43]

Similarly, the service water pipe from the crib house to the reactor and turbine buildings is encased in the circulating water pipe concrete encasement which is sitting directly on the rock. In view of the fact that the crib house is also sitting on the rock, the reactor-turbine building, crib house, and concrete encasement also act as a unit.

3.7.3.3.3 Intake Flume Retaining Wall Structure

The dynamic loading on the retaining wall is the combination of soil and water pressures during the earthquake. The resultant dynamic force due to soil was obtained by a method developed by Mononobe and Okabe, (Mononobe, N., and Matsuo, H., "On the Determination of Earth Pressure During Earthquakes," Proceedings, World Engineering Congress, Tokyo, 1929). The dynamic water pressures on both sides of the retaining wall were obtained by using a modification of the Westergaard Theory (Matsuo, H., and O'Hara, S., "Lateral Earth Pressure and Stability of Quay Walls During Earthquakes," Proceeding, Second World Conference on Earthquakes, Engineering, Vol. I, 1960). On the seaward side of the retaining wall, dynamic water pressure is equal to that computed by the Westergaard solution. On the landward side, the dynamic water pressure due to seepage was taken as 70% of the Westergaard value and acts together with the dynamic water pressure acting on the seaward side of the wall. The resulting values of the combined dynamic effects of soil and water pressures were added to the static pressures to obtain the maximum lateral forces. The retaining wall was analyzed for both an OBE and a DBE. The resulting stresses were below the allowables for both cases. [3.7.44]

3.7.3.3.4 Intake Flume Earth Embankment

The seismic consideration on earth embankment is primarily its resistance to sliding motion during an earthquake. The analysis of the slope stability of the earth embankment during an earthquake was proposed by Newmark (Newmark, N.M., "Effects of Earthquakes on Dams and Embankments," Goetechnique, Vol. XV, No. 2, June, 1965).

Three principal cases of sliding are considered:

1. Circular sliding surface,

2. Plane sliding surface, and

3. Block sliding horizontally.

The earth embankment was found to be capable of resisting the sliding effects during an SSE for all three cases considered.

3.7.3.3.5 Masonry Walls

Masonry walls are utilized at Quad Cities as firewalls, partition walls, radiation shielding, opening blockouts, and exterior walls. All masonry walls at Quad Cities 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 IEB 80-11. [3.7.45]

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 DBE. For the generation of attachment loads, the following damping levels are used for piping: 0.5% for OBE and 2.0% for DBE, and for other systems, 1.0% for OBE and 2.0% for DBE.
Equivalent static analyses are performed for the inertial considerations. The results of one horizontal 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 Quad Cities is 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 DBE. Movements are evaluated to ensure that safety-related items are not affected. Mortar-free cantilever walls are evaluated using a coefficient of friction of 0.33, to maintain a factor of safety of 1.5 for OBE and 1.0 for DBE.

3.7.4 Seismic Instrumentation

A strong-motion seismograph is located on the basement floor of the Unit 1 turbine building (elevation 547 feet 0 inches) in the northeast corner of the condensate pump room. The unit is mounted directly onto the floor in a corner out of the way of the normal traffic pattern. By mounting the seismograph directly on the floor of the turbine building basement, an accurate recording of ground motion originated in the rock substructure which underlies the plant foundations will be obtained. [3.7.46]

The unit is self-contained and is powered by rechargeable batteries.

This single seismograph used in the Quad Cities plant is not installed for safety purposes but rather to record earthquake forces occurring at the site. The seismograph was installed in the Quad Cities plant at the request of the AEC, for the purposes of history and as an aid in evaluating local earthquakes. The seismograph serves as an earthquake assessment resource for Quad Cities' emergency planning. The Exelon Nuclear Standardized Radiological Emergency Plan (E-Plan) for Quad Cities defines emergency action levels, depending upon the earthquake magnitude recorded by the seismograph. [3.7.47]

If a seismic disturbance occurred in the range of and less than or equal to the OBE, a thorough visual inspection would be made of the plant equipment and instrumentation to check for any abnormalities. If conditions were found to be normal, plant operation would be continued or resumed. If the seismic disturbance exceeded the OBE, a thorough visual inspection would be made of the plant equipment and instrumentation to check for any abnormalities and the plant would be shut down. [3.7.48]

3.7.5 References

1. Quad Cities Station Units 1 and 2 Foundation Grouting Report (including Supplementary Information).

Table 3.7-1

VIBRATION DAMPING FACTORS FOR STRUCTURES AND ASSEMBLIES

Item	% of Critical Damping
Reinforced Concrete Structures	5.0
Steel Frame Structures	2.0
Bolted and Riveted Assemblies	2.0
Welded Assemblies	1.0
Vital Piping Systems	0.5
Standby Gas Treatment System Duct	1.0
[1]Reactor Pressure Vessel	3.0
Masonry Walls	2.0

[1] The damping factors for the shroud, guide tubes, CRD housing, and RPV stabilizer are considered as GENE proprietary information and are provided in GENE-771-71-1094, Revision 1, "Quad Cities Units 1 and 2 Shroud Repair Seismic Analysis," January 5, 1995.

Table 3.7-2

REACTOR-TURBINE BUILDING MODAL PERIODS (SEC.)

Direction	1st Mode	2nd Mode	3rd Mode	4th Mode	5th Mode
N-S	0.345	0.306	0.145	0.100	0.093
E-W	0.331	0.177	0.195	0.113	0.082

Table 3.7-3

DRYWELL MODAL PERIODS (SEC.)

Direction/Condition Mode:	1	2	3	4	5	6
N-S/Empty	0.39	0.23	0.20	0.17	0.14	0.077
N-S/Flooded	0.39	0.23	0.20	0.17	0.14	0.12
E-W/Empty	0.37	0.34	0.25	0.15	0.12	0.65
E-W/Flooded	0.37	0.34	0.25	0.15	0.12	0.116

3.8 DESIGN OF CATEGORY I STRUCTURES

3.8.1 Concrete Containment

The primary containment function for the Quad Cities Mark I design is provided by three interconnected steel structures: the drywell, vent system and pressure suppression chamber (torus or wetwell). 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, therefore, addressed in Section 3.8.4. Functional design of both primary and secondary containment is covered in Section 6.2. [3.8.1]

3.8.2 Steel Containment

This subsection presents key aspects of the structural design of the Quad Cities primary steel containment. Design elements addressed for the major steel containment components include:

A. Physical layout,

B. Codes and standards,

C. Loading conditions,

D. Design and analysis procedures,

E. Acceptance criteria, 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). [3.8.2]

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 Figure 6.2-1).

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 steel suppression chamber is mounted on support structures which transmit loads to the concrete foundation of the reactor building (see Figures 6.2-1, 6.2-2, and 6.2-5).

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 air space. 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, as illustrated by Figures 6.2-1, 6.2-3, and 6.2-4.

The original design of the Mark I containment system considered postulated accident loads associated with the containment design (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 is presented in Section 6.2.1. Because these hydrodynamic loads had not been considered in the original design of the containment, a detailed re-evaluation was undertaken. This re-evaluation, referred to as the Mark I Program, involved tasks performed to restore the originally intended design safety margins for the Quad Cities plant. The Mark I Program culminated in the issuance of the plant unique analysis report (PUAR)[1] for Quad Cities 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. Due to the proximity of this anchorage to the vent/drywell junctions, no significant Mark I hydrodynamic loads propagate to the drywell. Therefore, with the exception of the main vent junctions, the drywell was not re-evaluated 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. [3.8.3]

The drywell spherical section is 66 feet in diameter and varies in thickness from 11/16 inches — 1-1/8 inches. The cylindrical neck section is 37 feet in diameter and varies in thickness from 3/4 inches — 1-1/2 inches. The spherical to cylindrical transition is 2-3/4 inches thick. The removable top head ranges from 1-1/4 inches — 1-7/16 inches in thickness. The drywell stands 111 feet 11 inches tall. Drywell materials are described in Table 6.2-1.

As noted in the arrangement drawings the drywell bottom is filled with concrete. Beneath the drywell is a 26-foot thick concrete fill from the spring line down. These concrete fills are in contact with an internal 6x1-inch continuous steel ring on the interior and the steel support skirt on the exterior. These shear rings, attached to the drywell at approximately a 19-foot 6-inch radius, 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 652 feet 8 inches. 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. [3.8.4]

3.8.2.1.2 Applicable Codes, Standards, and Specifications

The primary containments for Quad Cities Station Units 1 and 2 were designed, erected, pressure-tested, and N-stamped in accordance with the ASME Code, Section III, 1965 Edition including Addenda up to and including Winter 1965. [3.8.5]

3.8.2.1.3 Loads and Load Combinations

The loads applicable to the design of the drywell are defined as follows: [3.8.6]

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.

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 loss-of-coolant accident.

E = Design earthquake load, ground acceleration, horizontal =0.12g. vertical =0.08g.

E' = Maximum earthquake load, ground acceleration, horizontal =0.24g vertical =0.16g.

The load combinations applicable to the design of the drywell are defined as follows: [3.8.7]

Drywell Load Combinations (Including Penetrations)

Load Combination	Allowable Stress
D + H + T + E	ASME, Section III, Class B, without the usual increase for seismic loadings.
D + P + H + R + T + E	Same as the preceding, except that local yielding is permitted in the area of the jet force where the shell is backed up by concrete. In area not backed up by concrete, primary local membrane stresses at the jet force do not exceed 0.90 times yield point of the material at 300°F.
D + P + H + R + T + E'	Primary membrane stresses, in general, do not exceed the yield point of the material. The same criteria as above is applied to the effect of jet forces for this loading condition.

The drywell design pressure for Quad Cities is 56 psig whereas the design pressure for the Dresden Units is 62 psig. The ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Vessels" was changed by the Winter 1965 Addenda and is the reason for the apparent (but not actual) differences in the design pressure. The Winter 1965 Addenda was issued December 31, 1965 and became mandatory on July 1, 1966, which made it applicable to Quad-Cities 1 and 2 containments. [3.8.8]

Paragraph N-1312 of Section III states that the "design internal pressure" may differ from the "maximum containment pressure," provided the "design internal pressure" is not less than 90% of the maximum containment pressure. Thus, the specification for the Quad Cities containments specify a "maximum internal pressure" of 62 psig and a "design internal pressure" of 56 psig. Paragraph N-1314(b) states that the maximum allowable stress, S_{m}, shall not exceed the allowable stress in Section VIII of the Code.

Prior to the Winter 1965 Addenda, the allowable stress, S_{m}, was specified as 1.1 times the Section VIII allowables; therefore, the basic required shell thickness is essentially the same for containment vessels constructed before and after the Winter 1965 Addenda.

Prior to the Winter 1965 Addenda, the containment vessels were pneumatically over-pressure tested at 115% of the design pressure (1.15 x 62 = 71.3 psig in the case of Dresden) instead of 125% of the design pressure, as required by Section VIII. The Winter Addenda now requires a test in accordance with the rules of Section VIII and does not allow a reduction in the percent over-pressure (1.25 x 56 = 70 psig in the case of Quad Cities).

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

Section 6.2.1 addresses the drywell performance capabilities. With regard to the seismic structural design evaluation (See Section 3.7), Figures 3.8-2 through 3.8-4 present the OBE drywell results for displacements, shears, and moments, respectively, in the north-south direction. Results for displacements, shears, and moments in the east-west direction are presented in Figures 3.8-5 through 3.8-7, respectively.

Based upon a number of drywell analyses, the empty condition is more controlling than the flooded condition for seismic design. The flooded condition results in tensile stresses predominating, whereas the empty condition is controlled by compressive (buckling) stresses. Since the critical buckling allowable stresses are less than the tensile allowable stresses, the empty condition is considered to control. [3.8.9]

The bearing stresses on concrete due to shear from the OBE and DBE are 67 psi and 134 psi, respectively. The tensile stresses in the skirt plate from OBE and DBE are 1030 psi and 2060 psi, respectively. The tensile stresses in the anchor bolts connecting the skirt plate to the foundation due to the OBE and DBE are 9800 psi and 19,600 psi, respectively. The anchor bolts are embedded in the foundation concrete with a bearing plate on the end of the bolt a distance sufficient to develop the full tensile strength of the bolt. [3.8.10]

The drywell plates are restrained in the area of the skirt connection by the concrete inside and outside the drywell. Therefore this plate is subjected to membrane stresses only. For the OBE this stress is 2750 psi and for a DBE the stress is 5300 psi.

The allowable stress for the skirt plate material is 22,500 psi at 300°F and the allowable stress for the bolt material is 31,400 psi at 300°F. The allowable bearing stress in concrete is 750 psi.

In addition to using the conservative design values presented, no increase in allowable stresses for short term loading was used. Further, the design was reviewed for twice the presented loads to assure compliance with the design basis earthquake criteria. [3.8.11]

The drywell was designed for the additional effects due to seismic loads when it is already in a flooded condition. Two conditions were considered critical in evaluating the drywell capability under seismic loading. The first is when the drywell is flooded to the knuckle (elevation 629.9 feet). The second condition is when the drywell is flooded to the normal pool level (elevation 689.5 feet). In both cases, the critical buckling stresses are at the embedment of the drywell at elevation 576 feet. Stresses in the drywell under the combination of a flooded post-accident condition and either an OBE or a DBE remain below critical buckling stresses. [3.8.12]

Table 3.8-1 summarizes the drywell stresses at this point of embedment in the concrete (elevation 576 feet).

3.8.2.1.6 Testing and Inservice Inspection Requirements

Pressure and leak rate testing of the containment system is addressed in Section 6.2.6.

During leak rate testing, the Quad Cities-Unit 1 vessel was supported on the concrete fill in contact with the entire base. The Quad Cities Unit 2 vessel was supported by the [3.8.13] exterior steel support erection ring for the testing. The drywell was designed for the weight to be supported in this manner without creating local overstress conditions. The interior 1x6-inch ring provides additional strength in this area, and acting with the support ring aids in preventing overstress. These support methods are permanent and the test is representative of the operating condition.

3.8.2.1.7 Flued Head Penetrations

Piping penetrations are of two general types: i.e., those which accommodate movement, and those which experience relatively little movement. [3.8.14]

An example of a piping penetration for which movement provisions are made is shown in Figure 3.8-37. These penetrations have a guard pipe between the hot line and the penetration nozzle in addition to a double-seal arrangement. This permits the penetration to be vented to the drywell should a rupture of the process 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 one-piece forging with integral flues or nozzles. These fittings were designed to the ASME Pressure Vessel Code Section VIII. The penetration sleeve is welded to the drywell and extends through the 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 drywell relative to the process pipe. A double bellows arrangement permits remote leak testing of the penetration seal. A single or double ply bellows with a permanent bellows test enclosure or a temporary test fixture inside the drywell will also permit leak rate testing. The lines have been constrained at each end of the penetration assembly to limit the movement of the line relative to the containment, yet permit pipe movement parallel to the penetration.

Qualification tests have been conducted on a Pathway Bellows 26-inch Tandem Expansion Joint, which is virtually identical to the Quad Cities bellows. The object of these tests was to qualify the 26-inch expansion joint by flexing in a lateral offset condition and performing leakage and proof pressure tests to verify design integrity. The test specimen successfully passed all test conditions with no indication of failure due to life cycling, distortion due to applied proof pressure, or leakage of the inner and/or outer ply.

Lines which connect to a high-pressure system and do not have a double-seal penetration sleeve are the hydraulic lines to the control rod drives. The control rod drive penetrations consist of 362 small diameter stainless steel pipes shop welded to the drywell plate, of which 354 are used for control rod drive lines. 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 leak rate test.

Penetration details of piping lines that allow for relatively little movement are shown in Figure 3.8-38. The pipe sleeve which attaches to the drywell is designed for 56 psig, but because of structural thicknesses, can withstand a substantially higher pressure. No bellows are required, since drywell thermal expansion is minimal. A tabulation of the type of penetration used for each service is shown in Table 3.8-2.

The drywell is reinforced at penetrations by means of inserts heavier than the shell material as shown in UFSAR Figure 3.8-38 which provide the necessary material to maintain plate stresses within allowable limits. Reinforcing requirements are in accordance with the provisions of Sections III and VIII of the ASME Boiler Code for Class B Vessels.

Lines which open directly to the containment do not have separate penetration sleeves and are welded directly to the containment shell. The drywell shell is reinforced at these penetrations by means of inserts heavier than that shown in the schematic of Figure 3.8-38 which provide the necessary material to maintain plate stresses within allowable limits. Reinforcing requirements are in accordance with the provisions of Sections III and VIII of the ASME Boiler Code for Class B Vessels.

3.8.2.1.8 Electrical Penetrations

Electrical penetrations 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 elements shown in Figure 3.8-39. [3.8.15]

An assembly is sized to be inserted in and welded to the 12-inch schedule 80 penetration nozzles which are furnished as part of the containment structure.

Hermetic seals are provided at each end of the penetration, 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 the ASME Boiler and Pressure Code, Section III, Class B Vessel, 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 loss of coolant accident, and to maintain containment integrity for extended periods of time in a post-accident environment. These conditions, including the normal operating environmental condition, are shown in Table 3.8-3.

The low voltage assembly is suitable for voltages of 600 V or less and is designed for conductors varying in size from 18 — 4/0 AWG. The cables are grouped and passed through openings in the header plates in the same manner as shown in the generic representation given by Figures 3.8-40 and 3.8-43. A sealing compound is applied at each end of the penetration to seal the assemblies. A cable lead is terminated at either a splice or an environmental-resistant connector.

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-

41 shows a cutaway view of the containment penetration assembly for shielded signal cables. One type of circuit uses coaxial connectors mounted directly on the headerplates and is isolated from ground. Another type of circuit uses connectors mounted on the penetration assembly auxiliary structure.

A sectional view of the high voltage power cable penetration assembly is shown on Figure 3.8-42. 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 nonmagnetic stainless steel in order to eliminate the possibility of eddy current heating.

3.8.2.2 Vent System

3.8.2.2.1 Description of Structure

The Quad Cities Units 1 and 2 vent systems 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 3.8-8, 3.8-9, 6.2-4 and 6.2-5 show the proximity of the vent systems to other containment components. [3.8.16]

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 spherical transition segments at the penetration to the drywell (Figure 3.8-14). The drywell shell around each vent line-drywell penetration is 1-1/8-inches thick and is reinforced with a 1-1/2-inch thick reinforcing pad plate and a 3-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 3.8-15). Each spherical junction is constructed from six shell segments with thicknesses varying from 1/4 — 5/8 inch. 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 (SRVDL) are routed from the drywell through the vent line and penetrate the vent line inside the suppression chamber. Section 3.9.3.1 provides a discussion of the analysis of SRV 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 1/4-inch thick and has an ID of 4 feet 10 inches.

Ninety-six downcomers penetrate the vent header in pairs (Figures 3.8-16, 6.2-1, and 6.2-3). Two downcomer pairs are located in each vent line bay; four pairs are located in each non-vent 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 3.8-16, 3.8-17, and 3.8-18). The inclined segment is 3/8-inch thick and the vertical segment is 1/4-inch thick. The inside diameters of the inclined and vertical portions of the downcomer are 2 feet 0 inches and 2 feet 1/8 inches, respectively.

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 3.8-16, 3.8-17, and 3.8-18). In the plane of the downcomer pairs, the intersections are stiffened by a pair of 1/2-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 located in those vent line bays which house the SRV discharge line, and which extend to midlength of the neighboring non-vent bays (Figure 3.8-16). In this manner, 62% of all the downcomers are braced longitudinally. The longitudinal bracing patterns for the two Quad Cities units vary in some degrees because of the different locations of the SRV lines (Figures 3.8-16, 3.8-19, and 3.8-20). The ends of the horizontal pipe members near miter joints and centerlines of the non-vent bays are welded to the downcomer rings. The 3 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 (Figures 3.8-14 and 6.2-2). 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 1-3/4-inch thick nozzle. The bellows unit has a 7-foot 5-inch inside diameter and contains five convolutions which connect to a 1/2-inch thick cylindrical sleeve at the vent line and a 1-inch thick cylindrical sleeve at the torus nozzle end. A 1-1/2-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 1-3/4-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 2-3/4 inches.

Vent header deflectors are provided in both the vent line bays and the non-vent line bays (Figures 3.8-9, 3.8-17, and 3.8-18). The deflectors shield the vent header from pool swell impact loads which occur during the initial phase of a DBA 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 and extend from mounting flanges attached to 1-foot 7-inch outside diameter (OD) by 1/2-inch thick nozzles. The nozzles penetrate the vent line-vent header spherical junction (Figure 3.8-15).

The vent system is supported vertically by two column members at each miter joint location (Figures 3.8-21, 3.8-22, and 6.2-5). 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 2-3/4-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 3/4-inch thick pin plates. The lower ends of support columns are attached to 1-1/2-inch thick ring girder pin plates with 2-3/4-inch diameter pins and 3/4-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 SRV piping inside the vent line and suppression chamber (Figure 3.8-14, and 6.2-4). Loads acting on the SRV 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 SRV piping at the same support locations. Since the relative stiffness of the SRV discharge line with respect to other vent system components is small, the support provided by the SRV discharge line to the vent system is negligible.

3.8.2.2.2 Applicable Codes, Standards, and Specifications

The primary containments, including the vent systems described herein, for Quad Cities Units 1 and 2 were originally designed, erected, pressure-tested, and N-stamped in accordance with the ASME Code, Section III, 1965 Edition with Addenda up to and including Winter 1965. [3.8.17]

For the Mark I Program re-evaluation, the acceptance criteria generally followed the rules contained in the ASME Code, Section III, 1977 Edition with Addenda up to and including Summer 1977 for Class MC (Metal Containment) components and component supports. Further detail regarding structural acceptance criteria may be found in Section 3.8.2.2.5.

3.8.2.2.3 Loads and Load Combinations

The loads for which the Quad Cities Units 1 and 2 vent systems were 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 [4] (PULD) for Quad Cities. The PULD essentially implemented the methodologies defined in NUREG-0661. [3.8.18]

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. Safety relief valve 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 were 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 were fully evaluated and are described in the PUAR[1].

3.8.2.2.4 Design and Analysis Procedure

With the exception of the nonrepetitive pattern of the downcomer longitudinal bracing system, 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 non-vent line bay (Figure 6.2-3). To account for the nonrepetitive pattern of the longitudinal bracing system, 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 Quad Cities vent systems. The governing loads and a few chugging load cases, exhibit symmetric or anti-symmetric characteristics (or both) with respect to a 1/16 segment of the vent system. The analysis of the vent system for the majority of the governing loads was therefore performed for the two 1/16 segments described previously. [3.8.19]

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 vacuum breakers, 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 3.8-14 through 3.8-18) 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, determination of 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 the latter, 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 Quad Cities Unit 2 downcomer longitudinal bracing configuration (Figure 3.8-20) was used to obtain the response of the vent system to asymmetric loads. The Quad Cities 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 67.5° counter-clockwise rotation from true north (Figure 3.8-20). The model includes 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 assumed 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 the 180° beam model 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 Quad Cities 1 and 2 vent system analysis is based follow the rules contained in 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].[3.8-20]

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 vacuum breaker nozzles, 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, the vacuum breaker nozzles, 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 284°F, which exceeds the maximum DBA temperature of 281°F. Table 3.8-4 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 analysis described in Section 3.9.3.1.3.3.2.

As permitted in ASME Code Subsection NCA, Class 1 piping rules were employed in the stress analysis of this section of the SRV discharge line. 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 3.8-5.

Table 3.8-6 shows the maximum stresses and associated design margins for the major vent system components, component supports, and welds for the controlling load combinations.

Table 3.8-7 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 leak rate testing of the containment system is addressed in Section 6.2.6.

3.8.2.3 Suppression Chamber

3.8.2.3.1 Description of Structure

The Quad Cities Units 1 and 2 suppression chambers are constructed from 16 mitered cylindrical shell segments joined together in the shape of a torus. Figure 6.2-3 illustrates the configuration of each suppression chamber. Figures 3.8-8 through 3.8-10 show the proximity of the suppression chamber to other components of the containment. [3.8.21]

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 feet 6 inches, measured at midbay of each mitered cylinder. The ID of the mitered cylinders which make up the suppression chamber is 30 feet 0 inches. The suppression chamber shell thickness is typically 0.582-inches above the horizontal centerline, and 0.649-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, 3.8-10 and 3.8-12). A typical ring girder is located in a plane 4-inches from the miter joint and on the non-vent 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 13 feet 2-1/2 inches. Thus the ring girder web depth varies from 20 — 23-7/8-inches and has a constant thickness of 1-1/2 inches. The upper and lower portions of the ring girders are attached to the suppression chamber shell with 5/16-inch fillet welds (Figures 3.8-11 and 3.8-12).

The ring girders are laterally reinforced at the base of the vent header support columns by 1-inch thick plate assemblies (Figure 3.8-12). There are five such assemblies in the bays with SRV discharge lines in both units. In the non-SRV discharge line bays, there are no such assemblies in Unit 1, and two in Unit 2. In addition to these lateral stiffeners, the ring-girder-web plate-to-torus-shell fillet weld was increased from 5/16 — 5/8-inch over a 12-foot 0-inch long arc near the outside torus support column (Figure 3.8-10).

The suppression chamber is supported vertically at each miter joint by inside and outside columns and by a saddle support which spans the inside and outside columns (Figures 6.2-5, 3.8-10, and 3.8-11). The columns and column connection plate webs are perpendicular to the longitudinal centerline of the suppression chamber. 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 is geometrically continuous over the lower half of the suppression chamber. 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 wide-flange sections constructed from a 1-inch thick web plate with 1-1/4-inch thick flanges (Figure 3.8-11). The column base plate assemblies consist of a 2-7/8-inch thick base plate, a 1/2-inch thick lubrite plate, and a 3-1/8-inch bearing plate (Figure 3.8-11). The lubrite pad allows gross torus thermal growth in the radial direction to reduce stresses due to uniform thermal loads.

The connection of the column supports to the suppression chamber shell consists of the column web and flanges, 1-inch thick stiffener plates, and 1-1/4-inch thick column patch plates (Figure 3.8-12).

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 1-1/4-inch thick web plate, a 1-1/4-inch thick lower flange plate and saddle base plate assemblies (Figures 3.8-10 and 3.8-11). The saddle base plate assemblies consist of a 2-7/8-inch thick base plate, a 1/2-inch thick lubrite plate, and a 1-1/2-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 3/4-inch thick stiffener plates to ensure that buckling does not occur during peak loading conditions.

The anchorage of the suppression chamber to the basemat consists of eight, 1-3/4-inch diameter, epoxy-grouted anchor bolts provided at each saddle base plate location. The bolts are anchored through a 3-13/16-inch long slotted hole in the base plate to allow for thermal growth. A total of 16 anchor bolts at each miter joint provides the principal mechanism for transfer of uplift loads on the suppression chamber to the basemat.

Four seismic restraints, which provide lateral support for the suppression chamber, are located 90° apart (Figure 6.2-3). Each seismic restraint consists of a 2-inch thick pad plate welded to the bottom of the suppression chamber shell, a system of interlaced vertical gusset plates joined by a 7-inch diameter pin, and a 2-inch thick base plate with shear bars keyed and grouted into the basemat (Figure 3.8-13). The seismic restraints permit vertical and radial movement of the suppression chamber, while restraining longitudinal movement resulting from lateral loads acting on the suppression chamber. The pad plates distribute loads over a large area of the suppression chamber shell and provide an effective means of transferring suppression chamber lateral loads to the basemat.

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, emergency core cooling system ring header, 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.

To clarify the relation of the torus and the ring header, Figures 6.2-5, 3.8-23 and 3.8-24 are provided. Figure 6.2-5 is a section view of the torus showing the relation of the ECCS header to the torus. Figure 3.8-23 is a plan of the ring header which is attached to the torus by the 4 pipe tee connections and the 26 snubber supports located as shown. Figure 3.8-24 provides elevation views of the tee pipe and the strut connections. [3.8.22]

The 24-inch ECCS suction header encircles the torus and provides a manifold for the suction of various ECCS pumps. The location of the ECCS suction strainer is shown on Figure 3.8-24, Section A-A. The header is connected to the torus by four 20-inch outside diameter (OD) pipes spaced 90 degrees apart and was originally supported by twelve hangers which were connected to the torus shell. It has been shown that there was no evidence of excessive stressing of the torus shell or the ring header as a result of the hanger failures during start-up testing on May 28, 1972 as documented in special report No. 5 to Quad Cities Station, Torus Ring Header Support Failure. The ECCS torus ring header was re-evaluated under the Mark I Program, as discussed in Section 3.9.3. [3.8.23]

3.8.2.3.2 Applicable Codes, Standards, and Specifications

The primary containments, including the suppression chamber described herein, for Quad Cities Units 1 and 2 were originally designed, erected, pressure-tested, and N-stamped in accordance with the ASME Code, Section III, 1965 Edition with Addenda up to and including Winter 1965. [3.8.24]

For the Mark I Program re-evaluation, the acceptance criteria generally follow the rules contained in 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.

3.8.2.3.3 Loads and Load Combinations

The loads acting on the suppression chamber are categorized as follows: [3.8.25]

A. Dead weight loads,

B. Seismic loads,

C. Pressure and temperature loads,

D. Pool swell loads,

E. Condensation oscillation loads,

F. Chugging loads,

G. Safety relief valve discharge loads, and

H. 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] were 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 were 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 (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 Quad Cities have five ring girder stiffeners in the SRV bays (Figure 6.2-5); however, they differ in the number of ring girder stiffeners in the non-SRV bays. Unit 1 has zero; Unit 2 has two. Two analytical models were generated to address the submerged structure loads, one each for the SRV and non-SRV 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-SRV 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 3-1/2-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 — 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, as described in Section 3.8.2.3.3. 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 maximum total lateral load was conservatively assumed to be aligned about a principal suppression chamber azimuth (Figure 6.2-3) and transferred equally by two of the four seismic restraints. Once the seismic restraint loads were known, these values were compared with the allowable seismic restraint loads.

Loads on the seismic restraints result in a shear force and bending moment acting on the suppression chamber shell because of the eccentricity of the seismic restraint pin with respect to the shell middle surface. The effects of these shears and moments on the suppression chamber shell were evaluated using the analytical model of the suppression chamber described earlier. A distribution of forces which produce the desired shear and moment was applied to the suppression chamber shell at the perimeter of the seismic restraint pad plate. 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 Quad Cities Units 1 and 2 suppression chamber analyses are based follow the rules contained in 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 the maximum IBA temperature of 164°F. The allowable stresses for the vertical support system base plate assemblies were determined at 100°F. Table 3.8-8 shows the resulting allowable stresses for the load combinations with ASME Code Service Level B, C, and D limits.

Table 3.8-9 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.

The torus and its supports were also originally designed for a basic seismic acceleration of 0.3g compared to the DBE value of .24g as assigned to this site. It was subjected to this high acceleration value in combination with the effects of the drywell being flooded to elevation 689.5 feet.

Table 3.8-10 summarizes the actual and allowable stresses due to a flooded drywell and a 0.3g seismic load.

3.8.2.3.6 Testing and Inservice Inspection Requirements

Pressure and leak rate testing of the containment system is addressed in Section 6.2.6.

It should be noted that leak-testing of the torus was done with the torus structurally supported by the existing vertical columns. Temporary supports or structures were not required or utilized for the test procedure.

3.8.3 Internal Structures of Steel Containment

Class I structures located within the primary containment include the reactor concrete pedestal and the concrete bioshield wall. Structural evaluation for the reactor pedestal and bioshield wall are addressed in Sections 3.7.2 and 3.9.3.

3.8.4 Other Seismic Category I Structures

The major structures covered in this section are the reactor, turbine, control room, and the 310-foot chimney. The reactor and turbine buildings are constructed as one integral structure and were analyzed as one composite structure as explained in Section 3.8.4.1. The turbine portion of the structural complex is Class II designed structure as explained in Section 3.8.6.

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:

Class I - those structures and equipment of which a failure thereof could cause significant release of radioactivity (i.e., calculated offsite doses in excess of 10 CFR 100 or 10 CFR 50.67 as applicable) or are vital to a safe plant shutdown.

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 exceptional requirements are described.

3.8.4.1 Reactor-Turbine Building

3.8.4.1.1 Description of the Structure

The description of the reactor-turbine building is provided in Section 3.7.2.1.1.

3.8.4.1.2 Applicable Codes, Standards, and Specifications

Design and materials are governed by the local and state building codes, the Uniform Building Code (UBC), the ASME Boiler and Pressure Vessel Code the AISC Structural Steel Code, the American Concrete Institute (ACI) Code and by special requirements and standards set forth to provide safety assurance in the event of specific occurrences not covered by the various codes.

3.8.4.1.3 Loads and Load Combinations

General requirements for the design of all structures and equipment include provisions for resisting dead, live, and operating loads and certain additional special loads.

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.
R	=	Jet force or pressure on structure due to rupture of any one pipe.
E	=	Design earthquake load ground acceleration horizontal =0.12g, vertical =0.08g.
E'	=	Maximum earthquake load ground acceleration horizontal =0.24g, vertical =0.16g.
L	=	Wind live load beyond normal building code requirements.

3.8.4.1.4 Design and Analysis Procedures

Analysis procedures for evaluating the effects of wind and tornado loads are described in Section 3.3 and seismic loads are described in Section 3.7.

Class I Structures Criteria

D + R + E	Normal allowable code stresses (AISC for structural steel. ACI for reinforced concrete). (See Table 3.8-11 for a more detailed summary of this criteria related to stress allowables. R loads applicable only to such structures as anchors.) The customary increase in design stresses when earthquake loads are considered is not permitted. [3.8.31]
D + R + E'	Stresses are limited to the minimum yield point as a general case. (See Table 3.8-11 for a more detailed summary of this criteria related to stress allowables. R loads applicable only to such structures as anchors.)
D + L	Withstand maximum potential loadings within code allowable stresses resulting from a wind velocity of 110 mph and ensure structures can sustain without catastrophic failure a wind velocity gust of 200 mph minimum.

3.8.4.1.5 Structural Evaluation

The time history records from the seismic analyses described in Section 3.7.2.1.1 are enveloped to determine the maximum values which are graphically presented in Figures 3.8-25 through 3.8-30. The resulting stresses are also given in tabular form in Tables 3.8-13 through 3.8-16. It is these maximum values that are used by the building designers as the design earthquake load requirements. Twice these values are used to ensure compliance with the design basis earthquake criteria presented previously. [3.8.32]

The allowable shear stresses set forth on Tables 3.8-13 thru 3.8-16 are derived from criteria established in the 1967 edition of the Uniform Building Code (UBC). The allowable shear stresses for building walls and the drywell shield wall are different because they have different H/D ratios. The allowable axial stresses tabulated are all the same.

The allowable axial stress in walls which was used was determined from criteria established by ACI Code 318-63 Section 2202.

Working stress allowables are used for the OBE and ultimate strength allowables were used for the DBE.

The diesel generator room is included in the analysis of the total building system as described in Section 3.7.2.1.1.1. [3.8.33]

The joint details of this reinforced concrete box structure are designed in accordance with "Design of Multi-story Reinforced Concrete Buildings for Earthquake Motion" by Blume, Newmark and Corning. Allowable stresses in the shear walls and diaphragms are in accordance with the UBC, 1967 edition, for both the design and maximum bases earthquake loads.

In summary, the figures showing maximum acceleration with respect to height are used for the seismic design of equipment in the rigid category as defined previously. The moment, shear, and displacement curves are used in the design of the buildings without any increase in allowable stresses for short term loading. The critical structures are also reviewed to assure that the structures can resist double the values presented in the figures in accordance with the safe shutdown criteria previously described.

3.8.4.1.6 Materials, Quality Control, and Special Construction Techniques

Selection of materials to resist these loads is based upon standard practice in the power plant field. [3.8.34]

3.8.4.1.7 Testing and Inservice Inspection Requirements

Visual weld inspection will be performed in accordance with guidelines prepared by the Nuclear Construction Issues Group, NCIG-01, Rev. 2, titled "Visual Weld Acceptance Criteria For Structural Welding at Nuclear Power Plants." [3.8.35]

3.8.4.2 Control Room

The dynamic analysis of the control room is described in Section 3.7.2.1.2. Results of the analysis in terms of shear, moment, and displacement maximum values are presented in Figures 3.8-31 through 3.8-33. The results are also summarized in Table 3.8-17. These values are used in design similarly to the results described in Section 3.7.2.1.1 for the reactor building analyses. [3.8.36]

3.8.4.3 Concrete Chimney

The structural description and dynamic analysis performed for the 310-foot concrete chimney are addressed in Section 3.7.2.1.3.

The results of this dynamic analysis obtained shears, moments, and displacements. The envelopes of the maximum values of these parameters are presented in Figures 3.8-34 through 3.8-36. [3.8.37]

As in the other structures' seismic analyses, the values obtained were used in the design of the structure to assure meeting code allowables without the usual increase for short term loadings. Also, the structure was reviewed to assure that the criteria were met for the maximum earthquake load (twice the parameters values). The factor of safety against overturning due to wind is 10.4, 13.4 for a small earthquake, and 6.7 for a large earthquake.

3.8.4.4 Fuel Pool

The seismic analyses of the fuel pool space, included in the reactor-turbine building analyses, is described in Section 3.7.3.3.1.

Shear values are shown on Figures 3.8-26 and 3.8-29. Maximum shear due to seismic load (OBE) is approximately 23,000 kips from the north-south direction. The resulting shear stress then, considering all shear walls at an elevation cut through the fuel pool, is computed to be about 32 psi. For the DBE (0.24g ground motion), the resulting stress is 64 psi, about two-thirds of the ACI Code shear allowable for normal working stresses. These values are based on both pools containing liquid and are reduced if only one pool is filled. Considering only one pool is filled, the shear stresses due to water pressure are approximately 18 psi. [3.8.38]

Combining the earthquake and pressure stresses, the shear stress is 50 psi (OBE) or 82 psi (DBE). If the plate bending approach, which was conservatively used to determine differential thermal loads, reduces the maximum effective shear area of the pool by a factor of two (one half tension and one half compression in a wall section), the DBE and pressure stress would increase to a value of 162 psi. This value is greater than the allowable but is well below any expected failure stress. In addition, the stresses described above do not consider the additional shear capacity of the reinforcing steel,which is added conservatism. Hence, it is concluded that the fuel pool can adequately sustain the DBE loads simultaneously with maximum thermal gradients and pressure distributions.

3.8.4.5 Class I Masonry Walls

As a result of IE Bulletin 80-11, a re-evaluation of 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 the Bechtel 180-Day Report in response to IE Bulletin 80-11. The analysis contained the function of the wall, the construction materials used, and the construction techniques used in the walls. Eighty-two walls were analyzed for this bulletin. They 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 (ACI 531-79) to determine adequacy of the walls. Walls which were found to be inadequate by this analysis were re-engineered to strengthen them, or were eliminated. [3.8.39]

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 at Quad Cities Units 1 and 2. The minimum embedment depth for wedge type expansion anchors is 4-1/2 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. [3.8.40]

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 had not occurred. Anchors which did not meet embedment criteria or which were undersize were reanalyzed. Anchors shown by reanalysis to be inadequate to support the design loads were replaced, or the expansion anchored plate assembly was 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 anchored assemblies supporting safety-related piping were inspected by applying the test torque to the individual anchors and inspecting for 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 at Quad Cities Units 1 and 2 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.

3.8.5 Foundations

The foundation conditions at the Quad Cities reactor building, and the analysis methods for evaluating stresses are described in Section 3.7.2.1.1.4.

The combined stresses of operating loads plus OBE or DBE for critical sections in the reactor building slab are presented in Table 3.8-12. The allowable stresses are also tabulated next to the actual stresses. In all cases the actuals are less than the allowables. [3.8.41]

The deep slab under the reactor was designed as a two-way slab subjected to a uniform load from the concrete structure and the reactor, and also to the effects of an OBE. The concrete stress resulting from this load combination was 143 psi, and reinforcing steel stress was 10,980 psi, both of which are less than the normal allowable stress of 1690 psi for concrete and 24,000 psi for reinforcing steel permitted by the ACI Code. [3.8.42]

The slab in the suppression chamber area was designed similarly to the reactor foundation slab. The most extreme load case for this slab is that of a suppression chamber column placed over the center of an assumed void. This load condition produces a concrete stress of 435 psi and reinforcing steel stress of 23,100 psi. Both are below the allowable stresses set forth in the preceding paragraph. [3.8.43]

3.8.6 Non-Class I Structures

Class II General Criteria

As noted previously, Class II defines all equipment and structures that are not in the category of Class I. This definition includes the turbine building, the crib house, service building, canal lift station, technical support center, maximum-recycle radwaste building, radwaste storage building, and the radwaste building. These structures and the Class II equipment are designed in accordance with normal practices for design of power plants in the State of Illinois including the local building codes and the basic codes listed for Class I in Section 3.8.4.1 as a minimum. [3.8.44]

In certain specific cases considerably more stringent criteria are used as design criteria. For example, the seismic requirements are set forth as a minimum to conform to a New "Uniform Building Code" Zone 1, but the turbine building was subjected to the same design basis seismic event as the Class I reactor building to obtain design loads. In fact, the two buildings were analyzed as a coupled system.

As an additional example, the crib house walls were investigated for lateral earth effect due to both the OBE and DBE. Stresses in the walls for both cases were below normal allowable ACI Code working stress limits of 20,000 psi for reinforcing steel and 1350 psi for concrete. Therefore, the crib house will not fail and isolate the plant from the river water source. [3.8.45]

3.8.7 References

1. "Quad Cities Nuclear Power Station Units 1 and 2 Plant Unique Analysis Report," Revision 0, May 1983, Transmitted in letter from B. Rybak (CECo) to H.R. Denton (NRC), June 27, 1983.

2. Letter from J.A. Zwolinski (NRC) to D.L. Farrar (CECo), February 15, 1986, "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 Program Plant Unique Load Definition, Quad Cities Station Units 1 and 2," General Electric Company, NEDO-24567, Revision 2, April 1982.

5. "Mark I Program Structural Acceptance Criteria Plant Unique Analysis Applications Guide, Task Number 3.1.3," Mark I Owners Group, General Electric Company, NEDO-24583-1, October 1979.

Table 3.8-1

DRYWELL STRESSES AT EMBEDMENT LOCATION
(ELEVATION 576 FEET)

I. Operational Basis Earthquake

Type of Stress	Without Seismic (psi)	With Seismic (psi)	Critical Buckling Stress (psi)
Stresses when the drywell is flooded to elevation 689.5 feet			
Meridional Stress	6,150	7,620	21,900
Circumferential Stress	21,600	23,100	38,000
Stresses when the drywell is flooded to elevation 629.9 feet			
Meridional Stress	6,340	6,650	18,700
Circumferential Stress	13,500	13,800	38,000

II. Design Basis Earthquake

Type of Stress	Without Seismic (psi)	With Seismic (psi)	Critical Buckling Stress (psi)
Stresses when the drywell is flooded to elevation 689.5 feet			
Meridional Stress	6,150	10,550	21,900
Circumferential Stress	21,600	24,600	38,000
Stresses when the drywell is flooded to elevation 629.9 feet			
Meridional Stress	6,340	6,990	18,700
Circumferential Stress	13,500	14,120	38,000

Table 3.8-2

MAJOR PENETRATION CLASSIFICATION

Penetration	Quantity		Type	Size (in.)
X-7	4	Primary steam	1	20
X-8	1	Primary steam drain	1	3
X-9	2	Primary feedwater	1	18
X-10	1	RCIC, steam supply	1	3
X-11	1	HPCI steam supply	1	10
X-12	1	Shutdown cooling suction	1	20
X-13	2	RHR (LPCI) pump injection	1	16
X-14	1	Cleanup supply	1	6
X-16	2	Core spray	1	10
X-17	1	Vessel head spray (spare)	1	4
X-18	1	Drywell floor drain sump discharge	2	3
X-19	1	Drywell equipment drain sump discharge	2	3
X-20	1	Clean demin water supply	2	3
X-21	1	Service air	4	1
X-22	1	Instrument air	4	11/4
X-23	1	Closed cooling water inlet	1	8
X-24	1	Closed cooling water outlet	1	8
X-25	1	Drywell vent	3	18
X-26	1	Drywell purge	3	18
X-36	1	Old CRD system return (spare)	1	4
X-39	2	Containment cooling spray system	2	10
X-47	1	Standby liquid control inlet	1	1 1/2

Type 1 penetrations allow for movement; Type 2 penetrations allow for relatively little movement; Type 3 and type 4 penetrations are welded directly to the containment shell. Note that Head spray (X-17) and old CRD return (X-36) piping has been deleted; Penetrations X-17 and X-36 are now capped inside the drywell. Type 1 penetrations allow for movement; Type 2 penetrations allow for relatively little

Table 3.8-3

ELECTRICAL PENETRATIONS ENVIRONMENTAL DESIGN CONDITIONS

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 +2.5 psig	0 psig
Relative Humidity (R.H.)	20% — 100%	20% — 100%
Limits of R.H. vs. Lifetime	>50% RH<2% time >70% RH<1% time >90% RH<0.5% time	>50% RH<2% time >70% RH<1% time >80% 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 (See Figure 3.8-39 for an illustration) leak rate shall not exceed the limits 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 (See Figure 3.8-39 for an illustration) leak rate shall not exceed the limits established by the Primary Containment Leakage Rate Testing Program.

Parameter	Inside Containment
Temperature	281°F
Pressure	62 psig
Relative humidity	100% RH

Table 3.8-4

ALLOWABLE STRESSES FOR VENT SYSTEM
COMPONENTS AND COMPONENT SUPPORTS

ITEM		MATERIAL	MATERIAL* PROPERTIES (ksi)	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‡‡ 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	50.81
				LOCAL PRIMARY MEMBRANE	28.95	50.81
				PRIMARY + SECONDARY‡‡ STRESS RANGE	67.83	N/A

Table 3.8-4 (continued)

ALLOWABLE STRESSES FOR VENT SYSTEM
COMPONENTS AND COMPONENT SUPPORTS

ITEM		MATERIAL	MATERIAL* PROPERTIES (ksi)	STRESS TYPE	ALLOWABLE STRESS (ksi)	
					SERVICE** LEVEL B	SERVICE‡ LEVEL C
COMPONENTS	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‡‡ 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‡‡ STRESS RANGE	67.83	N/A

Table 3.8-4 (continued)

ALLOWABLE STRESSES FOR VENT SYSTEM
COMPONENTS AND COMPONENT SUPPORTS

ITEM		MATERIAL	MATERIAL* PROPERTIES (ksi)	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‡‡ 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‡‡ STRESS RANGE	67.83	N/A

Table 3.8-4 (continued)

ALLOWABLE STRESSES FOR VENT SYSTEM
COMPONENTS AND COMPONENT SUPPORTS

ITEM		MATERIAL	MATERIAL* PROPERTIES (ksi)	STRESS TYPE	ALLOWABLE STRESS (ksi)	
					SERVICE** LEVEL B	SERVICE‡ LEVEL C
COMPONENT SUPPORTS	COLUMNS^	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
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 TO TABLE 3.8-4

*	MATERIAL PROPERTIES TAKEN AT MAXIMUM EVENT TEMPERATURES.
**	SERVICE LEVEL B ALLOWABLES ARE USED WHEN EVALUATING NOC I, SBA II, IBA I, DBA I, AND DBA II LOAD COMBINATION RESULTS.
‡	SERVICE LEVEL C ALLOWABLES ARE USED WHEN EVALUATING THE DBA III LOAD COMBINATION RESULTS.
‡‡	THERMAL BENDING STRESSES ARE EXCLUDED WHEN EVALUATING PRIMARY-PLUS-SECONDARY STRESS RANGE.
^	STRESSES DUE TO THERMAL LOADS MAY BE EXCLUDED WHEN EVALUATING COMPONENT SUPPORTS

Table 3.8-5

CLASS MC PIPING ACCEPTANCE CRITERIA

CODE* EQUATION	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**	A, B	3.0 S_m	49.50
13**	A, B	3.0 S_m	49.50
Fatigue‡	A, B	1.0	--

* See NB-3652 and NB-3653 of the ASME Code.
** Required only if Equation 10 is not satisfied.
‡ Cumulative fatigue usage calculation per NB-3653.

Table 3.8-6

MAXIMUM VENT SYSTEM STRESSES FOR CONTROLLING LOAD COMBINATIONS

ITEM		STRESS TYPE	LOAD COMBINATION STRESSES (ksi)*									
			SBA II*		IBA I*		DBA I*		DBA II*		DBA III*	
			CALCULATED STRESS	RATIO TO** ALLOWABLE	CALCULATED STRESS	RATIO TO ** ALLOWABLE	CALCULATED STRESS	RATIO TO ** ALLOWABLE	CALCULATED STRESS	RATIO TO ** ALLOWABLE	CALCULATED STRESS	RATIO TO ** ALLOWABLE
COMPONENTS	DRYWELL SHELL	LOCAL PRIMARY HEAD RANGE	17.07	0.59	12.60	0.44	10.56	0.49	17.39	0.46	20.35	0.40
		PRIMARY AND SECONDARY STRESS RANGE	61.09	0.90	47.44	0.70	N/A	N/A	50.26	0.06	N/A	N/A
	VENT LINE	PRIMARY HEAD RANGE	10.15	0.94	16.18	0.00	17.03	0.38	16.94	0.00	25.57	0.75
		LOCAL PRIMARY HEAD RANGE	9.06	0.34	8.69	0.30	5.39	0.14	9.09	0.24	10.21	0.20
		PRIMARY AND SECONDARY STRESS RANGE	30.82	0.45	26.91	0.40	N/A	N/A	27.75	0.41	N/A	N/A

Table 3.8-6

MAXIMUM VENT SYSTEM STRESSES FOR CONTROLLING LOAD COMBINATIONS

ITEM		STRESS TYPE	LOAD COMBINATION STRESSES (ksi)*									
			SBA II*		IBA I*		DBA I*		DBA II*		DBA III*	
			CALCULATED STRESS	RATIO TO** ALLOWABLE	CALCULATED STRESS	RATIO TO ** ALLOWABLE	CALCULATED STRESS	RATIO TO ** ALLOWABLE	CALCULATED STRESS	RATIO TO ** ALLOWABLE	CALCULATED STRESS	RATIO TO ** ALLOWABLE
COMPONENTS	VENT LINE/VENT HEADER SPHERICAL JUNCTION‡	PRIMARY HEAD RANGE	9.47	0.49	7.91	0.41	7.39	0.38	8.13	0.42	10.07	0.30
		LOCAL PRIMARY HEAD RANGE	15.91	0.55	13.35	0.46	13.67	0.47	14.52	0.50	20.04	0.39
		PRIMARY AND SECONDARY STRESS RANGE	48.23	0.71	35.32	0.52	N/A	N/A	39.15	0.50	N/A	N/A
	VENT HEADER	PRIMARY HEAD RANGE	17.46	0.91	14.66	0.76	10.68	0.97	17.05	0.93	25.90	0.77
		LOCAL PRIMARY HEAD RANGE	20.93	0.72	9.27	0.32	10.96	0.50	10.59	0.49	19.07	0.39
		PRIMARY AND SECONDARY STRESS RANGE	51.67	0.76	29.27	0.43	N/A	N/A	47.30	0.70	N/A	N/A

Table 3.8-6

MAXIMUM VENT SYSTEM STRESSES FOR CONTROLLING LOAD COMBINATIONS

ITEM		STRESS TYPE	LOAD COMBINATION STRESSES (ksi)*									
			SBA II*		IBA I*		DBA I*		DBA II*		DBA III*	
			CALCULATED STRESS	RATIO TO** ALLOWABLE	CALCULATED STRESS	RATIO TO ** ALLOWABLE	CALCULATED STRESS	RATIO TO ** ALLOWABLE	CALCULATED STRESS	RATIO TO ** ALLOWABLE	CALCULATED STRESS	RATIO TO ** ALLOWABLE
COMPONENTS	DOWNCOMER	PRIMARY HEAD RANGE	0.52	0.44	3.00	0.20	11.00	0.62	5.67	0.29	16.25	0.77
		LOCAL PRIMARY HEAD RANGE	20.05	0.69	9.96	0.34	16.63	0.44	16.92	0.45	10.96	0.39
		PRIMARY AND SECONDARY STRESS RANGE	34.70	0.51	10.05	0.16	N/A	N/A	34.81	0.51	N/A	N/A
	SUPPORT COLLAR PLATE	PRIMARY HEAD RANGE	1.09	0.10	1.14	0.06	3.12	0.16	1.43	0.07	3.20	0.48
		LOCAL PRIMARY HEAD RANGE	6.20	0.22	5.07	0.18	9.97	0.26	5.40	0.15	10.22	0.37
		PRIMARY AND SECONDARY STRESS RANGE	57.50	0.05	3.41	0.50	N/A	N/A	49.20	0.73	N/A	N/A

Table 3.8-6

MAXIMUM VENT SYSTEM STRESSES FOR CONTROLLING LOAD COMBINATIONS

ITEM		STRESS TYPE	LOAD COMBINATION STRESSES (ksi)*									
			SBA II*		IBA I*		DBA I*		DBA II*		DBA III*	
			CALCULATED STRESS	RATIO TO** ALLOWABLE	CALCULATED STRESS	RATIO TO ** ALLOWABLE	CALCULATED STRESS	RATIO TO ** ALLOWABLE	CALCULATED STRESS	RATIO TO ** ALLOWABLE	CALCULATED STRESS	RATIO TO ** ALLOWABLE
COMPONENT SUPPORTS	SUPPORT COLUMNS	BENDING	9.70	0.50	6.73	0.35	3.07	0.16	11.71	0.60	6.93	0.27
		TENSILE	3.06	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	0.35	0.91	0.91	0.90	0.90	0.04	0.04
		COMPRESSION	5.14	0.42	3.56	0.45	3.46	0.29	3.39	0.28	4.42	0.27
		INTERACTION	0.99	0.99	0.00	0.00	0.46	0.46	0.91	0.91	0.58	0.50
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

* See Section 3.8.2.2.3 for discussion of load combinations.

** See Table 3.8-1 for allowable stresses.

‡ Local stresses are reported at the vent line - vent header junction. For local stresses at the vent line - SRVDL penetrations, see Table 3.8-4.

Table 3.8-7

STRESS ANALYSIS RESULTS - CLASS MC PIPING

ASME Code Paragraph	Code Equation	Service Level	Stress (ksi)/Usage	
			Calculated	Allowable
NB-3652	9	Design	2.95	24.75
	9	C	12.71	37.13
NB-3653	10	A, B	74.35*	49.50
	12	A, B	3.10	49.50
	13	A, B	48.46	49.50
	Fatigue	A, B	0.18	1.0

* This is acceptable in accordance with the ASME Code, as long as equations 12 and 13 (from NB-3653) are satisfied.

Table 3.8-8

ALLOWABLE STRESSES FOR SUPPRESSION CHAMBER
COMPONENTS AND SUPPORTS

Item		Material	Material Properties (ksi)[Note 1]	Stress Type	Allowable Stress (ksi) Service Level B[Note 2]	Service Level C[Note 3]	Service Level D[Note 4]
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 and Secondary Stress Range [Note 5]	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 and Secondary Stress Range [Note 5]	69.51	N/A	N/A

Table 3.8-8 (Continued)

ALLOWABLE STRESSES FOR SUPPRESSION CHAMBER
COMPONENTS AND SUPPORTS

Item		Material	Material Properties (ksi)Note 1	Stress Type	Allowable Stress (ksi) Service Level BNote 2	Service Level CNote 3	Service Level DNote 4
Component Supports	Column ConnectionNote 6	SA-516 Grade 70	$S_{\{y}$= 35.86	Membrane	21.52	28.69	43.04
				Extreme Fiber	26.90	35.87	53.80
	SaddleNote 6	SA-516 Grade 70	S_y= 35.86	Membrane	21.52	28.69	43.04
				Extreme Fiber	26.90	35.87	53.80
Welds	Ring Girder to Shell	SA-516 Grade 70	S_{mc}= 19.30	Primary	10.62	19.72	22.91
			S_y= 35.86	Primary and Secondary	31.85	N/A	N/A
	Column Connection to Shell	SA-516 Grade 70	S_{mc}= 19.30	Primary	10.62	19.72	22.91
			S_y= 35.86	Primary and Secondary	31.85	N/A	N/A
	Saddle to Shell	SA-516 Grade 70	S_{mc}= 19.30	Primary	10.62	19.72	22.91
			S_y= 35.86	Primary and Secondary	31.85	N/A	N/A

Table 3.8-8 (Continued)

ALLOWABLE STRESSES FOR SUPPRESSION CHAMBER
COMPONENTS AND SUPPORTS

Note 1 Material properties are taken at the maximum event temperature.

Note 2 Service Level B allowables are used when evaluating SBA III, IBA I, IBA III, IBA IV, and DBA II load combination results.

Note 3 Service Level C allowables are used when evaluating IBA V and DBA IV load combination results.

Note 4 Service Level D allowables are used when evaluating DBA I load combination results.

Note 5 Thermal bending stresses may be excluded when comparing primary-plus-secondary stress range values to allowables.

Note 6 Stresses due to thermal loads may be excluded when evaluating component supports.

Table 3.8-9

MAXIMUM SUPPRESSION CHAMBER STRESSES FOR CONTROLLING LOAD COMBINATIONS

Item		Stress Type	Load Combination 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	15.98	0.83	16.14	0.84	15.39	0.80	19.60	0.55
		Local Primary Membrane	20.81	0.72	19.22	0.66	14.80	0.51	26.23	0.49
		Primary and Secondary Stress Range	68.45	0.98	65.35	0.94	47.71	0.69	N/A	N/A
	Ring Girder	Primary Membrane	18.04	0.93	15.20	0.79	19.09*	0.99	30.16	0.84
		Local Primary Membrane	26.21	0.91	24.25	0.84	28.17*	0.97	33.37	0.62
		Primary and Secondary Stress Range	55.14	0.79	53.57	0.77	57.41*	0.83	N/A	N/A

Table 3.8-9 (Continued)

MAXIMUM SUPPRESSION CHAMBER STRESSES FOR CONTROLLING LOAD COMBINATIONS

Item		Stress Type	Load Combination 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
Supports	Column Collection	Membrane	13.00	0.60	12.54	0.58	9.12	0.42	18.91	0.66
		Extreme Fiber	13.06	0.49	12.58	0.47	9.15	0.34	19.61	0.55
	Saddle	Membrane	18.03	0.84	16.65	0.77	12.71	0.59	36.54	0.93
		Extreme Fiber	18.09	0.67	16.71	0.62	12.74	0.47	32.81	0.91
Welds	Ring Girder to Shell	Primary	22.44**	1.00	22.12**	0.98	21.62‡	0.94	22.37	0.80
		Secondary	44.66	0.83	45.85	0.85	50.30*	0.93	N/A	N/A
	Column Connection to Shell	Primary	18.39**	0.82	17.74**	0.79	12.90	0.86	17.60	0.63
		Secondary	18.47	0.34	17.79	0.33	12.94	0.24	N/A	N/A
	Saddle to Shell	Primary	17.40	0.85	17.36	0.85	11.74	0.57	26.66	0.70
		Secondary	17.50	0.28	17.44	0.28	11.78	0.19	N/A	N/A

Table 3.8-9 (Continued)

MAXIMUM SUPPRESSION CHAMBER STRESSES FOR
CONTROLLING LOAD COMBINATIONS

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* These results are controlled by the zero ring girder stiffener model.

** This local primary membrane stress has an allowable based on 1.5 S_{mc}.

‡ These results are controlled by the zero ring girder stiffener model and this local primary membrane stress has an allowable based on 1.5 S_{mc}.

Table 3.8-10

TORUS STRESSES WITH A FLOODED DRYWELL (0.3g SEISMIC LOAD)

	Actual	Allowable
Torus Plate Stress, ksi	16.00	23.30
Torus Ring Stress, ksi	21.97	26.20
Bracing Stress, ksi	13.70	17.90
Seismic Anchors		
Pin — Shear Stress, ksi	14.30	24.00
Plates — Flexural Stress, ksi	21.50	29.30
Column Interaction		
Formula Ratio	.683	1.33

Table 3.8-11

ALLOWABLE STRESSES FOR CLASS I BUILDING TYPE STRUCTURES

Loading Conditions	Reinforcing Steel Max. Allowable Stress	Concrete Max. Allowable Compression Stress	Concrete Max. Allowable Shear Stress	Concrete Max. Allowable Bearing	Structural Steel Tension On the Net Section	Structural Steel Shear On Gross Section	Structural Steel Compression on Gross Section	Structural Steel Bending
1. *(Dead loads plus live loads, plus operating load) plus (seismic)** loads (0.12g)	0.5 F_y (A-615 40 grade) 0.4 F_y (A-615 60 grade)	0.45 f'_c	1.1 $(f'_c)^{1/2}$	0.25 f'_c	.60 F_y	0.40 F_y	Varies with slenderness ratio	0.66 F_y — 0.60 F_y
2. *Dead loads plus live loads, plus operating loads plus wind loads	0.667 F_y	0.60 f'_c	1.467 $(f'_c)^{1/2}$	0.333 f'_c	0.80 F_y	0.53 F_y	Varies with slenderness ratio	0.88 F_y — 0.80 F_y
3. (Dead loads, plus live loads, plus operating loads,) plus (seismic) ‡ loads (0.24g)	Safe shutdown of the plant can be achieved‡‡							
4. (Extended Power Uprate Loads)	Refer to Section 3.9.3.1.3.4							

F_y = Minimum yield point of the material.
f'_c = Compressive strength of concrete.
* Loadings are defined in Section 3.8.4.1 as D loads.
** Load E, design basis earthquake of Section 3.8.4.1.
‡ Load E´, maximum earthquake of Subsection 3.8.4.1.
‡‡ 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.

Table 3.8-12

STRESSES IN REACTOR BUILDING SLAB
(All Stresses Are in PSI)

Location	Concrete Stress Actual	Concrete Stress Allowable	Reinforcing Stress Actual	Reinforcing Stress Allowable
OBE Combination				
Slab under reactor	143	1,690	10,980	24,000
Slab under suppression chamber	435	1,690	23,100	24,000
DBE Combination				
Slab under reactor	232	3,200	18,500	54,000
Slab under suppression chamber	482	3,200	25,600	54,000

Table 3.8-13

STRESSES IN REACTOR BUILDING WALLS
OBE, N-S DIRECTION (STRESSES IN PSI)

Element	Stress	Seismic Horizontal	Seismic Vertical	Grav.	Total	Allow.
Section 666 feet 6 inches to 690 feet 6 inches						
Building walls	Shear	89			89	184
	Axial	11	9	112	132	1690
Drywell shield wall	Shear	---				102
	Axial	---	6	77	83	1690
Section 647 feet 6 inches to 666 feet 6 inches						
Building walls	Shear	118			118	184
	Axial	28	13	163	204	1690
Drywell shield wall	Shear	41			41	102
	Axial	---	10	126	136	1690
Section 623 feet 0 inches to 647 feet 6 inches						
Building walls	Shear	117			117	184
	Axial	52	18	221	291	1690
Drywell shield wall	Shear	55			55	102
	Axial	47	22	276	345	1690
Section 595 feet 0 inches to 623 feet 0 inches						
Building walls	Shear	108			108	184
	Axial	79	32	403	514	1690
Drywell shield wall	Shear	98			98	102
	Axial	124	25	305	454	1690

Table 3.8-13 (continued)

STRESSES IN REACTOR BUILDING WALLS
OBE, N-S DIRECTION (STRESSES IN PSI)

Element	Stress	Seismic Horizontal	Seismic Vertical	Grav.	Total	Allow.
Section 571 feet 6 inches to 595 feet 0 inches						
Building walls	Shear	46			46	184
	Axial	53	22	274	349	1690
Drywell shield wall	Shear	47			47	102
	Axial	110	13	163	286	1690
Section 554 feet 0 inches to 571 feet 6 inches						
Building walls	Shear	46			46	184
	Axial	57	21	253	331	1690
Drywell shield wall	Shear	24			24	102
	Axial	147	12	153	312	1690

Table 3.8-14

STRESSES IN REACTOR BUILDING WALLS
OBE, E-W DIRECTION (Stresses In PSI)

		Seismic				
Element	Stress	Horizontal	Vertical	Grav.	Total	Allow.
		Section 666 feet 6 inches to 690 feet 6 inches				
Building walls	Shear	67			67	184
	Axial	20	9	112	141	1690
Drywell shield wall	Shear	---			---	102
	Axial	---	6	77	83	1690
		Section 647 feet 6 inches to 666 feet 6 inches				
Building walls	Shear	108			108	184
	Axial	45	13	163	221	1690
Drywell shield wall	Shear	42			42	102
	Axial	---	10	126	136	1690
		Section 623 feet 6 inches to 647 feet 6 inches				
Building walls	Shear	124			124	184
	Axial	70	18	221	309	1690
Drywell shield wall	Shear	87			87	102
	Axial	47	22	276	372	1690
		Section 595 feet 0 inches to 623 feet 0 inches				
Building walls	Shear	77			77	184
	Axial	85	32	403	520	1690
Drywell shield wall	Shear	71			71	102
	Axial	114	25	305	444	1690

Table 3.8-14 (continued)

STRESSES IN REACTOR BUILDING WALLS
OBE, E-W DIRECTION (Stresses In PSI)

Element	Stress	Seismic Horizontal	Seismic Vertical	Grav.	Total	Allow.
Section 571 feet 6 inches to 595 feet 0 inches						
Building walls	Shear	33			33	184
	Axial	42	22	274	338	1690
Drywell shield wall	Shear	33			33	102
	Axial	110	13	163	286	1690
Section 554 feet 0 inches to 571 feet 6 inches						
Building walls	Shear	33			33	184
	Axial	45	21	253	319	1690
Drywell shield wall	Shear	17			17	102
	Axial	122	12	153	287	1690

Table 3.8-15

STRESSES IN REACTOR BUILDING
DBE, N-S DIRECTION (Stresses In PSI)

By inspection, all the compressive stresses in the building walls and drywell are satisfactory for cond. two in both directions;

Most critical cond. at Section 595 feet 0 inches to 623 feet 0 inches

$f_b = 2(85) + 2(32) + 403(1.25) = 734$ psi < 2870 psi

Section	Element	Shear	Allow.
66 feet 6 inches to	Building walls	178	282
690 feet 6 inches	Drywell shield wall	---	156
647 feet 6 inches to	Building walls	236	282
647 feet 6 inches	Drywell shield wall	82	156
623 feet 0 inches to	Building walls	234	282
647 feet 6 inches	Drywell shield wall	110	156
595 feet 0 inches to	Building walls	216	282
623 feet 0 inches	Drywell shield wall	196	156
571 feet 6 inches to	Building walls	92	282
595 feet 0 inches	Drywell shield wall	94	156
554 feet 0 inches to	Building walls	92	282
531 feet 6 inches	Drywell shield wall	48	156

Table 3.8-16

STRESSES IN REACTOR BUILDING WALLS
DBE, E-W DIRECTION (Stresses In PSI)

Section	Element	Shear	Allow.
666 feet 6 inches to	Building walls	134	282
690 feet 6 inches	Drywell shield wall	---	156
647 feet 6 inches to	Building walls	216	282
647 feet 6 inches	Drywell shield wall	84	156
623 feet 0 inches to	Building walls	248	282
647 feet 6 inches	Drywell shield wall	174	156
595 feet 0 inches to	Building walls	144	282
623 feet 0 inches	Drywell shield wall	142	156
571 feet 6 inches to	Building walls	66	282
595 feet 0 inches	Drywell shield wall	66	156
554 feet 0 inches to	Building walls	66	282
571 feet 6 inches	Drywell shield wall	34	156

Table 3.8-17

SUMMARY OF SEISMIC ANALYSIS OF CONTROL ROOM (STRESSES IN PSI)

Element	Type of Stress	Seismic (Stresses in PSI) Horizontal	Vertical	Gravity	Total	Allowable
OBE E-W Direction						
South wall	Shear	79	-	-	79	164
Elev. 595 ft 0 in. – 641 ft 2 in.	Axial	53	7	84	144	1690
OBE N-S Direction						
East wall	Shear	85	-	-	85	164
Elev. 595 ft 0 in. – 641 ft 0 in.	Axial	242	7	84	333	1690
DBE E-W Direction						
South wall	Shear	158	-	-	158	252
Elev. 595 ft 0 in. – 641 ft 2 in.	Axial	106	14	84	204	1690
DBE N-S Direction						
East wall	Shear	170	-	-	170	252
Elev. 595 ft 0 in. – 641 ft 0 in.	Axial	484	14	84	582	1690

3.9 MECHANICAL SYSTEMS AND COMPONENTS

This section addresses the design of mechanical systems and components. Due to the limited scope of the ASME Code edition applicable when construction permits are issued for Quad Cities Station, only the reactor pressure vessels were categorized as ASME Code Class 1. The design transients (thermal cycles) applicable to the reactor vessels and current 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 description of operational vibration analyses of the recirculation system, an example of typical seismic equipment evaluations, 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.

Design of the Quad Cities control rod drive systems is addressed in Section 4.6.

Section 3.9.5 summarizes the layout, design bases, and qualification of the reactor vessel internals.

Finally, inservice inspection and testing programs for pumps and valves are 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 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 33).

3.9.1 Special Topics for Mechanical Components

This subsection presents the design transients and addresses the fatigue evaluation for the reactor pressure vessels at Quad Cities. Overall acceptance criteria, loading conditions and a discussion of the design evaluation of the vessels and supports are presented in Section 3.9.3.1.2. The evaluation of reactor vessel internals is covered in Section 3.9.5.

3.9.1.1 Design Transients

The construction permits for Quad Cities Units 1 and 2 were issued on February 15, 1967. At that time, the ASME Code covered only reactor vessels. Pumps, piping and valves were built primarily to the USAS B31.1 Power Piping Code rules. Thus, Quad Cities Station originally had no ASME Code Class 1, 2 or 3 designed piping systems. [3.9.1]

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, operating transient cycles through March 1988 were redefined using actual plant data for Units 1 and 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. In particular, the actual number of safety relief valve (SRV) blowdowns at Unit 1 exceeded the original design basis, thereby necessitating the revised fatigue evaluation. Cycle predictions for the scram, heatup, cooldown, and loss of feedwater heater transients were also predicted to exceed the original design basis within 40 years of operation. Revised usage factors (see table below) were calculated for the vessel and components based on the predicted cycles for 40 years. [3.9.2]

Fatigue Usages, Summary for Limiting Components		
Component	EPU Fatigue Usage Factor, U	Code Allowable
Shroud support	0.820	1.0
Support Skirt	0.862	1.0
Feedwater Nozzle	0.748	1.0
Closure Studs	</=1.0	1.0

Several developments made it prudent to further refine the RPV evaluation:

A. An estimation of the time for the closure studs to reach a usage of 1.0 was desirable; and

B. A rapid cooldown event, associated with a stuck-open relief valve, which is bounded only by the SRV blowdown event, occurred in April 1989 at Unit 1. This, along with a similar event in 1980, which was inadvertently counted as a cooldown, caused the original design basis of one SRV blowdown cycle to be exceeded.

The original RPV stress report included a fatigue analysis for the reactor vessel components based on a set of design basis duty cycles. In the current analysis[1], fatigue usage factors were recalculated based, in most cases, on methods found in the original stress report, but using the new 40-year cycle predictions. This approach resulted in the original design cycle numbers being applied for all but the following six transients.

A. Scram,

B. Heatup,

C. Cooldown,

D. Loss of feedwater heaters,

E. Batch feedwater addition during hot standby or plant cooldown, and

F. Safety relief valve blowdown.

For scram, heatup, cooldown, and loss of feedwater heaters, predicted cycles were increased, based on cycle counting results. The batch feedwater addition during hot standby or plant cooldown original design basis transient was determined to be not applicable for operation at Units 1 and 2, so the number of these cycles was reduced.

Information from the original RPV stress report was used for all but two components, the feedwater nozzle and the support skirt. For the feedwater nozzle, a more recent analysis was available. Therefore, the new feedwater nozzle fatigue evaluation considered the revised analysis, information concerning rapid cycling, removal of batch feedwater addition cycles where appropriate, and the inclusion of the additional 40-year transient cycles in arriving at new fatigue usage factors. For the RPV support skirt, results from the revised fatigue analysis, which are based on the predicted 40-year cycles, were used.

With the first application of the Extended Operating Domain and Equipment Out-of-Service for Quad Cities, the feedwater nozzle fatigue analysis was revised [23]. It assumed 18 month fuel cycles and a Final Feedwater Temperature Reduction (FFWTR) to 230°F for 14 days followed by a coastdown to 70% power over a period of 12 weeks. The feedwater temperature at the end of this coastdown was 210°F. The methods and further assumptions along with the results of that analysis [23] were outlined in the GE proprietary document. [3.9.3]

Specific calculations of stress and fatigue for the SRV blowdown transient were performed for the recirculation inlet nozzle, the shroud support, the support skirt, and the feedwater nozzle. These RPV components were selected because of their high fatigue usage factors, relative to other RPV components. The allowable number of SRV blowdown cycles was determined from these specific analyses. For the remaining components, the SRV blowdown fatigue usage was conservatively estimated at being equal to the worst usage of the analyzed components above.

Each reactor vessel component considered in the original fatigue evaluation with a significant predicted usage factor, except the closure studs, was individually analyzed for fatigue usage for the revised duty cycles. An evaluation was then done to estimate the number of SRV blowdown cycles which resulted in a total usage of 1.0. Closure studs were not considered here because the predicted usage exceeds 1.0 without SRV blowdown cycles.

Utilizing the aforementioned methods, vessel fatigue usages were evaluated based on 40-year cycle predictions, including 12 cycles of SRV blowdown and 20 additional cycles of heatup, as shown in Table 3.9-1. The usage results based on this set of duty cycles is provided in Table 3.9-2.

A review of Table 3.9-2 shows that the fatigue usage values for all components except the closure studs are at or below the value of 1.001. The results for the control rod drive (CRD) hydraulic return nozzle, which do not account for the nozzle being capped, are conservative. The fact that the head spray nozzle has also been capped has no effect on the results in Table 3.9-2, as head spray is not a limiting transient for any of the vessel components except the head spray nozzle, and that nozzle is exempt from fatigue analysis. In fact, the head spray transient has been retained in the fatigue evaluation (despite the line being capped) only because several head spray transients occurred in the past at Quad Cities. Therefore, the analyzed cycles in Table 3.9-1 have been adopted as the revised fatigue design basis cycles for those components.

The revised RPV fatigue evaluation was approved by the NRC in February 1991[2].

The RPV fatigue evaluation was reviewed for potential impact based on the revised seismic analysis in support of the core shroud modification. There was no impact on this RPV fatigue evaluation.

Additional analysis has been performed to determine the time at which the closure studs are predicted to reach a usage value of 1.0. The cycle counting procedure included methods for predicting numbers of cycles for a given year of operation. These were used in conjunction with the usage per cycle values for boltup, pressure test and unbolt (one value), for heatup-cooldown and scram to determine fatigue usage as a function of years of operation. As part of the long term plan to resolve the vessel closure studs fatigue usage, additional analysis was performed in May 1999. The results of this analysis show that the cumulative fatigue usage factor (CFUF) for the vessel closure studs is less than 1.0 for both Units 1 and 2 at the end of the forty-year design life. Further analysis was performed in 2003 in support of flood-up of the RPV using Feedwater/Condensate resulting in additional limitation on the number of several vessel stress cycles as described in Table 3.9-1A. The results of these analyses show that the usage factor meets the allowable limit of 1.0 established in the ASME Section III Code and as a result justifies forty years of operation. See Section 5.3.1.7.

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 3.9.3.1.

3.9.2 Dynamic Testing and Analysis

The following subsections discuss analyses performed to assess piping vibration, provide results of a typical seismic analysis for mechanical equipment, and discuss the evaluation and testing of vibration of the reactor internals.

3.9.2.1 Piping Vibration, Thermal Expansion, and Dynamic Effects

Vibration analyses were performed on representative recirculation systems to determine the effects of excitation internal to the system. Because of the insignificant vibration induced stress levels and the geometric similarity between plants, analyses were not performed on each plant. [3.9.4]

Outlined below are the principal assumptions, results, and a discussion of an analysis applicable to the Quad Cities Station:

A. Recirculation system model:

   Two dimensional lumped mass system with 1% structural damping. Auxiliary lines were excluded.

B. Excitation:

   Excitation was taken as that from pump motor imbalance, sinusoidally varying, with a variable peak amplitude. Mechanical excitation: 5.83—27.83 cps.

C. Results:

1. Natural frequency of system	
Fundamental of system	0.868 cps
2nd frequency	8.308 cps
3rd frequency	12.039 cps
4th frequency	28.050 cps
2. Peak Deflection	0.0002 at 8.303 cps
3. Maximum vibration-induced stress	30 psi

The maximum stress level of 30 psi for the lateral vibration analysis is very low. A two-dimensional model of the recirculation loop piping neglects the torsional modes of vibration. The first torsional natural frequency is usually slightly higher than the lateral natural frequency. The second, third, and fourth torsional frequencies increase in the same fashion as the translational natural frequencies. It is reasoned that if a torsional mode with a natural frequency near the pump speed exists, it will be one of the higher harmonics of the torsional natural frequencies. Hence, if it is excited, the deflections will be similar to those obtained from the lateral vibrational analysis and the combined effect is negligible.

This insignificant stress level precludes the need for a more refined model and includes all possible sources of internal excitation.

The safety-related Main Steam (MS) piping and the safety-related Feedwater (FW) piping will have increased flow rates and flow velocities in order to accommodate extended power uprate. The MS and FW piping will experience increased vibration levels, approximately proportional to the square of the flow velocities and also in proportion to any increase in fluid density. (Other piping systems are not affected.) The ASME code requires some vibration test data be taken and evaluated per the nuclear regulatory guidelines for these high energy piping systems, when initially operated at extended uprate conditions. Vibration data for the MS and FW piping inside containment, must be acquired using remote sensors. A piping vibration startup test program, which meets the ASME code, per the regulatory guidelines, will be performed for each unit. This program is outlined below.

The piping vibration levels of two large piping systems within containment for each plant will be monitored during initial plant operation at the new extended power uprate operating conditions. The startup vibration test program performed for each unit is expected to show that these piping systems are vibrating at acceptable levels during extended power uprate conditions.

The two piping systems that are affected by an extended power uprate, which must be monitored for vibrations for each plant, are the Main Steam Line system piping and the Feedwater system piping. These two piping systems will be monitored for vibration, because the mass flow rates in these piping systems will increase noticeably during extended power uprate operations. The mass flow rates in these systems will increase approximately in proportion to the extended power level increase. The flow induced vibration levels for these two piping systems will simultaneously increase in an amount, which is at least in proportion to the increase in the fluid density and the square of the fluid velocity. The vibration levels may even be higher if other flow induced vibration mechanisms occur.

The main steam piping system, four separate steam lines, goes from the reactor pressure vessel to the steam turbine. Main steam piping runs, which are inside containment, will be monitored with remote vibration sensors. Main steam piping runs outside containment can be monitored during a system walkdown by visual observations, accompanied by hand-held vibration monitoring instruments, and if necessary, by the use of remote sensors.

The feedwater piping system originates outside of containment (also separate lines) and then goes through the containment penetrations to the reactor pressure vessel. Again, feedwater system piping runs inside containment will be monitored with remote vibration sensors and feedwater system piping outside of containment can be monitored during a system walkdown by visual observations, accompanied by hand-held vibration monitoring instruments and if deemed necessary, also by using of remote sensors.

The vibration stress levels for these two piping systems must stay below certain criteria. The allowable alternating (vibration) stress levels are quantified in Section III of the ASME code. The allowable alternating vibration stress levels will be used to pre-establish an acceptance criterion for each vibration sensor used for monitoring this piping vibration.

Vibration data will be collected at approximately 50%, 75% and 100%, of the original licensed power and at select power increments up to the maximum extended power uprate condition. For this extended power uprate, the maximum power level at which data will be taken is at 100% uprated power or the maximum power level attainable, given any balance of plant limitations. First, the measured vibration levels at the current rated thermal power will be compared to the acceptance criteria. Using this information, the vibration levels expected at the new, higher power levels shall be extrapolated. Vibration data will be collected at interim Test Conditions, which correspond to each (5%) step increase in power level above original licensed power, to the final power uprate level, and compared to the acceptance criteria. As the plant steps up to the new, maximum extended power level, the piping vibration levels shall be monitored to ensure the newly, measured vibration levels are acceptable. In this manner, the vibration monitoring testing can proceed as the plant operates for the first time at each new power level, and at the same time avoid the remote possibility of incurring high vibrations and damaging the plant equipment (piping), before appropriate corrective actions can be taken.

Once a unit achieves the maximum reactor power level planned for extended power uprate, the test program is considered complete. After testing, additional vibration data analysis shall be performed to ensure the vibration levels are indeed acceptable.

As part of the piping vibration test program, a Test Specification, Test Plan and Procedure, Preliminary Test Report and Final Test Report will be prepared, to properly direct and document each phase of this test program, which will be performed for each unit.

Subsequent to the EPU startup testing described above, it was determined that EPU power operation resulted in higher than desirable acoustic vibration frequencies in the 140 to 160 Hz range. Acoustic Side Branches (ASBs) are installed in the inlet piping to the Main Steam Safety Valves and Electromatic Relief Valves to de-tune the main steam system from forming these acoustic vibrations at EPU flow velocities. The ASBs are passive devices that reduce the acoustic vibration loading on the RPV Steam Dryer and other components connected to the main steam lines.

3.9.2.2 Seismic Qualification of Safety-Related Mechanical Equipment

The analysis of the core spray pump, which is described in the following paragraphs, provides an example of the general methods applied to the multitude of equipment in the plant and the types of considerations used in the plant design. This Class I mechanical equipment has been evaluated to assure compliance with the seismic requirements and criteria. Major mechanical equipment, in general, was analyzed to determine the natural period of vibration and this was evaluated against the building response to assure that system integrity is maintained for the operating and design basis earthquake loadings. [3.9.5]

Equipment such as ECCS pumps, etc., were evaluated by determining the equipments' natural frequency and then evaluating the magnification of earthquake motion that would result. As an example, the core spray pump system consists of a support, motor, and pump. [3.9.6]

The motor is mounted over the pump and the pump anchored to the building. Spring stiffness of the supporting network was determined and two directions of motion were then evaluated. The most critical period of the two directions dictates the maximum degree of amplification. In this case this period is 0.017 seconds which indicates a very rigid system with respect to the support structure and no amplification results. The building acceleration for the elevation under consideration then provides the horizontal load to the pump system. Using this acceleration and considering the building vertical acceleration to act simultaneously the seismic forces and stresses were determined. For the core spray pump the maximum stress for the design basis earthquake resulted in a maximum shear of 768 psi vs. an AISC Code allowable of 10,000 psi and a tensile maximum of 2910 psi vs. an allowable of 14,000 psi. These values are the maximums for the design basis earthquake and occur in the anchor bolts. Other components of this system have lower stresses. The allowable stress per the criteria for maximum conditions is yield which is more than a factor of 2 greater than the code allowables shown above. This approach to equipment is typical of the method of evaluation and results obtained on the Class I equipment. In addition, the evaluation of Class I equipment for piping nozzle loads is discussed in Section 3.9.3.1.2.3.

The seismic testing and analysis of instrumentation and electrical equipment is described in Section 3.10.

The Extended Operating Domain and Equipment Out-of-Service for Quad Cities[23] discusses the seismic/LOCA evaluation that was performed for the GE fuel. Compared to the surrounding structures (shroud, vessels, etc.), a change in fuel channel thickness could be significant. The fuel is significant mass in the horizontal RPV and internals mathematical model and is modeled as a separate element. Also, since the fuel is coupled to the shroud, vessel and shield wall, the seismic response of these structures is affected. [3.9.7]

The effects of AREVA (now Framatome) ATRIUM 10XM fuel on the seismic response of the reactor pressure vessel internals was evaluated by GEH.[32] It was demonstrated that for ATRIUM 10XM fuel, there would be an insignificant effect on the loadings. The fuel assembly liftoff during a combined seismic and LOCA event is acceptable compared to criteria for liftoff.[24]

3.9.2.3 Vibration Testing of Reactor Vessel Internals

This section addresses the overall approach for assuring the integrity of the Quad Cities reactor internals under flow-induced vibration loads. Ordinarily, a reactor prototype or "first-of-a-kind" design would warrant extensive vibration testing and analysis. Test and analysis results would then be correlated to ensure a good match between experimental and predicted results. In the case of Quad Cities, however, a large body of both experimental and analytical data already existed for other similar GE BWRs.

The GE criteria for selecting BWR plants to be vibration tested is to test each new reactor core support structure where there has been a significant design departure from a reactor that has been previously vibration tested (e.g., the first reactor of each standard reactor design). Since all BWR jet pump plants are geometrically similar, it is not expected that there will be a great deal of difference in the vibration response of the various reactors of approximately the same size. However, on new reactor designs where vessel diameters change significantly, or where flow velocities increase significantly from a reactor that has already been tested, vibration tests would be warranted. [3.9.8]

The reactor core support structure of the Quad Cities reactor is identical in design to the Dresden Units 2 and 3 reactors (i.e., the same set of design drawings used for each reactor); therefore the vibrational characteristics of these reactors, such as natural frequencies, mode shapes and damping factors, are the same. Furthermore, the flow paths, flow velocities, and fluid pressures, temperatures and densities in the Quad Cities reactors are essentially identical to Dresden Units 2 and 3; therefore, the character of the exciting forces which result from fluid flow are essentially identical in these reactors. When the core support structures of two or more reactors have the same vibration characteristics and the same forcing functions, the vibration response of all these plants will be essentially the same.

Preliminary examination of vibration data collected on various BWR jet pump plants of different sizes (e.g., vessel diameters of 188, 205, 224, and 251 inches) and flow rates indicates that, even for large changes in structure sizes, there is practically no vibration of any component during balanced flow conditions which vary from zero to full flow. For the transient unbalanced flow conditions (i.e., one recirculation pump speed much greater than the other), there is no specific vibration displacement criteria because for each different amplitude of vibration there would be a corresponding number of permissible cycles.

For Dresden Unit 2, the 40-year steady-state vibration criteria was initially used to judge the acceptability of transient vibrations. These criteria specify the amplitude of vibration which could safely be permitted on a continuous basis for the entire 40 years of plant operation without causing a component to fail. Using these steady-state criteria to judge the acceptability of short duration transient vibrations is an extremely conservative approach and yet transient vibration measurements of all components in the Dresden Unit 2 reactor easily satisfied these strict criteria, except for the jet pump riser brace.

Instead of a displacement criteria, a fatigue usage factor criteria can be developed. A calculation was made to determine the acceptability of exceeding the steady-state criteria for the Dresden Unit 2 jet pump riser brace on a short-term transient basis and it was found that the total fatigue usage value, based on the transient observed at Dresden Unit 2 was 0.3 (ASME Code allowable = 1.0). Even though this amount of fatigue usage is acceptable, a more desirable and much more conservative alternative of instituting procedural controls to avoid unbalanced pump operation has been accomplished. This, in effect, eliminates the excess vibration caused by the pump restart transient (which accounted for a fatigue usage value of 0.28 out of a total of 0.3). These same procedural controls apply to Quad Cities. In addition, the riser brace design on Quad Cities has been improved over the Dresden Unit 2 design such that the peak stresses are reduced by a factor of 3 for a given displacement.

Sufficient vibration data were obtained from Dresden Unit 2 to assure that, with procedural controls, no unacceptable vibration will be present in the Dresden/Quad Cities plants.

In summary then, any vibration of the Quad Cities reactor internals will be well within acceptable levels without actual vibration testing at Quad Cities for the following reasons:

A. The geometrical design and the vibration characteristics of the Quad Cities core support structures are identical to the Dresden 2 and 3 reactors which have been vibration tested.

B. The flow paths, flow velocities, and general character of the exciting forces in Quad Cities are identical to the Dresden Units 2 and 3 reactors.

C. Vibration test results on Dresden Units 2 and 3 indicate that, with limited procedural controls, there will be no unacceptable vibration of the reactor internals at any flow rate (up to full flow) and at any power level. Therefore, there should be no unacceptable vibration in any of the Dresden/Quad Cities plants.

D. No unacceptable vibration has been observed during balanced flow for any size of BWR plant tested to date.

The overall scope of the BWR internals vibration test program, including a list of the BWR plants scheduled for test, a discussion of the vibration acceptance criteria and method of analysis, a description of a typical BWR vibration test plan, and the vibration test data obtained from prototype tests of the reactor internal structures of eight BWR power plants was included, as a GE proprietary report, in Amendment 19 to the Quad Cities FSAR. This information provided a greater insight into the understanding of how comprehensive and complete the BWR internals vibration testing program was. [3.9.9]

This information provided the evidence to show that for normal steady-state balanced flow conditions the vibration of the BWR reactor internals is well within the conservatively established criteria limits. Neither large dimensional changes in the internals (as in changing vessel diameters from 188 to 205 to 224 to 251 inches), nor extremely small dimensional changes, as in going from Dresden Unit 2 to 3, result in any appreciable increase in the vibration response of the structure. This is the result of the low-flow velocities and the small exciting forces inherent in a BWR.

This information also showed that for short-term transient unbalanced flow conditions every plant tested to date meets the ASME Code fatigue usage criteria. However, as a precautionary measure, GE has recommended that procedural controls be imposed on all jet pump BWRs to avoid unbalanced flow conditions regardless of how low the measured vibration amplitude is for this transient, until a more comprehensive understanding of the phenomenon can be reached.

It should be emphasized that the purpose of the BWR internals vibration testing program was not to confirm fabrication and construction matters (which are strictly quality control matters), but to test a representative sample of internal components in order to gain confidence that their structural integrity will not be violated as a result of long-term vibration during the lifetime of the plant.

Successful vibration tests conducted on Dresden Units 2 and 3 are the necessary technical evidence to support the vibration integrity of the identical Quad Cities reactor design. Evidence from Dresden Unit 2 has been confirmed by tests on the Dresden Unit 3 reactor internals. Therefore, the proof of the adequacy of the design of the Dresden/Quad Cities generation of reactors has been presented and further additional confirmation of these conclusions was obtained during startup tests at Quad Cities.

As recommended by the Advisory Committee for Reactor Safeguards confirmatory vibration tests were conducted on the Quad Cities reactor vessel internals. This program supplemented the data obtained from the Dresden tests which were designed in accordance with GE's criteria for testing a "first-of-a-kind" plant. The vessel internal components monitored were the same as those tested on Dresden Unit 3. Specifically, the following components were tested: [3.9.10]

A. Shroud — the horizontal displacement of the shroud was measured.

B. Jet Pump Assembly Riser Pipe — the strain in the riser braces for two jet pump riser pipes was measured.

C. Jet Pumps — the horizontal radial motion of two pumps with respect to the reactor pressure vessel was measured.

The vibration rates of the various reactor internal components instrumented were detected by sensors mounted directly on those components. The vibration amplitude signals from these sensors was amplified and displayed by an oscillograph-type recorder and also recorded on magnetic tape.

These tests were performed as part of the power ascension program at 50, 75 and 100% power. Vibration measurements were monitored during steady-state and pump trip conditions. In addition, measurements were monitored during transient conditions when changing from one test point to the next.

The Flow Induced Vibration response of the reactor internals with Increased Core Flow to 108% of rated was evaluated with the first application of the Extended Operating Domain and Equipment Out-of-Service for Quad Cities. Increased vibration amplitudes evaluated from the increased core flow analysis were well below the acceptance criteria because of the large margin to the acceptance criteria in the original design. The methods and assumptions along with the results of that analysis[23] were outlined in the GE proprietary document. [3.9.11]

As part of the steam dryer replacement program, the Quad Cities Unit 2 dryer and main steam lines were instrumented for the purpose of measuring the pressure loads acting on the dryer. Structural analyses were performed to demonstrate the adequacy of the replacement dryer design using predicted loads based on main steam line strain gage measurements obtained during startup with the replacement dryer. The results determined that the replacement dryer satisfies 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

As described in Section 3.9.1.1, Quad Cities had no ASME Code Class 1, 2 or 3 designed piping systems due to the limited nature of the ASME code edition effective when the Quad Cities construction permits were issued.

The following subsections have been organized to provide a discussion of the design for 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 have been designated as safety Class I are either vital to safe plant shutdown or systems and 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. [3.9.12]

3.9.3.1.1 Reactor Pressure Vessel and Supports

The reactor vessels at Quad Cities Station Unit 1 and 2 are 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. [3.9.13]

Stabilizer brackets, located below the vessel flange, are connected to tension bars with flexible couplings. 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 Quad Cities reactor pressure vessels were designed according to the ASME Code, Section III, 1965 Edition, including the Summer 1965 Addenda. Applicable code cases and exceptions to the Summer 1965 Addenda are described in Section 3.2. [3.9.14]

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, and by special requirements and standards set forth to provide safety assurance in the event of specific occurrences not covered by the various codes. [3.9.15]

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 and vessel supports. [3.9.16]

The applicable loads for the reactor vessel and supports 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.

T = Thermal load on containment, reactor vessel, and internals due to LOCA.

E = Operating basis earthquake (OBE) load, ground horizontal g=0.12. vertical g=0.08.

E´ = Design basis earthquake (DBE) load, ground horizontal g=0.24, vertical g=0.16.

L = Wind live load beyond normal building code requirements.

Following are the load combinations used for the reactor vessel and vessel supports.

Reactor Vessel and Primary Internals [3.9.17]

D + E Stresses which occur as a result of the maximum possible combination of loadings encountered in operational conditions are within the stress criteria of ASME Code, Section III, Class A Vessel.

D + E´ The primary, and primary plus secondary stresses take 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 + T Primary stresses are within the stress criteria of ASME Code, Section III Class A. The primary and primary plus secondary stresses are examined and take 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.

For the reactor vessel, primary membrane stresses have been limited to 90% of the material yield strength. [3.9.18]

Reactor Primary Vessel Supports [3.9.19]

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.

A stress limit of .85 f'_c is used in calculating loads involving DBE, pipe rupture, and tornado. Combining a 1.0 load factor with these loads is acceptable for two reasons. The first and primary reason is that the remote probability of the occurrence of these loads classifies them as ultimate loads. The second reason is that the concrete design is conservative because in bending analyses compression reinforcing in the opposite face is not considered, and in most cases reinforcing is under design, thus the concrete is stressed at a lower level than the steel. This means yielding of the section is controlled by the tensile reinforcing stress which is limited to .90 Fy. There is still an additional factor of safety between this value and the ultimate strength of the reinforcing steel.

D + H + R + E´ Stresses do not exceed: [3.9.20]
- 115 ksi for bolt tensile stress.
- 66 ksi for bolt shear stress.

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. [3.9.21]

3.9.3.1.1.3 Design Evaluation

Results of the design evaluation of the reactor pressure vessel were submitted as Appendix C to Amendment 14 of the Quad Cities FSAR.

In addition to the seismic dynamic analysis, the upper truss reactions were analyzed for jet forces. Loads due to the various jet force combinations were derived for each truss member. As in the other seismic analyses, double earthquake criteria were reviewed and met. [3.9.22]

The RPV support was originally evaluated for an OBE shear force and overturning moment of 1370 kips and 42,500 kip-foot respectively[10]. These forces are considerably higher than the revised OBE forces obtained from the new analyses performed as part of the core shroud repair (528 kips shear and 7,460 kip-foot moment)[11]. Therefore, the existing design evaluation shown below remains conservative.

For the vessel support skirt, the maximum primary general membrane stress ($S_{\{m\}}$) for the design basis earthquake is within the ASME Code Section III allowable. When jet forces are added to the design basis earthquake, the maximum primary general membrane stress is also within the ASME Code Section III allowable (.9 times yield strength). The maximum primary plus secondary stress ranges for the support skirt are given in Figure 3.9-1. The stresses where the skirt is attached to the vessel bottom head are less than $3S_m$ = 80,000 psi and in the support skirt itself are less than $2S_y$ = 87,200 psi at 450°F. Maximum temperature differences during startup and shutdown are shown in Figures 3.9-2, 3.9-3 and 3.9-4. Figure 3.9-2 provides the node point locations in the reactor vessel skirt region. Startup and shutdown temperatures for these node points are plotted in Figures 3.9-3 and 3.9-4, respectively. [3.9.23]

The support skirt flange material is hull steel (modified SA-302GR.b) with chemistry per paragraph 3.4.1 of MIL-S-24094 GR.HT, 9/21/64. The transition piece which attaches the skirt to the vessel is hull steel for Unit 1 (same as skirt flange) and is SA-533 GR.B CL.1 for Unit 2. Both materials meet ASME Code Section III requirements.

The horizontal shear at the reactor skirt base is transferred to the top flange of the ring girder by 60 high-strength bolts in a friction-type connection. Oversized holes for installation of these bolts will not reduce the friction force available for restricting the horizontal shear because the proof load on the bolts is greater than the maximum tension to which any bolt will be subjected under all load conditions. [3.9.24]

The horizontal shear at the ring girder base is transferred to the top of the concrete pedestal by 120 anchor bolts in a bearing-type connection. The effects of the oversize holes on the design of the connection are eliminated by filling the space between the anchor bolts and ring girder with aluminum epoxy.

The RPV stabilizer system is designed with a spring constant of 40,000 kips/in. Table 3.9-3 lists the allowable stresses.

The total frictional force due to a coefficient of friction of 0.15 and a proof load of 480 kips per bolt is 4320 kips or 3.15 times the original OBE shear load of 1370 kips. However, even if the coefficient of friction is assumed zero, the bolts as bearing-type connections could resist a total horizontal shear of 6630 kips or 4.85 times the OBE shear load of 1370 kips. Therefore, the high-strength bolt connection of the RPV skirt flange to the ring girder is more than adequate for the expected design load. [3.9.25]

With regard to the other vessel support elements, allowable stresses have been tabulated for the concrete reactor support pedestal and the reactor skirt ring girder anchor bolts. Tables 3.9-4 and 3.9-5 present allowable stresses for the concrete pedestal and ring girder anchor bolts, respectively, in accordance with the design basis load combinations from Section 3.9.3.1.1.2. The maximum stresses on the concrete reactor support pedestal and the reactor ring girder anchor bolts are less than the allowable stresses. [3.9.26]

In addition to the aforementioned summary of design basis maximum stresses for vessel support elements, CECo was requested to provide similar results for an additional load combination. The requested load combination included concurrent dead load, postulated thermal load, pipe break jet load, and DBE. Results were requested for the vessel concrete pedestal, ring girder anchor bolts and ring girder high strength bolts. Maximum stresses under the requested load combination are summarized for these vessel support elements in Table 3.9-6. [3.9.27]

Such a loading combination was not included among the original design criteria because it was not considered to be a credible circumstance. The load combination of dead load, postulated thermal load, pipe break jet load, and OBE is considered to be conservative and was acceptable on the Dresden Unit 2 and 3 applications. There are numerous factors which must all be considered to act simultaneously in the combination of loads considered with the DBE including dead loads, operating pressure loads, and live loads; in addition to the thermal expansion forces which would not necessarily exist simultaneously with the largest jet force. If all of these factors had to be considered to act simultaneously with just the right break location occurring, as well as the peak DBE load occurring in the exact direction where the jet load and DBE loads are directly additive, then the criteria that would be used would assure that a safe shutdown could be accomplished. Such assurance would be derived by considering the energy absorption capability of the structures versus the energy input and would entail an extensive series of assumed conditions. The loading criteria adopted gives full consideration to safety and is conservative, but also remains within practical bounds.

By virtue of the fact that the allowable stresses have been met in Table 3.9-6, it can be concluded that such a safe shutdown criterion would be met.

A description of the seismic analysis for the vessel support system is provided in Section 3.7.2.1. It should be noted that the vessel does not rock under earthquake loads because the tension force that results from an earthquake does not exceed the clamping force of the bolts holding the skirt to the ring girder. Thus, the plates cannot separate and there can be no stretch of the bolts or increase in their tensile load. [3.9.28]

In the case of the anchor bolts connecting the ring girder to the concrete pedestal, the maximum stress level in any of these bolts due to OBE loading is 0.65 ksi. This produces a maximum anchor bolt stretch of 0.006 inches. The effect of this stretch is to rotate the reactor vessel about its base and cause a horizontal deflection of the vessel above its base. Taking a point halfway up from the base to the lateral support at the shield wall (which corresponds approximately to elevation 630 feet), the horizontal displacement caused by anchor bolt stretch is calculated to be 0.0014 inches. At this same point the displacement due to an earthquake is noted as 0.160 inches. Since the displacement due to anchor bolt stretch is less than 1% of that due to an earthquake, its influence on rocking response is considered negligible.

The materials for the following Class I component supports are as follows: [3.9.29]

Drywell	4000 psi reinforced concrete
Reactor Vessel	4000 psi reinforced concrete
	ASTM A-316 steel
Torus	4000 psi reinforced concrete
	ASTM A-516 GR. 70 steel
Reactor Recirculation Pumps	ASTM A-36 steel
Reactor-Drywell Stabilizer and Support	ASTM A-36 steel
	ASTM A-307 GR. B steel
	ASTM A-201 GR. B steel
	ASTM A-283 GR. C steel
	4000 psi reinforced concrete
Modulus of Elasticity of Concrete Steel	Minimum 3,500,000 psi
	29,000,000 psi

In general, thermal gradients do not govern in the design of Class I component supports. For example, the reactor-drywell stabilizer and the reactor recirculating pumps are supported utilizing hydraulic or spring-type snubbers which are specifically designed to accommodate relative thermal movement. A thermal gradient does exist between the support components which is automatically compensated by the snubber.

An area with significant thermal gradient consideration is the joint between the reactor vessel skirt support ring girder and the concrete support pedestal. The general criteria used for determining the adequacy of this support structure is dependent on the various loading combinations.

For a load combination of operating loads and the operating basis earthquake (0.12g ground acceleration) allowable stresses are the AISC values without the usual increase for earthquake loads. For the same operating loads coupled with a DBE (0.24g ground acceleration), an allowable of 90% yield is used. For the special case where the operating loads include initial start-up thermal transient loads an allowable of yield stress is set as the stress limit.

For the design of the ring girder supporting the RPV and the anchor bolt connections to the concrete pedestal, the maximum thermal gradient used was conservatively 80°F between these components. Of this 80°F, one half is considered to be included in the operating loads and the other half is considered to be special transient loads that could occur during the initial startup phase of plant turnover from construction of the reactor. For the maximum case, i.e., the maximum gradient and the DBE, the stress in the critical component is approximately 84% of yield. Hence, it can be seen that even for this extremely unlikely combination of loadings the stresses are satisfactory. It is emphasized that the code allowable can be increased by a factor of 1.3 when considering earthquake loads. This extra code allowance was not used, as mentioned previously.

The lower portion of the reactor vessel pedestal did not consider temperature gradients in the design since sufficient openings exist to permit air movement and the large mass will heat and cool relatively uniformly.

Support details are provided on the following sketches, Figures 3.9-5 through 3.9-9.

In summary, the purchase specifications were not used as the vehicle to assure adequate design, but specific reviews of specific components do provided assurance the seismic criteria has been met.

Details of the ring girder connection and concrete pedestal are provided in Figure 3.9-6.

3.9.3.1.2 Mechanical Equipment

The following section describes the acceptance criteria, loading conditions, and evaluations applicable to mechanical equipment (pumps and valves) at Quad Cities 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 and the basic codes listed above for Class I. [3.9.30]

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 mechanical equipment original design. [3.9.31]

D + T + H + E — Stresses remain within code allowable. ASME Code, Section III Class C and TEMA C, shell side. ASME Code, Section VIII. TEMA C on tube side.

D + T + H + E´ — Same as Section 3.9.3.1.1.2 for P + D + T loading condition.

3.9.3.1.2.3 Evaluation

For equipment subject to Mark I hydrodynamic loads (see Section 6.2 for discussion of the nature and origin of these loads), equipment nozzle connections were remodeled as anchors for computer analyses of the attached piping. Stresses at equipment nozzles were computed using the governing load combinations listed in Table 3.9-11. [3.9.32]

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 3.9-11 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)[4], 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 3.9-11. 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 Quad Cities Station. This information has been organized under four distinct phases of the design process:

1. Original design,
2. 79-14 Program,
3. Mark I Program, and
4. 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 or Extended Power Uprate Program, have replaced the original evaluations for many of the Safety Class I piping systems. For piping not reanalyzed under the 79-14 Program (Section 3.9.3.1.3.2), the Mark I Program (Section 3.9.3.1.3.3), or the Extended Power Uprate Program (Section 3.9.3.1.3.4) the original design basis remains applicable. Revision status and responsible design organization are now tracked by computerized methods. [3.9.33]

3.9.3.1.3.1.1 Acceptance Criteria

Piping at Quad Cities was originally designed in accordance with the USAS B31.1 Code, 1967 Edition. Design criteria are discussed further in the following subsections. [3.9.34]

3.9.3.1.3.1.2 Loading Conditions

Using the nomenclature of Section 3.9.3.1.1.2, following are the load combinations applicable to Class I original piping design. [3.9.35]

D + E	Allowable USAS B31.1 stresses
D + E´	Yield Stress or special analysis showing strains limited to preclude failure by deformation which would compromise any emergency core cooling system (ECCS) components or prevent safe shutdown.

Class II piping, normally designed in accordance with USAS B31.1, was considered in a more stringent manner when the Class II portion joins a Class I section of piping. In such cases the Class I portion that was analyzed dynamically was extended to include the Class II portion sufficiently to assure that the Class II section could not adversely affect the Class I integrity. [3.9.36]

3.9.3.1.3.1.3 Evaluation

The Class I piping systems, as noted previously, were analyzed to assure compliance with the criteria by one of two methods: dynamic or force-deflection curves. Dynamically analyzed systems utilized the computerized response spectra method. In this method the piping was modeled by a series of discrete masses interconnected by weightless springs. [3.9.37]

The system was then subjected to a translatory motion in each of the three mutually perpendicular directions of the global axis system. The program utilized the appropriate floor response spectra to determine appropriate spectral accelerations after computation of the mode frequencies and shapes. A 0.5% damping factor was used on piping. For each mode the displacements and inertia forces were determined and the inertia forces of each mode were used as an external loading condition. The total combined modal results were obtained by taking the square root of the sum of the squares for each parameter, i.e., moments, shears, and displacements. In addition to the items noted, the computer program accounted for the effects of curved members and elbows by use of stress intensification factors which are functions of the pipe diameter, thickness, and bend radius.

For Class I systems, the boundaries of the piping system model used in the seismic analysis extended well beyond the stress analysis boundaries set by the first normally closed valves.

This was done to provide confidence that the dynamic loading influence of the Class I piping outside of (but attached to) the critical Class I portion of the system model is adequately accounted for.

All dynamic analyses used 1/2-percent of critical damping for both the OBE and DBE except for the standby gas treatment system, where 1-percent of critical damping was used. [3.9.38]

Buried pipes are treated in a special manner that considers the piping building connections to move with the building and be acted upon by a series of "soil springs" with terminal movement equal to soil deflections. These soil movements are considered in each of two horizontal directions to determine the worst effect on the piping. Thus, it is assured that such buried pipes can withstand seismic events. [3.9.39]

The type of seismic analysis given a particular system was not related to the intra or extra-containment location of the piping. The selection of analyses depended on the pipe size and on whether the piping is buried in the soil or located inside a building structure. [3.9.40]

The type of seismic analysis used for Class I piping located within the turbine building, reactor building, service building complex depends on the nominal pipe size of the system. Line sizes 10-inches and larger were given a detailed dynamic analysis using a computer based response spectra method incorporating a modal analysis technique. Line sizes 8 inches and smaller were analyzed by a set of seismic design curves (also known as Blume curves). None of the preceding analyses considered relative movements between building structures. The turbine building, reactor building, and service building are actually one single structure. No relative movements exist between these buildings at points where piping passes between them.

For Class I piping buried in the soil, a static flexibility type analysis was performed. Dynamic building and soil movements were applied to the piping and the resisting effect of the soil on pipe movement is included. Rigid anchors were assumed at points where the piping enters building structures, to simulate the fact that the piping is embedded in the concrete wall or floor at the point of entry.

<u>Detailed Dynamic Analysis</u>

The piping to be analyzed was modeled by a series of discrete masses interconnected by massless elastic elements. The elastic elements have the stiffness characteristics of either straight or curved pipe elements.

No restraint credit was taken in the seismic analysis for variable and constant support hangers or for sliding supports.

Where possible, the seismic analysis included all significant sized piping between anchor points. If only a portion of the system was analyzed, the analysis terminal points were modeled so that the loading influence of the unanalyzed portions of the piping are accurately depicted. The weight of valves and other inline components was included in the analysis. Where valves and other inline components could introduce significant rotary inertia, this was also considered in the piping analysis. Piping cannot necessarily be considered anchored at equipment nozzles. The mass of equipment and stiffness of equipment foundations may participate significantly in the response of the piping system. This effect was considered in the formulation of the analytical model.

The piping model was assumed to be mounted in a rigid frame or box. The rigid box was subjected to a translatory motion in each of the three mutually perpendicular directions of the global axis system for the problem.

Stress intensity was based on both probable maximum moment distribution and on absolute sum moment distribution according to the formula:

$$S_E = \sqrt{S_b^2 + 4S_t^2} \quad = \text{Resultant Stress}$$

$$S_b = i\frac{M_b}{Z} \quad = \quad \text{Bending Stress}$$

$$S_t = \frac{M_t}{2Z} \quad = \quad \text{Torisonal Stress}$$

where:

I = stress intensification factor
M_b = resultant bending moment
M_t = torisonal mement
Z = section modulus of pipe

Before the seismic loads were used in the combined stress analysis, the worst horizontal excitation results were directly combined with the vertical excitation results.

In cases where seismic stresses were relatively low in a system and the seismic deflections were large, i.e., on the order of 4 or more inches, clearances were checked to ensure that the piping will not be damaged by striking any nearby structure, component, etc.

Combined Stress Analysis Per USAS B31.1

Seismic stresses, dead weight stresses and pressure stresses were combined in accordance with 102.3.2(d) of B31.1 - 1967. That is:

$$\sigma = \frac{A_f}{A}P + |S_{EW}| + S_{EE}$$

$$S_{EE} = \text{The larger of } |S_{EX}| + |S_{EY}| \text{ or } |S_{EZ}| + |S_{EY}|$$

$$\sigma = \text{Total combined longitudinal stress}$$

$$A_f = \text{Flow cross sectional area of pipe}$$

$$A = \text{Metal cross sectional area of pipe}$$

$$P = \text{Internal design pressure}$$

$$S_{EE} = \text{Combined seismic stress}$$

$$S_{EX} = \text{Stress due to x direction earthquake excitation}$$

$$S_{EY} = \text{Stress due to y direction earthquake excitation}$$

$$S_{EZ} = \text{Stress due to z direction earthquake excitation}$$

$$S_{EW} = \text{Stress due to dead weight}$$

The weight and seismic stresses used in this equation include the stress intensification factors for piping components given in Appendix D of B31.1. Because the stress intensifications associated with unreinforced branch connections are large, it was necessary in some cases to add a reinforcement pad at the branch point. This is an advisable procedure when the piping has adequate restraint to maintain seismic stress at a relatively low level throughout the system except at the branch point.

Operating Basis Earthquake

The value of combined stress, sigma, for the OBE as calculated from the preceding equation shall not exceed 1.2 S_h, where S_h is the allowable stress per Table A-2, B31.1. For the standby gas treatment system (SBGTS) no allowable stress was given in B31.1 for the ASTM A211 pipe material. For this system the value of S_h was taken from USAS B31.3, Table 302.3.1B. The 20% increase in S_h for occasional loading is allowed in Section 102.2.4 of B31.1.

Design Basis Earthquake

The DBE criteria was met by demonstrating that a safe shutdown of the reactor could be made during the DBE. For the safeguard piping this would imply no loss of function. In general, if all stresses are within the yield stress of the piping material it is obvious that no loss of function can occur.

In some cases the conservative approach of combining stresses directly in the preceding equation resulted in an apparent overstress. However, when these points were analyzed by superposition of weight and seismic at the loading component level, they were found to meet the stress criteria.

Simplified Dynamic Analysis (Seismic Design Curves)

For Class I systems sizes 8 inches and smaller, a simplified dynamic analysis was used.

The simplified dynamic analysis was presented in the form of design curves that were used to:

A. Select piping spans whose first period was removed from the period of the predominate peak (or peaks) of the floor response spectra for the building structure,

B. Assure that seismic stresses were not greater than the allowable per the design stress criteria given above,

C. Assure that seismic deflections were not large enough to cause damaging contact between pipe and any nearby structure, equipment, etc., and

D. Determine seismic restraint design loads.

Period versus Span Curves

These curves show the first (fundamental) period of a span of piping versus the length of the span. The periods plotted were for a beam with a simply supported end and a fixed end. Regions of span are shown where the piping is considered "rigid," "flexible," and "resonant." Near the center of the resonant region is the period corresponding to the predominant building period. Piping having span periods at or near the predominant building period could theoretically be driven to a very large response. Obviously, one design objective was to avoid resonance.

The rigid, flexible, and resonant regions were selected by:

Rigid $\frac{T_b}{T_p} \geq 2.0$

Flexible $\frac{T_b}{T_p} \leq 0.7$

Resonant $0.7 \leq \frac{T_b}{T_p} \leq 2.0$

where

T_b = Predominant Building Period
T_p = First Period of Pipe Span

The building seismic analysis floor response spectra show that T_b is about 0.16 seconds. Therefore:

Rigid $T_p \leq \frac{.16}{2}$

$T_p \leq .08$ sec.

Flexible $T_p \geq \frac{.16}{70}$

$T_p \geq .23$ sec.

The allowable spans were:

Size (inches)	Maximum Span Rigid Range (feet)	Minimum Span Flexible Range (feet)
1/4	6 1/2	11 1/2
1/2	8	14
3/4	9	16
1	10	18
1 1/2	12	21
2	13 1/2	23
2 1/2	15 1/2	27
3	17	30
4	19	33
6	23	40
8	26	44

Seismic Deflection versus Span Curves

These curves showed the first mode seismic deflection of a simply supported beam representing the pipe span. The response was based on the ground acceleration spectra curve for the OBE recommended by John A. Blume & Associates for the Quad Cities Station site. The input accelerations were selected by using the periods shown on the period versus span curves.

A significant feature of these curves was a line showing the deflection needed to produce a bending stress of 3700 psi in the pipe span. To assure that code allowable stresses were met when seismic loading was combined with other appropriate loadings, the seismic deflection due to the OBE was never allowed to exceed that shown by the 3700 psi curve. In addition, to protect against damaging contact between pipe and surrounds, seismic deflection greater than 2 inches was not allowed.

Seismic Restraint Load versus Span Curves

These curves gave the seismic restraint reactions for a span. Again the model was a simply supported beam and the response was based on the ground response spectra. The input accelerations were selected from the response spectra using the periods shown on the period vs. span curves.

To account for the continuity of the piping across a restraint attachment point, the reactions for all piping spans supported from a restraint were added.

Valves and Other Concentrated Weights

All concentrated weights such as valves, heavy flanges, etc, were restrained against lateral motion if possible. If this could not be done, the span containing the valve, flanges, etc., was reduced to less than one-half of the maximum rigid span.

Support reactions for a span containing a valve were found in the conventional manner described above except that one-third of the concentrated weight was also distributed to the two restraints in proportion to the proximity of the concentrated weight to the restraints.

Elevations Above 579 Feet

The design curves were based on the response to the ground acceleration spectra. This assumed that the piping was physically located low enough in the building so that building amplification was of no consequence. However, at some higher elevation building amplification would cause a significant increase in the piping response. Elevation 579 feet (approximately at the centerline of the torus) was selected as the elevation below which the curves were directly applicable. For piping above 579 feet, having flexible spans, the design curve deflections and restraint loads were multiplied by three to account for amplification.

Operating Basis Earthquake and Design Basis Earthquake

The design curves were based on the accelerations associated with the OBE. The piping design criteria requires that Class I piping satisfy normal code stress requirements during an earthquake of this intensity. The 3700 psi seismic stress limit was selected to insure that normal code stress requirements of the design criteria would always be satisfied. To provide a seismic restraint design that was compatible with the piping design, for loading due to the OBE Class I seismic restraints were designed per Division 120 and 121 of ANSI B31.1.

A further requirement of the piping design criteria was to assess the effect of an earthquake of twice the intensity of the OBE. The design goal for the DBE was to maintain a safe shutdown capability for the nuclear energy system. Since the design curves are based on the OBE, all deflections, stresses, and reactions as determined from the curves were doubled to obtain DBE values. The allowable seismic stress for the DBE was obviously two times the OBE allowable, or 7400 psi. The maximum allowable seismic deflection, however, remained at 2 inches. The 7400 psi seismic stress limit assured that code stress requirements of the design criteria were met. To provide a seismic restraint design that was compatible with the piping design, for loading due to the DBE Class I seismic restraint was designed per Division 120 and 121 of B31.1, except that restraint stress did not exceed the yield stress.

The results of the original stress analyses of various piping systems are provided in Table 3.9-7. This summary provides the maximum values obtained on the various sections and includes code intensification factors. The seismic stresses were based upon a reanalysis using the combined Golden Gate Park and Housner spectrum for piping sizes greater than 8 inches in diameter. [3.9.41]

It should be noted that the B31.1 Code, to which piping was originally designed, made no provision for emergency condition loadings. B31.1 allowed primary stresses of 1.2 times the allowable stress at operating temperature for loads of short duration. For the Quad Cities recirculation system the longitudinal stresses due to pressure, dead weight and maximum (DBE) seismic inertia were compared to 1.8 times the allowable stress at operating temperature. The 1.8 value used in the criteria is a 50% increase over the code allowable. This is consistent with B31.7 and Code Case 70 of B31.7. The loadings mentioned in the above criteria are equivalent to the loadings used for calculated primary stresses in Equation 9 of B31.7. The 50% increase in allowable stress is allowed by Code Case 70 for emergency conditions. [3.9.42]

3.9.3.1.3.2 79-14 Program

An extensive program was implemented to fulfill the requirements of NRC IE Bulletin 79-14. The bulletin required the following: [3.9.43]

A. Walkdown safety-related piping 2 1/2 inches and larger,

B. Evaluate and reconcile nonconformances,

C. Evaluate operability,

D. Reconcile as-built versus as-designed, and

E. Restore system to FSAR limits.

Under these requirements all affected piping systems were reanalyzed using original design criteria.

The final 79-14 piping evaluation was performed using two methods of analysis. In the first, large bore systems, which were originally analyzed by computer (10 inches or larger), were reanalyzed by computer. In the second, smaller diameter piping systems (2 1/2 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 are 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. [3.9.44]

Following the initial operability evaluations for 79-14, all systems and restraints were restored to original design limits. [3.9.45]

3.9.3.1.3.3 Mark I Program

The primary containments for the Quad Cities Station Units 1 and 2 were designed, erected, pressure-tested, and N-stamped in accordance with the ASME Code, 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 SRV operation. [3.9.46]

The new loads were identified by the NRC as a generic open item for utilities with Mark I containments. To determine the magnitude, time characteristics, etc., of the dynamic loads in a timely manner and to identify courses of action needed to resolve any outstanding concerns, the utilities with Mark I containments formed the Mark I Owners Group. The Mark I Owners Group established a two-part program consisting of: the Short-Term program (STP) to demonstrate safety-to-failure margins of at least 2.0 to justify continued operation, which was completed in 1976, and a submittal of the "Mark I Containment Program Load Definition Report" (LDR)[3], PUAAG[4], and supporting reports on experimental and analytical tasks of the Long-Term Program (LTP). 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 objective of the LTP was 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 Quad Cities Units 1 and 2. Differences between Quad Cities Units 1 and 2 exist primarily in the torus attached piping (TAP) systems and their corresponding branch connections. Furthermore, the containments (i.e., drywell, wetwell, vent system, etc.) for Quad Cities Units 1 and 2 are very similar to the containments for Dresden Units 2 and 3. Since the suppression chambers at Quad Cities Station are similar to those at Dresden Station, the subscale and fullscale tests performed for Dresden are applicable to Quad Cities Units 1 and 2.

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. [3.9.47]

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].

For 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, with the exception of MS and SRVDL piping inside the drywell, for which B31.1 Code rules apply.

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 [4]. 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 Code Subsection ND, and are summarized in Table 3.9-9.

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 and is addressed in Section 3.8.

Main Steam and Safety Relief Valve Discharge Lines Inside Drywell

The acceptance criteria for the main steam (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 DBE 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 Quad Cities TAP analysis is based follow the rules contained in the ASME Code, Section III, Division 1 up to and including the 1977 Summer Addenda for Class 2 piping. 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 Code. Table 3.9-12 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 [3.9.48]

The wetwell SRVDL is analyzed as ASME Code Class 3 piping, 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 (including the T-quencher) the governing load combinations are shown in Table 3.9-8. The appropriate Service Level is identified for each combination; for the piping combinations, the applicable ASME Code equation is also provided.

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 excepts that the SRVDL have 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 through 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 since they are 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

$P + DW <= 1.0\ S_h$

$P_o + DW + [OBEI^2 + SRVD^2 + SVD^2]^{1/2} <= 1.2\ S_h$

$P_o + DW + [SSEI^2 + SRVD^2 + SVD^2]^{1/2} <= 1.8\ S_h$

Secondary stresses plus pressure and dead weight

$TE + OBED <= S_A$

$TE + SSED <= S_A$

$TE + OBED + P + DW <= S_A + S_h$

$TE + SSED + P + DW <= S_A + S_h$

where

S_h = Basic material allowable stress at maximum (hot) temperature from the Allowable Stress Tables 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 loads,

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 TAP are presented in Table 3.9-11.

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 and are consistent with the methods used to develop the loading functions defined in the load definition report. For those loads considered in the original design but not redefined by the load definition report (LDR)[3], either the results of the original analysis were used or a new analysis was performed, based on the methods employed in the original plant design. [3.9.49]

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 inside the wetwell includes the performance of a structural analysis of T-quencher, 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 line supports due to the loads was considered.

Table 3.9-10 summarizes the maximum Class 3 wetwell SRVDL 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, 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-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, including 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[4]. 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 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 [6] 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, including Quad Cities 1 and 2. The resulting cumulative usage factors are below 0.5, demonstrating that further plant unique fatigue evaluations are not warranted. Therefore, the Quad Cities Unit 1 and 2 TAP is qualified based on this generic evaluation.

The maximum piping stress for large bore TAP lines and small diameter torus internal lines are adequate for the loads, load combinations, and acceptance criteria limits in NUREG-0661[5] and PUAAG[4].

In 1986, inconsistencies were identified between the as-built configuration and analysis documentation for a Dresden Unit 2 TAP system. In order to determine if similar inconsistencies existed on other Mark I lines, the Piping Configuration Verification Program was initiated. The first task of the program consisted of walkdowns of the large bore piping at Dresden and Quad Cities. The walkdown information was then compared to the analysis documentation, and inconsistencies were documented. The inconsistencies were reviewed in detail to determine if they were actual discrepancies. All discrepancies were resolved by one of two approaches: [3.9.50]

1. The field condition was changed to match the configuration in the analysis documentation; and

2. The piping and/or supports were reanalyzed to show they were acceptable as-is.

In either case, the appropriate calculations and drawings were revised to accurately represent the as-analyzed and field condition resolutions.

All of the discrepancies have been resolved and FSAR compliance has been demonstrated. Root causes have been determined and actions to prevent recurrence are in effect. Based on actions taken, similar discrepancies are not expected to recur.

3.9.3.1.3.4 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.

3.9.3.1.3.4.1 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 envelop or equal the power uprate conditions and no further review is performed.

For minor changes resulting in a "change factor" between 1.0 and 1.05 (or 5%), the increase 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 are not limited to, the industry practice of enveloping 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 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. Flow changes were found to be significant only for the Main Steam and Feedwater/Condensate systems.

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.

Furthermore, per industry practice piping systems that have operating temperatures less than 150°F did not require evaluation for thermal change effects.

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 increases of 5% were considered minor and acceptable by engineering judgement 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 increased by 15%, but the piping component analysis showed a 20% margin to allowable, the 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, a 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 system 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 the torus water) is provided in more detail below, by component type:

Pipe Stress

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.

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 as 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 a 1/2 inch "cushion" provided on each end of the range. Therefore a minimum of 1/2 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 1/2 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 increase 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.

Displacements at Interferences

Some piping models have displacement checks at certain locations where there may be interferences with nearby structures (i.e., 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 a 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 Rector 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. Since 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 effect 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.

3.9.3.1.3.4.2 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 indicates 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 B31.1 – 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 nor 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 loads. The EPU evaluation approach for the TSV loads is based on an acceptance criteria for the 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, there 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, thermal, and safety relief valve (SRV) and pipe break 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 time relationship between the significant loads resulting from TSV, SRV discharge and pipe break events, no combination of these loads is required.

To demonstrate piping pressure boundary integrity subsequent to a TSV closure event, the piping, pipe supports and supporting structures were evaluated for the following additional loading combinations (LC):

Piping:

LC 1 Dead Load + Pressure + TSV Loads
LC 2 Dead Load + Pressure + $[(\text{TSV Loads})^2 + (\text{SSE Loads})^2]^{1/2}$

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 (LC 1) and 3% when combined with seismic (LC 2) in accordance with guidance contained in Reg. Guide 1.61.

For evaluation of the supporting drywell steel, where supports from different main steam lines are attached to the same drywell steel, the TSV 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.

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 + $(SSE^2 + TR^2)^{1/2}$	FAULTED 1.60 x AISC Allowable <0.95 x Fy**
EXPANSION ANCHOR BOLTS	
DW + TH + TR	SAFETY FACTOR = 4
DW + TH + $(SEE^2 + TR^2)^{1/2}$	SAFETY FACTOR = 2
PIPE SUPPORT COMPONENTS	
DW + TH + TR	ASME LEVEL C
DW + TH + $(SSE^2 + TR^2)^{1/2}$	ASME LEVEL D
PIPING	

APPLICABLE TSV LOAD COMBINATIONS	ACCEPTANCE CRITERIA
DW + P + TR	ASME Level C
DW + P + $(SSE^2 + TR^2)^{1/2}$	ASME Level D

*TR = Transient Loads such as TSV
** 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 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.

3.9.3.2 Pressure Relief Devices

Discussion of 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 Quad Cities design (Section 3.9.3.3.1) and for the Mark I program (Section 3.9.3.3.2).

Original design criteria remain applicable for supports not redesigned under the Mark I Program.

3.9.3.3.1 Original Design

The original design basis for hanging and supporting the piping systems followed the requirements of USAS B31.1, Part 5 of Chapter II, insofar as hanger spacing is concerned. [3.9.51]

The materials used in the fabrication of hangers, anchors and supports was equal to or exceeded 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 conditions.

The materials used in the fabrication of anchors, guides, and restraints for dynamic loads due to pipe rupture and design seismic loading is equal to or exceeds the design criteria that the material shall be within 90% of the material yield stress for 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 meet the requirements of ANSI B31.1.0.

Plant operability procedures provide for routine inspection of fluid level in all hydraulic units during scheduled outages.

The design approach used for hangers, restraints, snubbers, etc. for statically analyzed piping systems is the same as that used for dynamically analyzed systems.

3.9.3.3.2 Mark I Program

Safety Relief Valve Discharge Lines Supports Inside Wetwell [3.9.52]

Acceptance criteria for the stress analysis of the wetwell SRVDL piping are based on the Plant Unique Analysis Applications Guide (PUAAG)[4]. Stress allowables are based on ASME Code, 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 is in accordance with the structural design specification and is 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 3.9-15. The appropriate service level is identified for each combination; for the piping combinations, the applicable Code equation is also provided. Included in this table are some original design basis load combinations, required for completeness of the stress evaluations.

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 3.9-16. All critical stress values are within the AISC Code requirements, and therefore meet the requirements of the ASME Code, Subsection NF.

Safety Relief Valve Discharge Lines Supports Inside Drywell

The acceptance criteria for the SRVDL vent line supports are in accordance 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 3.9.3.1.2.2):

Upset conditions

$F_{DW} + F_{TE} + [F_{OBEI}^2 + F_{OBED}^2 + F_{SRVD}^2 + F_{SVD}^2]^{1/2}$

Emergency conditions

$F_{DW} + F_{TE} + [F_{SSEI}^2 + F_{SSED}^2 + F_{SRVD}^2 + F_{SVD}^2]^{1/2}$

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 3.9-17.

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 3.9-18 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 the ASME Code, Section III, Subsection 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 the American Concrete Institute (ACI) Standard 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 3.9-11. Allowable stresses are given in Table 3.9-12. 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.

3.9.4 Control Rod Drive Systems

A discussion of the evaluation of the control rod drive (CRD) system is contained 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 a description of the physical layout of the reactor pressure vessel internals (Section 3.9.5.1), 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: [3.9.53]

A. Shroud, including tie rods with spring stabilizers,
B. Baffle 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 spargers,
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 horizontal 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 through 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 and on Unit 1 during the refueling outage in 1996. 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. 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). [3.9.54]

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 3.9.10 for the reactor vessel cut away isometric for illustration of the component arrangement.

The bottom of the shroud is welded to a rim on the baffle plate. The baffle plate outer diameter is welded to the reactor vessel and the inner diameter is supported by columns extending to the bottom head.

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 section 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 ferric, 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 (including support pieces), 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 thrusts of the seismic design.

The reactor fuel supports (see Figure 3.9-10) are the four-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. The control rods pass through slots in the center of the support piece. Each fuel support piece is removed before removal of the control rod with integral velocity limiter. [3.9.55]

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. 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. 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. Core plate wedges are installed on Unit 1 and Unit 2 in the space between the core plate and core shroud to assure that core plate sliding will not occur during a seismic event.

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 the10 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 a 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 is welded to the baffle plate on the lower end.

The hydraulic and operational effects of the jet pump design are discussed in Section 5.4.1.

Feedwater sparger integrity is discussed in Sections III.3.5 and II.3.2 of Amendment No. 5 for the Dresden Unit 3 Plant Design and Analysis Report Docket No. 50-249, which also included a discussion of the core spray sparger integrity. The following paragraphs, however, cover some of the features unique to the feedwater sparger. [3.9.56]

Four feedwater spargers are utilized in the reactor. Each sparger is approximately 70° in arc length and mounted to the inside reactor vessel surface. The sparger uses a triple-sleeve design; which is three concentric thermal sleeves, the innermost of which conducts feedwater to the sparger arms. Each sparger is supported by the thermal sleeve, and a bracket mounted to each end of the sparger. The arms are attached to the sleeve by a forged tee. The feedwater sparger is removable. The feedwater nozzle has had the carbon steel cladding bored out in order to reduce thermal stresses caused by differential thermal expansion between the stainless steel sparger and carbon steel vessel. The sleeves are welded to the nozzle. Tangential differential expansion is taken up by tangential slots cut in the bracket mounted to each end of the feedwater sparger. Pressure differentials, jet reactions, and earthquake loadings are all added; these stresses within the sparger are all within ASME Code Section III for Class A vessels. 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 in Amendment #5 for the Dresden Unit 3 Plant Design and Analysis Report.

The feedwater nozzle inner bore, the thermal sleeves, and the feedwater spargers were modified in 1980 on Unit 2 and in 1982 on Unit 1. The modifications consisted of removing (by machining) the cladding from the feedwater nozzles, boring the inside diameter of the safe-ends to accommodate the new feedwater sparger seal surfaces and installing the new design spargers. The modified thermal sleeve is a double seal/triple thermal sleeve, which has dual piston ring seals and an interference fit.

The new sparger/thermal sleeve design extends the service life of the feedwater nozzles by limiting the amount of feedwater leakage past the thermal sleeves, which will prevent thermal fatigue cracking in the nozzles and safe-end bores. Inspection intervals and methods were originally established in Table 2 of NUREG-0619, which stipulated a combination of visual inspections, liquid penetrant inspections and ultrasonic inspections.

Quad Cities performed the inspections in accordance with NUREG-0619 Table 2 between 1983 and 1993 on Unit 2, and between 1986 and 1996 on Unit 1. By letter dated March 8, 1995 the NRC approved a Quad Cities request to perform automated ultrasonic inspections in lieu of dye penetrant inspections. The nozzles were inspected using the automated ultrasonic technique starting in 1995 on Unit 2 and 1998 on Unit 1. The BWR Owners Group also submitted proposed alternate inspection requirements to NUREG-0619 to the NRC in General Electric Report GE-NE-523-A71-0594 dated October 30, 1995. The NRC issued a Safety Evaluation (TAC M94090), dated June 5, 1998 for GE-NE-523-A71-0594 that accepted the proposed alternate inspections. The NRC issued a Final Safety Evaluation (TAC MA6787) dated March 10, 2000 for GE-NE-523-A71-0594, Revision 1, dated August 1999. The SER permits ultrasonic inspections meeting ASME Section XI Appendix VIII as an alternate to the liquid penetrant inspections originally stipulated in Table 2 of NUREG-0619. The automated ultrasonic inspection technique used at Quad Cities since 1995 meets ASME Section XI Appendix VIII as executed by the Performance Demonstration Initiative. GE-NE-523-A71-0594, Revision 1 was reissued as GE-NE-523-A71-0594-A, Revision 1, dated May 2000 to denote the approved status of the alternate inspection requirements. Quad Cities will continue to utilize the inspection methods and inspection frequencies stipulated in GE-NE-523-A71-0594-A, Revision 1, May 2000.

The resultant bracket loads are given to the vessel manufacturer so the vessel brackets can be sized to meet the Section III criteria.

Deficiencies with the QA program of the subcontractor who built these feedwater spargers were identified by the NRC. Following a detailed review of the documentation for the material for these spargers, GE and CECo found no deficiencies in the spargers for Quad Cities.

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 a pair of supply lines routed in the reactor vessel to accommodate differential movement between the shroud and the vessel. The supply line pairs terminate at a common vessel nozzle. Each half has distribution nozzles pointed radially inward and downward at a slight angle to achieve specified distribution pattern.

To prevent possible gas entrapment in the nonflowing portion of the stainless steel core spray piping and nozzles, an opening sized to provide a flow in the line of approximately five gal/min into the reactor vessel with normal operating pressure differential between the shroud head and the region outside the core spray line has been provided.

The standby liquid control system injects solution through 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 do not have 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. [3.9.57]

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 twelve tubes are for the source range monitor (SRM) detectors and eight for the intermediate range monitor (IRM) detectors.

The other 41 are for the local power range monitor incore (LPRM) detector strings, and a smaller guide tube for the traveling incore probe (TIP). 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: [3.9.58]

A. Provide an adequate distribution of coolant flow within the reactor, and

B. 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 is designed 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 shroud and jet pumps form an inner vessel which is sufficiently leak tight, despite thermal expansion allowance, to permit reflooding the core as described in Section 6.3.3.1.1 following a design basis LOCA.

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, the ASME Code Section III is used as a guide to determine limiting stress intensities and cyclic loading for the core internal structure. [3.9.59]

The loading conditions which occur during excursions or LOCAs 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 LOCA. 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 LOCA. The reactor core structural components are designed so that deformations produced by accident loadings will not prevent insertion of control rods.

3.9.5.3 Design Bases

This section presents the details of key evaluations performed for the reactor vessel internals, excluding fuel and control rod assemblies.

3.9.5.3.1 Pressure Loadings

Three sets of differential pressure loading have been considered in the design of the reactor pressure vessel internals. The pressure distribution estimated to occur at steady state design power level results in the differential pressures summarized in Table 3.9-19. In addition, Table 3.9-19 provides differential pressures resulting from rapid depressurization of the reactor as a result of a steam line rupture and a recirculation line rupture. [3.9.60] The Table 3.9-19 differential pressure (deltaP) values remain applicable for ATRIUM 10XM fuel.[24]

With the first application of the Extended Operating Domain and Equipment Out-of-Service for Quad Cities, the mechanical evaluation of the reactor internals and fuel assembly analysis was revised[23] as outlined in the GE proprietary document. It assumed Increased Core Flow to 108% of rated as described in Section 4.4.3.1. It also included the combined and alternate Final Feedwater Temperature Reduction (FFWTR) option. The methods and assumptions along with the results of that analysis[23] were outlined in the GE proprietary document. [3.9.61]

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[12]. 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[13, 14].

As part of the Extended (Licensed) Power Uprate (LPU @ 2957 MWt), structural integrity assessment of the key reactor internal components was performed (Reference 28). The reactor internal pressure differences and the acoustic and flow induced loads as result of the postulated Recirculation line break (LOCA) (including GE14 fuel) were used as input to the LPU evaluation.

3.9.5.3.1.1 Thermal-Hydraulic Model

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. [3.9-62]

The recirculation line break is assumed to be a guillotine line severance at the pressure vessel outlet. 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 2511 MWt with 98 x 10^6 lb/h 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 five 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 3.9-11. 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.

Table 3.9-20 delineates the pressure force acting on major components.

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.

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 (Licensed) Power Uprate (LPU @ 2957 MWt), structural integrity assessment of the key reactor internal components was performed (Reference 28). The thermal hydraulic analysis data, reactor internal pressure differences, and the acoustic and flow induced loads due to the postulated Recirculation line break (LOCA) were used as input to the LPU evaluation.

3.9.5.3.1.2 Recirculation Line Rupture

The recirculation line rupture (double-ended) causes high flow rates 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. [3.9.63]

Although the flow rate from the downcomer is high, the pressure change rate in the mixing plenum is only about 20 psi/s assuming no admission 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 current 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 recirculation line.

Results of the recirculation line break are shown in Table 3.9-19 and compared to component capabilities. The guide tubes and core plate are related to the scram capability.

The calculated maximum pressure differential across the core for the recirculation line break does not increase above that at rated conditions, well below the 45 psi pressure differential required for fuel bundle lifting. The calculated maximum pressure differential across the fuel channel would be bounded by the initial value for the recirculation line break because of the rapid core inlet flow decreases to about 40% 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 flow rate 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 millisecond 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. [3.9.64]

Pipe rupture studies such as those performed at GE and Battelle Memorial Institute investigated fracture mechanics and provided 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 flaw 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 flaw that 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 previously, 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.

With the first application of the Extended Operating Domain and Equipment Out-of-Service for Quad Cities, an analysis of the reactor internals loads and fuel assembly analysis was revised[23] as outlined in the GE proprietary document. It specifically analyzed the postulated sudden break in the recirculation line. It assumed Increased Core Flow to 108% of rated as described in Section 4.4.3.1. The methods and assumptions along with the results of that analysis[23] were outlined in the GE proprietary document. [3.9-65]

A detailed thermal-hydraulic analysis of a double-ended guillotine break of the reactor recirculation suction line was performed in 1994 to obtain a more accurate definition of the asymmetric lateral blowdown loads that are applied to the core shroud[13]. The TRACG computer code was used to calculate a detailed pressure distribution in the downcommer 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. [3.9-66]

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 feedwater 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 non-symmetrical 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 laterial load. The resulting forces and core shroud moments from this analysis are summarized in Table 3.9-21 and Figures 3.9-19 and 3.9-20. 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. The result of the original design basis analysis is provided in Figure 3.9-12. This figure is a graphical representation of the applied core shroud lateral force as a function of time (milliseconds) after the line break occurs. A reassessment of the acoustic lateral load was performed as part of the core shroud evaluations and repairs. Based on the available industry information[14] and 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[17].

A finite element analysis of the core shroud was prepared to verify that the stresses are within the ASME Section III limits. This analysis and corresponding stress evaluation included all of the applied loading cases including recirculation line breaks and seismic.

The results of the TRACG recirculation line break analysis were independently verified by the BWRVIP using the COMPACT 3D computational fluid dynamics computer program. See References 14, 17, and Calculation 9389-64-DQ, Section 8 for further details of this analysis.

As part of the Extended (Licensed) Power Uprate (LPU @ 2957 MWt), structural integrity assessment of the key reactor internal components was performed (Reference 28). The thermal hydraulic analysis data, reactor internal pressure differences, and the acoustic and flow induced loads due to the postulated Recirculation line break (LOCA) (including GE14 fuel) were used as input to the LPU evaluation.

3.9.5.3.1.3 Steam Line Rupture

Following the instantaneous steam line (double-ended) rupture, critical flow is established in each broken line. Since the break is postulated to be upstream of the steam flow restrictor, the break area is the sum of one open steam line area plus one steam flow restrictor area at the other end of break. As shown in Figure 3.9-13, this break causes the system to depressurize at the rate of about 75 psi/s during initial steam blowdown. [3.9-67]

The design-break is assumed to have a constant break area of 2.4 ft^2. 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 cannot occur because the decompression rate is limited by the saturated upper core regions.

Internal forces following the instantaneous steam line break are shown on Figure 3.9-14. 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 two seconds are the result of saturating the previously subcooled lower plenum inventory. The high exit mass flow rate is associated with this. 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 ten 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 so small, due to the compressible effects of steam and the large expansion as the wave enters the pressure vessel, that it 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 as shown in Figure 3.9-15. This is because 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. [3.9-68]

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 (Figure 3.9-16). 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 line system pumps remain at full speed through the transient. Since they help to sustain interregion pressure differentials this is a conservative 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. With the first application of the Extended Operating Domain and Equipment Out-of-Service for Quad Cities, an analysis of the reactor internals loads and fuel assembly analysis was revised[23] as outlined in the GE proprietary document. It was analyzed at accident conditions which bound the postulated steam line rupture. It assumed Increased Core Flow to 108% of rated as described in Section 4.4.3.1. The methods and assumptions along with the results of that analysis[23] were outlined in the GE proprietary document. [3.9-69]

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[12]. Since this analysis was performed utilizing "best estimate" techniques it is not a design basis main steam line break analysis. The differential pressure calculated in this analysis are applicable only for use in structural flaw assessments and safety consequences evaluations. The results of the TRACG main steam line break analysis were independently verified by the BWRPIV using the RETRAN-02 computer program. [3.9-70]

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 3.9-14 it is only slightly over rated pressure differential. 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 maximum pressure differential tending to bulge the channel outward was calculated to be approximately 16 psi. Test data from a similar type fuel channel indicated that the deflections followed the elastic equation at room temperatures for stresses greater than twice 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. The GE8x8NB-3 interactive channel and the ATRIUM 10XM Advanced channel will perform in a manner similar to the standard channel in terms of stress loading and dimensional clearance.[22][24]

Even if it was 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. 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 one second after the accident. 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.

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 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. These dimensions and the guide tube design were derived to provide flexural stability during normal operation and collapse resistance during blowdown accident conditions. The differential pressures which would result from these conditions are tabulated in Table 3.9-19. Based upon a yield stress of 17,300 psi, the minimum collapse pressure is 54 psi of pressure differential across the guide tube.

When the OBE earthquake design reactions of 0.4g horizontal and 0.08g vertical are combined with the pressure differential of 32 psi across the lower shroud, the maximum resulting general primary membrane stress in the shroud support legs is 15,200 psi. The ASME Code allowable stress is 23,300 psi for this category of loading. It should be noted that the horizontal acceleration was derived from the RPV seismic analysis provided in Reference 31. The 0.4g value corresponds to the location on the RPV at which the internals are attached.

As part of the Extended (Licensed) Power Uprate (LPU @ 2957 MWt), structural integrity assessment of the key reactor internal components was performed (Reference 28). The reactor internal pressure differences and the acoustic and flow induced loads as result of the postulated Recirculation line break (LOCA) (including GE14 fuel) were used as input to the LPU evaluation.

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 low pressure coolant injection (LPCI) thermal shock transient. Three specific locations are summarized here and shown on Figure 3.9-17. [3.9-71]

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%.

The strain, while higher than the 5.0% strain permitted in the ASME Code for 10 cycles of loading, corresponds to about 6 allowable cycles of an extended type ASME Code curve which would apply to fewer loading cycles than 10. Figure 3.9-18 illustrates both the ASME Code 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 Code curve represents a similar criteria to that used in the ASME Code, 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 one allowable cycle of loading. It is emphasized that even a 10% strain range 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:

A.	Amplitude of alternating stress	180,000 psi
B.	Allowable ASME III cycles	220
C.	Maximum strain range	1.34%

At the inside of the shroud the total integrated neutron flux at end of life is 2.7 x 10^{20} nvt at 1 MeV. The maximum thermal shock stress in this region is 155,700 psi or 0.57% strain range. The shroud structural material is 304 stainless steel which does not suffer from irradiation embrittlement. It does experience hardening and an apparent loss in uniform elongation but not a loss in reduction of area. 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 x 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 x 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 x 10^{21} nvt (>1 MeV). The value of percent reduction in area for type 304 stainless steel is a minimum of approximately 38%[7] for a temperature of 550°F and neutron flux of 1 x 10^{21} nvt (>1 MeV) and a reduction in area of 52.5% [8] for a temperature of 750°F and neutron flux of 6.9 x 10^{21} nvt (>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 DBA. [3.9-72]

As part of the Extended (Licensed) Power Uprate (LPU @ 2957 MWt), structural integrity assessment of the key reactor internal components was performed (Reference 28). The reactor internal pressure differences and the acoustic and flow induced loads as result of the postulated Recirculation line break (LOCA) (including GE14 fuel) were used as input to the LPU evaluation.

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: [3.9-73]

1. Recirculation inlet nozzle

2. Mid-core inside of vessel

3. Control rod drive penetration

The results of the recirculation nozzle are as follows:

		Sleeve	Nozzle
A.	Amplitude of alternating stress	595,000 psi	215,000 PSI
B.	Allowable ASME III cycles	12	130
C.	Maximum strain range	4.5%	1.6%

The results at mid-core inside of vessel are 67,500 psi peak stress. More than 1000 such cycles would be imposed under the ASME III fatigue criteria. The total maximum vessel irradiation (1 MeV) at this point has been found to be 2.4 x 10^{17} nvt which is below the threshold level of any Nil Ductility Temperature shift for the vessel material. Therefore, irradiation effects can be ignored at all locations on the vessel. Irradiation effects are discussed in more detail in Section 5.3.2.

The results on the control rod drive penetration are:

A.	Amplitude of alternating stress	560,000 psi
B.	Allowable ASME III cycles	14
C.	Maximum strain range	3.7%

3.9.5.3.4 Seismic Loading

The most important single piece of equipment is, of course, the reactor vessel and its internals. A dynamic analysis was performed which was very similar to the analysis described in Section 3.7.2.1.6 for the reactor vessel. The analysis discussed in Section 3.7.2.1.6 was utilized for building and support design, whereas the analysis described in this section was used in evaluating the vessel internals. The difference between the two analyses is that, in the latter analysis, the vessel internals were modeled in much greater detail, considering specific internals in the mass model calculations. [3.9-74]

A core shroud repair designed to structurally replace the core shroud's horizontal welds H1 through H7 and provide vertical clamping forces on the shroud was installed on Unit 2 during the Q2R13 and Unit 1 during Q1R14 refuel 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 was generated with the core shroud repair installed. [3.9-75]

The internals model included specific masses for the shroud, CRD housings, top and bottom core plates, fuel, guide tubes, separator, dryer, and vessel heads in addition to the pedestal, support skirt, and reactor-turbine building. All relevant modes of vibration of the coupled system were considered. The damping factors utilized are defined in Table 3.7-1. As in the other dynamic analyses, the results of shears, moments, displacements and accelerations were obtained and used to evaluate the integrity of the various components. The DBE loading was also analyzed to assure compliance with the seismic criteria. For Extended (Licensed) Power Uprate (LPU), the effect of GE14 Fuel assembly properties on the seismic loads were assessed and found to result in no significant change in loads, (Reference 28).

The effects of Westinghouse SVEA-96 Optima2 fuel on the combined seismic/LOCA response of the reactor vessel internals and core were evaluated in a dynamic analysis of the fuel [30]. That analysis showed that introduction of Optima2 fuel results in no significant change in loads on reactor internals components and that all design criteria are met for the response of the Optima2 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[30]. The effects of AREVA (now Framatome) ATRIUM 10XM fuel on the seismic response of the reactor pressure vessel internals was evaluated by GEH[32]. It was demonstrated that for ATRIUM 10XM fuel, there would be an insignificant effect on the loadings. In addition, an AREVA evaluation demonstrates that all design criteria are met for the response of the ATRIUM 10XM fuel assemblies to the combined seismic and LOCA loading. The methods, computer programs, calculation and result used in this analysis are documented in an AREVA report[24].

3.9.6 Inservice Testing of Pumps and Valves

Presently, Inservice Testing (IST) of pumps and valves is governed by the fifth 10-Year Interval IST Program which will remain in effect through February 17, 2023. The IST program was developed in response to the requirements of 10 CFR 50.55a. [3.9-76]

In accordance with 10 CFR 50.55a, IST programs are updated at 10-year intervals to comply with the requirements of the edition 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 twelve (12) months prior to the start of the 10-year testing interval. The current IST program is based upon the requirements of the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) 2004 Edition through 2006 Addenda, consistent with the requirements of the applicable revision of 10 CFR 50.55a.

The 10 CFR 50.55a regulation recognizes that the requirements of later editions and addenda of the ASME Code may not be practical to implement due to design, geometry, and materials of construction of components and systems. Therefore, the regulation permits exceptions to impractical examination or testing requirements by the granting of specific relief requests. The fifth 10-Year Interval IST Program contains all the approved relief requests.

The construction permits for Quad Cities Units 1 and 2 were issued on February 15, 1967. 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 IST program contains essentially no ASME Code Class 1, 2, or 3 designed systems. The system classifications used as a basis for the Inservice Inspection (ISI) and IST programs 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 ISI/IST requirements. Components within the reactor coolant pressure boundary (RCPB), as defined in 10 CFR 50.2, are designated as ISI Class 1 as determined by 10 CFR 50.55a, with the exception allowed by 10 CFR 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, ISI requirements of Section XI of the ASME Code and IST requirements of the ASME OM Code are then assigned to these components, within the constraints of existing plant design. [3.9-77]

The extent of the Class 1, 2, 3, and MC designations for systems or portions of systems subject to the ISI/IST requirements are identified on the Quad Cities Piping and Instrumentation Diagrams (P&ID) and IWE (MC) program drawings. In accordance with Regulatory Guide 1.26, the ISI/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 core pressure boundary is addressed in Section 5.2.4. Inservice inspection for Class 2, 3, and MC 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 IST program for ISI Class 1, 2 and 3 pumps meet the requirements for ASME OM Code Subsection ISTB. Where these requirements were determined to be impractical, specific requests for relief have been approved by the NRC[9]. [3.9-78]

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 non-safety related pumps have been included in the IST program scope at the request of the NRC. Any pumps excluded from the scope of the IST program are based on the guidance provided by the Code. [3.9-79]

3.9.6.2 Inservice Testing of Valves

The IST program for ISI Class 1, 2 and 3 valves meet the requirements of ASME OM Code Subsection ISTC. Where these requirements were determined to be impractical, specific requests for relief have been approved by the NRC[9]. [3.9-80]

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 non-safety related valves have been included in the IST program scope at the request of the NRC. Any valves excluded from the scope of the IST program are based on the guidance provided by the Code. [3.9-81]

3.9.7 References

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9. Letter from J. S. Wiebe (NRC) to M. J. Pacilio (EGC), February 14, 2013, “Quad Cities Nuclear Power Station, Units 1 and 2 – Safety Evaluation in Support of Request for Relief associated with the Fifth 10-Year Interval Inservice Testing Program.”

10. GE Drawing - Vessel Loading, 886D485 P1, Sheet 7.

11. GENE-525-A100-0995, DRF 137-0010-8, "Analyses of the Dresden and Quad Cities Shroud Repair Hardware Seismic Design with Improved Tie Rod and Shroud Weld Crack Equivalent Rotational Stiffness," Appendix B.

12. GENE-523-A163-1194, "Quad Cities and Dresden Main Steam Line Break Analysis with TRACG Model," November 9, 1994.

13. GENE-L12-00819-05, "Core Shroud Blowdown Load Calculation During Recirculation Suction Line Break by TRACG Analysis for Dresden Nuclear Power Station Units 2 and 3, and Quad Cities Unit 1 and 2."

14. BWR-VIP Report No. SL-4942, "BWR Core Shroud Evaluation Load Definition Guideline."

15. NRC Letter to D. L. Farrar, Core Shroud Repair Safety Evaluation (TAC Nos. M91301 and M91302), June 8, 1995.

16. NRC Letter to D. L. Farrar, Core Shroud Repair Supplemental Safety Evaluation (TAC Nos. M91301 and M91302), September 11, 1995.

17. SL-4971, Rev. 1, "Final Evaluation of the Core Shroud Flaws at the H5 Horizontal Weld for Dresden Unit 3," Chapter 3.0.

18. GENE-771-68-1094, Rev. 4, "Shroud and Shroud Repair Hardware Stress Analysis - Shroud Repair for H1 through H7 Welds for QC Units 1 and 2."

19. GENE-771-68-1094 Supplemental A to Rev. 4, Shroud Mechanical Repair Program Quad Cities, Supplement A to Shroud and Shroud Repair Hardware Stress Analysis, April 1995. (Proprietary Information).

20. GENE-771-69-1094, Rev. 1, "Backup Calculations for Shroud Repair Shroud Stress Report for QC Units 1 and 2," (Proprietary Information).

21. GENE-771-70-1094, Rev. 3, "Backup Calculations of Shroud and Shroud Repair Hardware Stress Analysis for QC," (Proprietary Information).

22. "General Electric Standard Application for Reactor Fuel" (GESTARII) Amendment 21, General Electric Company, February 1991, NEDE-24011-P-A-10 and U.S. Supplement, March 1991, NEDE-24011-P-A-10-0.

23. "Extended Operating Domain and Equipment Out-of-Service for Quad Cities Nuclear Power Station Units 1 and 2," General Electric Document NEDC-31449, dated July 1987. Subsequent Revision 1 dated April 1992.

24. ANP-3305P, Revision 3, "Mechanical Design Report for Quad Cities and Dresden ATRIUM 10XM Fuel Assemblies," AREVA Inc., August 2016.

25. "Response to ComEd Questions on the Dresden and Quad Cities' Seismic/LOCA Mechanical Evaluation," letter to R. J. Chin (ComEd-NFS) from J. H. Riddle (SPC), number JHR:97:028, dated January 27, 1997, with attachment.

26. "Core Shroud Modification Project Earthquake Analysis - Fuel," NDIT No. QC-0215-01, dated January 26, 1996.

27. Deleted.

28. NEDO-32961, "Safety Analysis Report for Quad Cities 1 & 2 Extended Power Uprate." March 2001.

29. NEDC-33187P, "Safety Evaluation in Support of the New Steam Dryer for Quad Cities Units 1 & 2," May 2005.

30. Westinghouse Report NF-BEX-06-24, Rev 3, "SVEA-96 Optima2 Reactor Seismic Loads – Dresden and Quad Cities," February 21, 2006 (included in OPTIMA2-TR028QC-SEISMIC, "Optima2 Reactor Seismic Loads - Quad Cities," February 2006).

31. Quad Cities Reactor Pressure Vessel Design Report.

32. 002N8478, Rev 0, Quad Cities Mixed Core Reactor Vessel Member Loads, September 2015.

33. EC 626581, Rev. 0, “ASME Code Reconciliation for Use of Higher Allowable Stresses.”

Table 3.9-1

SUMMARY OF DESIGN BASIS AND PREDICTED THERMAL CYCLES
FOR THE REACTOR PRESSURE VESSEL

Cycle Description	Units 1 and 2 Original Design Basis Allowable (Note 1)	Unit 1 Cycle Prediction Year 40 (Note 2)	Unit 2 Cycle Prediction Year 40 (Note 2)	Unit 1 and 2 Revised Design Basis Allowable (Note 3)
Plant cooldown	119	286	274	286
Plant heatup	120	292	275	298
Safety relief valve blowdown	1	6	1	12
Reduction of power for plant shutdown	119	50	44	119
Turbine roll with feedwater injection	120	50	44	120
Head spray injection	119	3	5	119
Loss of feedwater heaters — full	80	114	77	114
Loss of feedwater heaters —partial	80	6	42	80
Loss of feedwater flow	80	15	42	80
Scram	200	294	275	294
Batch feedwater addition during hot standby or plant cooldown	595	0	0	202

Notes:

1. Original cycles formed the original basis for Quad Cities design. These were the originally analyzed values.

2. Predicted cycles for each unit are based upon extrapolating actual counted cycles through March 1988 over the full 40 year plant life and are thus predictions of the actual cycles that each unit will experience.

3. Revised cycles provide new basis for Quad Cities design allowables and envelop the predicted cycles for both units.

Table 3.9-1A

SUMMARY OF DESIGN BASIS PREDICTED STRESS CYCLES
FOR THE RPV STUDS

Stress Cycle	Max. No. of Cycles
Bolt/Unbolt	46
Startup/Shutdown	230
SCRAM	215
SRV Blowdown	3
Hydrotest	48

Notes:

1. Revised cycles for the RPV studs are based on the results of GENE Analysis GENE-0000-0023-2510-LTR1. The actual number of cycles is tracked by the Quad Cities Engineering Department to ensure that the maximum number of cycles assumed in the analysis are not exceeded.

Table 3.9-2

REACTOR PRESSURE VESSEL
COMPONENT FATIGUE USAGES BASED ON REVISED THERMAL CYCLES

Component	Old Usage (A)	Delta Usage For 286 Cycles of Startup/Shutdown (B)	Delta Usage For 294 Cycles of Scram (C)	Delta Usage for 114 Cycles of Loss of FW Heater (D)	Delta Usage for 12 Cycles of SRV Blowdown (E)	Total New Usage (A+B+C+D+E)
Recirculation Outlet						
Safe end (SS)[Note 1]	0.030	0.014	0.019	0.000	0.039	0.102
Nozzle (LAS)[Note 1]	0.110	0.053	0.068	0.000	0.039	0.270
Recirculation Inlet						
Safe end (SS)	0.000	0.000	0.000	0.000	0.020	0.020
Thermal sleeve (SS)	0.220	-0.089[Note 2]	0.150	0.000	0.020	0.301
Nozzle (LAS)	0.002	0.003	0.005	0.000	0.020	0.029
Feedwater						
Safe end (CS)[Note 1]	0.480	0.011	0.044	0.000	0.003	0.538
Nozzle (LAS)	0.382	0.008	0.023	0.001	0.003	0.416
Core Spray						
Safe end (SS)	0.004	0.003	0.006	0.000	0.039	0.051
Nozzle (LAS)	0.013	0.008	0.019	0.000	0.039	0.079

Table 3.9-2 (continued)

REACTOR PRESSURE VESSEL
COMPONENT FATIGUE USAGES BASED ON REVISED THERMAL CYCLES

Component	Old Usage (A)	Delta Usage For 286 Cycles of Startup/Shutdown (B)	Delta Usage For 294 Cycles of Scram (C)	Delta Usage for 114 Cycles of Loss of FW Heater (D)	Delta Usage for 12 Cycles of SRV Blowdown (E)	Total New Usage (A+B+C+D+E)
CRD Hydraulic Return						
Safe end (SS)	0.007	0.003	0.002	0.000	0.039	0.052
Nozzle (LAS)	0.090	0.040	0.030	0.000	0.039	0.199
CRD Penetration						
(SS)	0.001	0.000	0.000	0.000	0.039	0.040
(INCONEL)	0.000	0.000	0.000	0.000	0.039	0.039
(LAS)	0.000	0.000	0.000	0.000	0.039	0.039
2-inch Instrument Nozzle (LAS)[Note 4]	0.063	0.028	0.063	0.000	0.039	0.192
Support Skirt (LAS)	0.935[Note 3]	0.026	0.001	0.000	0.039	1.001
Refuel. Cont. Skirt (LAS)	0.020	0.028	0.062	0.000	0.039	0.149
Shroud Support (LAS)	0.340	0.226	0.002	0.000	0.013	0.580

Table 3.9-2 (continued)

REACTOR PRESSURE VESSEL
COMPONENT FATIGUE USAGES BASED ON REVISED THERMAL CYCLES

Component	Old Usage (A)	Delta Usage For 286 Cycles of Startup/Shutdown (B)	Delta Usage For 294 Cycles of Scram (C)	Delta Usage for 114 Cycles of Loss of FW Heater (D)	Delta Usage for 12 Cycles of SRV Blowdown (E)	Total New Usage (A+B+C+D+E)
Closure Flange Region Flange (LAS)	0.006	0.008	0.019	0.000	0.039	0.072
Studs (LAS)	0.850	0.692	0.173	0.000	0.050	1.765
Vessel Shells (LAS)	0.002	0.003	0.006	0.000	0.039	0.050

Notes:
1. SS = Stainless Steel; LAS = Low Alloy Steel; CS = Carbon Steel
2. Negative delta usage includes removal of conservatism in original analysis
3. Based on revised analysis, which included all revised cycles except SRVB
4. Total Fatigue Usage for Unit 2 Nozzle N-11B is bounded by value listed

Table 3.9-3

DESIGN BASIS STRESS SUMMARY
FOR REACTOR VESSEL RING GIRDER

Item	Allowable
Bracket	
— bending	22 ksi
— shear	14 ksi
Rod — tension	136 ksi
Springs	100% of spring capacity

Table 3.9-4

DESIGN BASIS STRESS SUMMARY
AT BASE OF CONCRETE REACTOR PEDESTAL

Loading Combination	Allowable Stresses (psi)		
	f_s	f_c	v
D + H + E	2400 0	1690	183
D + H + R + E	5400 0	2700	280

Nomenclature:
D = dead load
H = postulated thermal load
R = jet load
E = operating basis earthquake
f_s = reinforcing tensile stress
f_c = concrete compressive stress
v = tangential shear stress

Table 3.9-5

DESIGN BASIS STRESS SUMMARY
FOR IN REACTOR RING GIRDER ANCHOR BOLTS

Loading Combination	Allowable Stresses (psi) f_s	f_v
D + H + E	14000	10000
D + H + R + E	21000	15000

Nomenclature:
D = dead load
H = postulated thermal load
R = jet load
E = operating basis earthquake
f_s = bolt tensile stress
f_v = bolt shear stress

Table 3.9-6

ALLOWABLE STRESSES IN REACTOR VESSEL SUPPORT ELEMENTS
DUE TO REQUESTED LOAD COMBINATION[Note 1]
D + H + R + E'

Vessel Support Element	Stress Component	Allowable Stress (psi)
Ring girder anchor bolts	Tensile stress	23400
	Shear stress	16700
Concrete pedestal	Reinforcing tensile stress	60000
	Concrete compressive stress	3180
	Tangential shear stress	330
Ring girder high strength bolts	Tensile stress	115000
	Shear stress	66000

Nomenclature:
D = dead load
H = postulated thermal load
R = jet load
E' = design basis earthquake

Notes:
1. Results for this load combination were requested in Question 12.5 of FSAR Amendment 16.

Table 3.9-7

UNIT 1
TABULATION OF MAXIMUM STRESSES FOR ORIGINAL CLASS I PIPING

Item	System	S&L Drawing	Point Number	Material	O.D. (In.)	Thickness (In.)	Intensification Factor	Component	Weight Stress (psi)[Note 1]	Pressure Stress (psi)	OBE Total Seismic Stress (psi)[Note 1]	OBE Combined Stress Sigma (psi)	OBE Allowable Stress (psi)	DBE Allowable Stress (psi)	DBE Combined Stress Sigma (psi)	DBE Yield Stress (psi)
1	Core spray pump section 1A and 1B	M-465	350 TGNTR	ASTM A106 GR. B	4.500	.237	1.92	.250" Reinf. Branch Conn.[Note 2]	2,535	180	6,559	17,640	18,000	13,118	30,234	31,900
2	Core spray pump discharge 1A	M-466	470 TGNTBP	ASTM A106 GR. B	12.750	.375	.297	.375" Reinf. Branch Conn.[Note 2]	1,143	3,025	3,050	15,478	18,000	7,100	24,537	31,900
3	Core spray pump discharge 1B	M-466	245 TGNTR	ASTM A106 GR. B	8.625	.322	1.90	.500" Reinf. Branch Conn.[Note 2]	941	2,323	4,947	13,529	18,000	9,894	22,928	31,900
4	Core spray from drywell to RX-16A	M-467	45 BEND	A312 or A376 GR. TP 304	10.750	.593	1.00	Elbow	209	4,803	862	5,854	17,460	1,724	6,716	18,200
5	Core spray from drywell to RX-16B	M-467	145 BEND	A312 or A376 GR. TP 304	10.750	.593	1.0	Elbow	525	4,803	2,489	7,797	17,460	4,978	10,286	18,200
6	RHRS pump suction 1A, 1B, 1C, and 1D	M-468	95 TGNTBP	ASTM A106 GR. B	20.000	.375	7.93	Branch Conn.	396	2,077	1,087	13,837	18,000	2,174	22,457	31,900
7	RHRS pump discharge 1A and 1B	M-469	530 TGNTBP	ASTM A106 GR. B	12.750	.375	1.84	.375" Reinf. Branch Conn.[Note 2]	670	1,627	7,703	17,035	18,000	15,406	31,208	31,900
8	RHRS pump discharge 1C and 1D	M-470	345 TGNTR	ASTM A106 GR. B	18.000	.438	2.71	.625" Reinf. Branch Conn.[Note 2]	1,027	2,001	3,789	15,052	18,000	7,578	23,320	31,900

Table 3.9-7 (Continued)

UNIT 1
TABULATION OF MAXIMUM STRESSES FOR ORIGINAL CLASS I PIPING

Item	System	S&L Drawing	Point Number	Material	O.D. (In.)	Thickness (In.)	Intensification Factor	Component	Weight Stress (psi)[Note 1]	Pressure Stress (psi)	OBE Total Seismic Stress (psi)[Note 1]	OBE Combined Stress Sigma (psi)	OBE Allowable Stress (psi)	DBE Allowable Stress (psi)	DBE Combined Stress Sigma (psi)	DBE Yield Stress (psi)
9	RHRS service water pump suction 1-1001A	M-471	200 TNGTBP	ASTM A106 GR. B	24.000	.375	8.98	Branch Conn.	581	534	1,032	15,018	18,000	2,064	24,285	31,900
10	RHRS service water pump suction 1-1002A	M-471	445 TNGTBP	ASTM A106 GR. B	24.000	.375	8.98	Branch Conn.	572	534	1,269	17,066	18,000	2,538	28,461	31,900
11	RHRS service water pump suction 1-1003A	M-471	45 BEND	ASTM A106 GR. B	12.750	.375	1.00	Elbow	29	1,729	3,497	5,254	18,000	6,992	8,750	31,900
12	RHRS service water pump disch. to htr. 1-1005A	M-472	255 TNGTBP	ASTM A106 GR. B	16.000	.375	2.25	w/Fillet Weld Branch Conn.[Note 2]	387	3,224	1,243	6,892	18,000	2,486	9,689	31,900
13	RHRS service water pump disch. to Htr. 1-1005B	M-472	75 TNGTBP	ASTM A106 GR. B	16.000	.375	2.25	w/Fillet Weld Branch Conn.[Note 2]	684	3,224	2,094	9,474	18,000	4,188	14,186	31,900
14	RHRS service water pump disch. to Htr. 1A	M-472	560 Bend	ASTM A106 GR. B	16.000	.375	1.00	Elbow	1,401	3,224	8,852	13,479	18,000	17,704	22,331	31,900
15	RHRS service water pump disch. to Htr. 1B	M-472	720 Bend	ASTM A106 GR. B	16.000	.375	1.00	Elbow	207	3,224	14,309	17,740	18,000	28,618	32,049	31,900
16	RHRS htr. exch. 1A & 1b service water return	M-473	68 TGNTR	ASTM A106 GR. B	12.750	.375	2.17	Branch Conn.	2,712	2,521	3,562	16,135	18,000	7,124	23,865	31,900
17	RHRS htr. exch. 1A & 1b service water return	M-473	405 TGNTBP	ASTM A134 A283 GRB	30.000	.375	3.80	.750" Reinf. Branch Conn.[Note 2]	578	6,257	1,378	11,215	11,760	2,756	16,173	31,900
18	Reactor building cooling water	M-474	5 Anch.	ASTM A106 GR. B	8.625	.322	1.00	Anchor	913	506	737	2,156	18,000	1,474	2,893	31,900

Table 3.9-7 (Continued)

UNIT 1
TABULATION OF MAXIMUM STRESSES FOR ORIGINAL CLASS I PIPING

Item	System	S&L Drawing	Point Number	Material	O.D. (In.)	Thickness (In.)	Intensification Factor	Component	Weight Stress (psi)Note 1	Pressure Stress (psi)	OBE Total Seismic Stress (psi)Note 1	OBE Combined Stress Sigma (psi)	OBE Allowable Stress (psi)	DBE Allowable Stress (psi)	DBE Combined Stress Sigma (psi)	DBE Yield Stress (psi)
19	Reactor building cooling water	M-474	85 Bend	ASTM A106 GR. B	8.625	.322	1.00	Elbow	1,318	506	575	2,399	18,000	1,150	2,974	31,900
20	HPCI pump suction	M-475	70 TGNTBP	ASTM A106 GR. B	16.000	.375	4.22	.250" Reinf. Branch Conn.Note 2	386	644	3,222	15,870	18,000	6,444	29,467	31,000
21	Reactor feed (outside) and HPCI discharge	M-480	500 TGNTBP	ASTM A106 GR. B	14.000	1.094	1.51	1.00" Reinf. Branch Conn.Note 2	220	3,149	8,219	15,892	18,000	16,438	28,303	30,000
22	HPCI turbine steam supply	M-476	285 TNGTBP	ASTM A106 GR. B	10.750	.594	3.76	Branch Conn.	122	4,357	2,707	14,994	18,000	5,414	25,172	25,900
23	HPCI turbine steam to penetration	M-476	390 TGTBP	ASTM A106 GR. B	20.000	1.031	2.35	1.00" Reinf. Branch Conn.Note 2	491	4,730	3,613	14,374	18,000	7,226	22,865	25,900
24	HPCI turbine steam exhaust	M-476	130 TGNTBP	ASTM A106 GR. B	20.000	.375	4.04	.375" Reinf. Branch Conn.Note 2	560	1,007	3,485	17,348	18,000	6,970	29,128	31,000
25	Main steam (outside)	M-477	105 Anch.	ASTM A106 GR. B	20.000	1.031	1.00	Anchor	828	4,113	6,527	11,468	18,000	13,054	17,995	25,900
26	Pressure suppression sheet 1	M-478	165 TNGTBP	ASTM A106 GR. B	6.625	.280	1.57	.500" Reinf. Branch Conn.Note 2	2,355	414	4,162	10,645	18,000	8,324	17,180	31,900
27	Pressure suppression sheet 2	M-479	100 TGNTR	ASTM A106 GR. B	20.000	.375	4.04	.375" Reinf. Branch Conn.Note 2	1,375	969	2,849	17,872	18,000	5,608	29,382	30,000
28	Standby gas treatment	M-483 M-484	560 Bend	ASTM A211 A245 GR. A	24.000	.140	1.00	Elbow	794	715	11,294			22,598	24,097	31,900

Table 3.9-7 (Continued)

UNIT 1
TABULATION OF MAXIMUM STRESSES FOR ORIGINAL CLASS I PIPING

Item	System	S&L Drawing	Point Number	Material	O.D. (In.)	Thickness (In.)	Intensification Factor	Component	Weight Stress (psi)Note 1	Pressure Stress (psi)	OBE Total Seismic Stress (psi)Note 1	OBE Combined Stress Sigma (psi)	OBE Allowable Stress (psi)	DBE Allowable Stress (psi)	DBE Combined Stress Sigma (psi)	DBE Yield Stress (psi)
29	Reactor feed inside drywell	M-486	10 Bend	ASTM A106 GRB	12.75	1.000	1.00	Elbow	68	3,135	9,081	12,283	18,000	18,102	21,364	31,900
30	Reactor feed inside drywell	M-486	150 Bend	ASTM A106 GRB	12.75	1.000	1.00	Elbow	116	3,135	7,998	11,249	18,000	15,996	19,247	31,900
31	Main steam inside drywell	M-464	10 Bend	ASTM A106 GRB	20.00	1.031	1.00	Elbow	300	4,113	6,002	10,415	18,000	12,004	16,417	31,900
32	Main steam inside drywell	M-464	220 Bend	ASTM A106 GRB	20.00	1.031	1.00	Elbow	761	4,113	5,623	10,497	18,000	11,246	16,120	31,900
33	Main steam inside drywell	M-464	470 TGNTBP	ASTM A106 GRB	20.00	1.031	2.04	Branch Connection 1.00 In. Reinf.Note 2	489	4,113	2,672	10,561	18,000	5,344	16,012	31,900
34	Main steam inside drywell	M-464	555 Bend	ASTM A106 GRB	20.00	1.031	1.00	Elbow	741	4,113	5,319	10,173	18,000	10,638	15,492	31,900

Table 3.9-7 (Continued)

UNIT 2
TABULATION OF MAXIMUM STRESSES FOR ORIGINAL CLASS I PIPING

Item	System	S&L Drawing	Point Number	Material	O.D. (In.)	Thickness (In.)	Intensification Factor	Component	Weight Stress (psi)Note 1	Pressure Stress (psi)	OBE Total Seismic Stress (psi)Note 1	OBE Combined Stress Sigma (psi)	OBE Allowable Stress (psi)	DBE Allowable Stress (psi)	DBE Combined Stress Sigma (psi)	DBE Yield Stress (psi)
1	Core spray pump suction 2A and 2B	M-487	115 TGNTR	ASTM A106 GR. B	12.750	.375	3.61	.250" Reinf. Branch Conn.Note 2	508	349	4,348	17,882	18,000	8,696	32,317	31,900
2	Core spray pump discharge 2A	M-488	Note 3													
3	Core spray pump discharge 2B	M-488	Note 3													
4	Core spray from drywell to RX 2-1403	M-489	5 Anchor	A312 or A376 GR. TP 304	10.750	.593	1.00	Anchor	475	4,802	897	6,174	17,460	1,794	7,071	18,200
5	Core spray from drywell to RX 2-1404	M-489	105 Anchor	A312 of A376 GR. TP 304	10.750	.593	1.00	Anchor	483	4,802	1,700	6,985	17,460	3,400	8,685	18,200
6	RHRS Pump suction 2A, 2B, 2C, and 2D	M-490	Note 3													
7	RHRS pump discharge 2A and 2B	M-491	Note 3													
8	RHRS pump discharge 2C and 2D	M-492	Note 3													
9	RHRS service water pump suction 2-1001A	M-493	225 TGNTBP	ASTM A106 GR. B	24.000	.375	8.98	Branch Conn.	548	1,449	616	11,401	18,000	1,232	17,433	31,900

Table 3.9-7 (Continued)

UNIT 2
TABULATION OF MAXIMUM STRESSES

Item	System	S&L Drawing	Point Number	Material	O.D. (In.)	Thickness (In.)	Intensification Factor	Component	Weight Stress (psi)Note 1	Pressure Stress (psi)	OBE Total Seismic Stress (psi)Note 1	OBE Combined Stress Sigma (psi)	OBE Allowable Stress (psi)	DBE Allowable Stress (psi)	DBE Combined Stress Sigma (psi)	DBE Yield Stress (psi)
10	RHRS service water pump suction 2-1002A	M-493	45 TGNTBP	ASTM A106 GR.B	24.00	.375	8.98	Branch Conn.	485	1,449	684	11,946	18,000	1,368	18,089	31,900
11	RHRS service water pump suction 2-1003A	M-493	Note 3													
12	RHRS service water pump disch. to htr. 2-1005A	M-494	Note 3													
13	RHRS service water pump disch. to htr. 2-1005B	M-494	Note 3													
14	RHRS Service Water Pump Disch. to htr. 2A	M-494	Note 3													
15	RHRS service water pump disch. to htr. 2B	M-494	Note 3													
16	RHRS htr. Exch. 2A & 2B service water return	M-495	Note 3													
17	RHRS htr. exch. 2A & 2B service water return	M-495	Note 3													
18	Reactor building cooling water	M-496	10 Bend	ASTM A106 GR. B	8.625	.322	1.00	Elbow	2,416	506	13,750	16,672	18,000	27,500	30,422	31,900
19	Reactor building cooling water	M-496	50 Bend	ASTM A106 GR. B	8.625	.322	1.00	Elbow	1,459	506	12,002	13,976	18,000	24,004	25,969	31,900

Table 3.9-7 (Continued)

UNIT 2
TABULATION OF MAXIMUM STRESSES

Item	System	S&L Drawing	Point Number	Material	O.D. (In.)	Thickness (In.)	Intensification Factor	Component	Weight Stress (psi)Note 1	Pressure Stress (psi)	OBE Total^^ Seismic Stress (psi)Note 1	OBE Combined Stress Sigma (psi)	OBE Allowable Stress (psi)	DBE Allowable Stress (psi)	DBE Combined Stress Sigma (psi)	DBE Yield Stress (psi)
20	HPCI pump suction	M-497	165 TGNTR	ASTM A106 GR. B	16.00	.375	3.47	.375" Reinf. Branch Conn.Note 2	724	644	4,463	16,169	18,000	8,926	29,843	31,900
21	Reactor feed (outside) and HPCI discharge	M-502	345 TGNT	ASTM A106 GR. B	12.750	1.000	1.00	Tangent	768	3,134	12,524	16,426	18,000	25,048	28,950	31,900
22	HPCI turbine steam supply	M-498	65 Bend	ASTM A106 GR. B	10.750	.594	1.00	Elbow	257	4,357	11,175	15,749	18,000	22,350	26,962	31,900
23	HPCI turbine steam exhaust	M-498	40 TGNTBP	ASTM A106 GR. B	20.00	.375	4.04	.375" Reinf. Branch Conn.Note 2	855	1,007	2,905	16,197	18,000	5,810	27,933	31,900
24	Main steam (outside)	M-499	Note 3													
25	Pressure suppression sheet 1	M-500	115 TGNTBP	ASTM A106 GR. B	18.00	.375	7.38	Branch Conn.	176	900	1,968	16,723	18,000	3,936	29,088	31,900
26	Pressure suppression sheet 2	M-501	10 TGNTBP	ASTM A106 GR. B	20.000	.375	2.50	.750" Reinf. Branch Conn.Note 2	46	969	5,264	14,244	18,000	10,528	27,404	31,900
27	Standby gas treatment	M-505	140 TGNTR	ASTM A211 or A245 GR. A	24.000	.140	9.40	.125" Reinf. Branch Conn.Note 2	1,007	715	219	12,240	12,360	438	14,298	31,900
28	Reactor feed inside drywell	M-486	Note 3													
29	Reactor feed inside drywell	M-486	Note 3													
30	Main steam inside drywell	M-464	Note 3													

Table 3.9-7 (Continued)

UNIT 2
TABULATION OF MAXIMUM STRESSES

Item	System	S&L Drawing	Point Number	Material	O.D. (In.)	Thickness (In.)	Intensification Factor	Component	Weight Stress (psi)[Note 1]	Pressure Stress (psi)	OBE Total Seismic Stress (psi)[Note 1]	OBE Combined Stress Sigma (psi)	OBE Allowable Stress (psi)	DBE Allowable Stress (psi)	DBE Combined Stress Sigma (psi)	DBE Yield Stress (psi)
31	Main steam inside drywell	M-464	Note 3													
32	Main steam inside drywell	M-464	Note 3													
33	Main steam inside drywell	M-464	Note 3													

Notes

1. Includes intensification except for branch components.

2. Reinforcement added to reduce stress to acceptable level.

3. Piping configuration nearly identical to Unit 1, therefore no Unit 2 analysis was made. See Unit 1 Tab. for stress.

Table 3.9-8

SAFETY RELIEF VALVE DISCHARGE LINE
GOVERNING MARK I LOAD COMBINATIONS — CLASS 3 PIPING

Combination Number	Load Combination	Code Equation Note 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 + $[(C3P2)^2 + (TQWJ)^2 + (UWCP)^2 + (UWCA)^2 + (SRVD)^2 + (SR1I)^2 + (PS2I)^2 + (SSEI)^2]^{1/2}$	9	D

Notes:
1. See ND-3650 of the ASME Code.

Table 3.9-8 (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

Table 3.9-9

SAFETY RELIEF VALVE DISCHARGE LINE (SRVDL)
CLASS 3 MARK I PIPING ACCEPTANCE CRITERIA

Code Equation Note 1	Service Level	Stress limit	Allowable Stress (ksi) Note 2		Loads Combinations Note 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 ND-3650 of the ASME Code.

2. Carbon: SRVDL, ramshead, and reducer
   Stainless: T-Quencher arms.

3. See Table 3.9-8 for Load Combinations.

Table 3.9-10

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	4.02	15.0
	10	A, B	11.21	22.5
	9	B	12.72	18.0
	9	C	17.00	27.0
	9	D	22.59	36.0
T-Quencher	8	A	2.77	16.32
	10	A, B	0	27.58
	9[Note 1]	B	15.29	19.58
	9[Note 1]	C	15.40	29.38
	9[Note 1]	D	15.42	39.16

Notes:
1. Calculated Equation 9 stresses for Service Levels B, C, and D are approximately equal.

Table 3.9-11

GOVERNING MARK I LOAD COMBINATIONS—TORUS ATTACHED PIPING

Load Combination Number	Load Combinations Notes 1, 5, & 10	ASME Code Equation Note 2
A-1	P + DW + OL	8
A-2 Note 11	TE + THAM + TD + QAB_D	10 Note 3
A-3 Note 11	TE + THAM + TD + QAB_D + SSE_D	10 Note 3
A-4 Note 11	TE_1 + $THAM_1$ + TD_1 or TD_2 or TD_3 + $PCHUG_D$ + QAB_D + SSE_D	10 Note 3
A-5 Note 11	TE_1 + $THAM_1$ + TD_1 or TD_2 or TD_3 + $CHUG_D$ + QAB_D + SSE_D	10 Note 3
A-6 Note 6 & 11	TE_1 + $THAM_1$ + TD_3 + PSO_D	10 Note 3
A-7 Note 11	TE_1 + $THAM_1$ + TD_3 + PS_D + QAB_D + SSE_D	10 Note 3
A-8 Note 4 & 11	TE_1 + $THAM_1$ + TD_3 + CO_D + OBE_D	10 Note 3
A-9	TE_2 + $THAM_2$ + TD_4 + SSE_D	10 Notes 3 & 9
A-10	TE_3 + $THAM_3$ + TD + SSE_D	10 Note 3
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 Note 7	P_O + DW + PCHUG + $PCHUG_I$ + QAB + QAB_I + SSE_I + OL	9
D-2 Note 7	P_O + DW + CHUG + $CHUG_I$ + QAB + QAB_I + SSE_I + OL	9
D-3 Note 6	P_O + DW + PSO + PSO_I + VCLO + OL	9
D-4 Note 7	P_O + DW + PS + PS_I + VCL + QAB + QAB_I + SSE_I + OL	9
D-5 Note 4	P_O + DW + CO + CO_I + OBE_I + OL	9
T-1 Note 8	1.25P + DW	8

(Sheet 1 of 3)

Notes to Table 3.9-11

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, P_O	=	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
TE_3	=	Thermal Expansion Loads Under Shutdown Cooling Conditions
THAM	=	Thermal Anchor Movement Under Normal Conditions
$THAM_1$	=	Thermal Anchor Movement Under Accident Conditions
$THAM_2$	=	Thermal Expansion Loads Under Long Term Post-LOCA Conditions
$THAM_3$	=	Thermal Expansion Loads Under Shutdown Cooling 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	=	Pre-Chug Loads
CHUG	=	Post-Chug Loads
QAB_I, QAB_D	=	Inertia and Displacement Loads, respectively, from Torus Due to SRV Discharge
CO_I, CO_D	=	Condensation Oscillation Inertia and Displacement Loads, respectively
PS_D, PSO_D	=	Pool Swell Inertia Loads, with and without Pressure Differential, respectively
$PCHUG_I$, $PCHUG_D$	=	Pre-Chug Inertia and Displacement Loads, respectively, of Torus

Notes to Table 3.9-11 (Continued)

$CHUG_I$, $CHUG_D$ = Post-Chug Inertia and Displacement Loads, respectively, of Torus

2. Equations are defined in Subsection NC-3650 of the ASME Code.

3. As an alternate, Equation 11 of the ASME Code 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.

Notes to Table 3.9-11 (Continued)

6. Only piping out to the first isolation valve needs to be evaluated.

7. The larger of LOCA and SSE combined by the SRSS method or LOCA and OBE combined by the absolute sum method is used.

8. Hydrostatic test condition. DW for all lines shall be with lines full of water at 70°F.

9. As an alternate, meet Equation 10a of ASME Code.

10. Independent dynamic loads may be combined by SRSS.

11. 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.

Table 3.9-12

APPLICABLE ASME CODE EQUATIONS AND
ALLOWABLE STRESSES FOR MARK I TORUS ATTACHED PIPING

Stress Type	ASME Code Equation Number	Service Level[Note 3]	Stress Limit	Allowable Value (ksi)[Note 4]	Governing Load Combination Number[Note 1]
Primary	8	A	1.0 S_h	15.0/18.6	A-1, T-1
Primary	9	B	1.2 S_h	18.0/22.32	B-1, B-2
Primary	9	C	1.8 S_h	27.0/33.48	C-1 Through C-3
Primary	9	D	2.4 S_h	36.0/44.64	D-1 Through D-5
Secondary	10	A	1.0 S_a	22.5/27.9	A-2 Through A-10
Secondary	10a	A	3.0 S^a	45.0/55.8	A-9
Primary and Secondary	11	A	$S_h + S_a$	37.5/46.5	Note 2

Notes:

1. Governing load combination numbers are listed in Table 3.9-11.

2. See ASME Code, Section III, Subsection NC, paragraph NC-3652.3 for combination of loads.

3. Increased allowables as defined in NUREG-0661 have been utilized for piping systems which have been classified as non-essential.

4. Carbon steel/stainless steel.

Table 3.9-13

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Table 3.9-14

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TABLE 3.9.15

SAFETY RELIEF VALVE DISCHARGE LINE GOVERNING MARK I LOAD COMBINATIONS — CLASS 3 PIPING SUPPORTS

Combination Number[Note 1]	Load Combinations[Note 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]$^{1/2}$	B
4A	WGHT + [(A1P1)^2 + (TQWJ)^2 + (UWCP)^2 + (UWCA)^2 + (SRVD)^2 + (SR1I)^2 + (SSEI)^2]$^{1/2}$	C
4B	WGHT + THL1 + [(C3P1)^2 + (TQWJ)^2 + (UWCP)$^{\wedge 2}$ + (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]$^{1/2}$	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]$^{1/2}$	D
7A	WGHT + [(C3P2)^2 + (TQWJ)^2 + (UWCP)^2 + (UWCA)^2 + (SRVD)^2 + (SR1I)^2 + (CODG)^2 + (CO2I)^2 + (SSEI)^2]$^{1/2}$	D

TABLE 3.9.15 (Continued)

SAFETY RELIEF VALVE DISCHARGE LINE GOVERNING MARK I LOAD COMBINATIONS — CLASS 3 PIPING SUPPORTS

Combination Number[Note 1]	Load Combinations[Note 2]	Service Level
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	$WGHT + THL2 + [(C3P2)^2 + (TQWJ)^2 + (UWCP)^2 + (UWCA)^2 + (SRVD)^2 + (SR1I)^2 + (PS2I)^2 + (SSEI)^2]^{1/2}$	D

Notes:

1. Combination "A" = without thermal expansion load
   Combination "B" = with thermal expansion load

2. See Table 3.9-8 nomenclature for definition of individual loads.

Table 3.9-16

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	0.86 Note 1	1.0 Note 1
Beam end connection bolts	ASTM A325	9.3	17.5
Beam end header support plate	ASTM SA516 GR. 70	15.6	22.8
Support plate guide	ASTM SA516 GR. 70	13.7	15.2
Support plate bolts Note 2	ASTM A564; F_u = 190 ksi	41.0	62.7
Support plate welds (full penetration)	ASTM SA516 GR. 70	18.0	21.0
Ramshead lug retainer	ASTM SA316 GR. 70	27.2	28.0
Intermediate Support			
Beam	ASTM A53	0.76 Note 1	1.0 Note 1
Beam end connection bolts	ASTM A325	25.7	44.0
Beam end connection plate	ASTM SA516 GR. 70	0.7 Note 1	1.0 Note 1
Collar support strut	ASTM A53 GR. B	0.45 Note 1	1.0 Note 1
Collar bolts	ASTM A325	3.7	44.0

Notes:
1. These values are the results of an interaction equation.
2. ASTM A-193 Gr. B7 bolts used as replacement material in Unit 1 and Unit 2.

Table 3.9-17

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 3.9-18

LOAD COMBINATIONS — MARK I TORUS ATTACHED PIPING SUPPORTS

Load Combination Number	Load Combinations Notes 1, 3, and 6
S-1	$DW + OL + OBE_I$
S-2	$DW + OL + QAB + QAB_I$
S-3	DW Note 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 Note 2	$DW + OL + QAB + QAB_I + SSE_I + PCHUG + PCHUG_I$
S-8 Note 2	$DW + OL + QAB + QAB_I + SSE_I + CHUG + CHUG_I$
S-9	$DW + OL + OBE_I + CO + CO_I$
S-10 Note 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 Note 2 & 7	$DW + OL + QAB + QAB_I + PCHUG + PCHUG_I + TE_I + THAM_I + TD_3$ Note 4 $+ QAB_D + PCHUG_D$
S-15 Note 2 & 7	$DW + OL + QAB + QAB_I + CHUG + CHUG_I + TE_I + THAM_I + TD_3$ Note 4 $+ QAB_D + CHUG_D$
S-16 Note 2 & 7	$DW + OL + QAB + QAB_I + SSE_I + PCHUG + PCHUG_I + TE_I + THAM_I + TD_3$ Note 4 $+ QAB_D + SSE_D + PCHUG_D$
S-17 Note 2 & 7	$DW + OL + QAB + QAB_I + SSE_I + CHUG + CHUG_I + TE_I + THAM_I + TD_3$ Note 4 $+ QAB_D + SSE_D + CHUG_D$
S-18 Note 7	$DW + OL + OBE_I + CO + CO_I + TE_I + THAM_I + TD_3$ Note 4 $+ OBE_D + CO_D$
S-19 Note 2 & 7	$DW + OL + QAB + QAB_I + SSE_I + PS + PS_I + VCL + TE_I + THAM_I + TD_3$ Note 4 $+ QAB_D + SSE_D + PS_D$
S-20 Note 7	$DW + OL + PSO + PSO_I + VCLO + TE_I + THAM_I + TD_3$ Note 4 $+ PSO_D$
S-21 Note 7	$DW + OL + QAB + QAB_I + SSE_I + TE + THAM + TD + QAB_D + SSE_D$

Table 3.9-18

LOAD COMBINATIONS — MARK I TORUS ATTACHED PIPING SUPPORTS

Load Combination Number	Load Combinations Notes 1, 3, and 6
S-22	$DW + OL + SSE_1 + SSE_D + TE_2 + THAM_2 + TD_4$
S-23	$DW + OL + OBE_1 + OBE_D + TE_3 + THAM_3 + TD$
S-24	$DW + OL + SSE_1 + SSE_D + TE_3 + THAM_3 + TD$

1. See Table 3.9-11 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 non-water 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 3.9-19

REACTOR INTERNAL PRESSURE DIFFERENTIALS

Component	Δ P At Turbine-Generator Design Power*	Maximum Δ P Following A Steam Line Break[Note 2, 4]	Maximum Δ P Following a Recirculation Line Break*
Shroud support	N/C [25]	43.0 [43]	N/C [25]
Guide tube	N/C [17]	29.5 [30]	N/C [17]
Core plate	N/C [17]	29.5 [30]	N/C [17]
Lower shroud	N/C [25]	43.0 [43]	N/C [25]
Upper shroud	N/C [8]	20.0 [20]	N/C [8]
Shroud head	N/C [8]	20.0 [20]	N/C [8]
Dryers	N/C [2]	3.4[Note 3] [4[Note 1]]	N/C [2]
Channel box	N/C [9]	14.9 [16]	N/C [9]

Notes:

1. Evaluated from the outside steam-line break described in Chapter 15.
2. An additional thermal-hydraulic analysis was performed, for the results see References 12, 13, and 14.
3. Conservatively evaluated at a bounding 120° FFWT.
4. Values shown outside "[ ]" are based on Reference 29.

[ ] Values shown outside "[ ]" are based on 2957MWt. LPU Analysis at 108% Core Flow. Values inside "[ ]" are based on 2511MWt, pre-LPU.

* Bounded by ΔPs following a Steamline break.

N/C Not calculated in LPU Analysis.

Table 3.9-20

PRESSURE FORCES ACTING ON MAJOR REACTOR INTERNAL COMPONENTS

Major Component	Pressure Force[Note 1]
Shroud support	P_1-P_4
Guide tube	P_1-P_3
Core plate	P_1-P_3
Lower shroud	P_1-P_4
Upper shroud	P_3-P_4
Shroud head	P_3-P_4
Jet pump diffuser	P_1-P_4

HISTORICAL
This Table contains Historical Information Only

Notes:

1. Subscripts, refer to model nodes shown on Figure 3.9-11.

Table 3.9-21

RESULTANT CORE SHROUD LATERAL LOADS FOR A RECIRCULATION SUCTION LINE BREAK

Shroud Weld Designation	Shroud Weld Elevation (Inches)	Maximum Force (Kips)	Maximum Moment (Inch-Kips)
	434.26	0.00	0.00
	393.61	2.50	50.80
H1	391.38	2.90	60.05
	375.00	5.83	128.01
H2	357.88	8.88	315.74
H3	355.38	9.33	343.16
	317.20	16.12	761.94
	279.92	23.26	1494.91
H4	266.38	27.54	1887.56
	242.65	35.11	2576.91
	205.38	55.79	4255.01
H5	191.13	69.62	5255.30
H6	187.13	73.52	5536.79
	173.83	86.49	6474.08
	148.67	128.20	9146.43
H7	131.50	153.44	11674.81
	120.88	169.07	13243.75
Maximum Load for Extended Load Line Limit[2]			
N/A	120.88	235.55 (179.8 x 1.31)	17462.3 (13330 x 1.31)

Notes:

1. Determined from the TRACG recirculation line break analysis (Reference 13).

2. From LPU (Licensed Power Uprate) Analysis @ 2957MWt, Reference 38, where 1.31 is the Limiting Load Multiplier for the Long Operating Condition in the MELLL region with a feedwater temperature reduction option.

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 EQ equipment that require seismic qualification (referred to in this section as replacement). Seismic qualification of other components and piping systems are described in Sections 3.7, 3.8, and 3.9. [3.10-1]

The original seismic criteria and design bases for Quad Cities Station have not changed. Specific seismic requirements for Class I electrical equipment and instrumentation have also not changed, although the anchorage and support for Class I electrical equipment were modified later for several motor control centers, switchgears and racks to satisfy the requirements of IE Notice 80-21. Instrumentation mounted on these racks has not undergone additional seismic qualification beyond the original design. [3.10-2]

Seismic qualification of replacement instrumentation and electrical equipment satisfy the requirements of IEEE 344-1975 and Regulatory Guide 1.100 and plant design criteria as given in DC-SE-01-DQ. [3.10-3]

The Quad Cities station work control system data base provides a list of components and instrumentation with their safety classification. [3.10-4]

Quad Cities Station has received the SER[1] on the Seismic Qualification Utility Group (SQUG) Program. To address USI A-46, SQUG has developed the Generic Implementation Procedure[2] (GIP), Revision 2, or imply "GIP-2" which relies primarily on the use of existing earthquake and testing experience data to verify the seismic adequacy of generic classes of equipment. The NRC endorsed GIP-2 with comments via Supplement No. 1 to GL 87-02. Quad Cities Station committed to the implementation of GIP-2 with SSER No. 2 comments in a letter[3] dated September 21, 1992. The NRC subsequently approved Quad Cities's approach and schedule[4]. The SER identifies that the implemented Quad Cities' SQUG Program meets the purpose and intent of the criteria for resolution of USI A-46. [3.10-4a]

3.10.1 Seismic Qualification Criteria

The original seismic design criteria for the Quad Cities Station were developed by John A. Blume and Associates. The details of generation of seismic input loads are provided in Section 3.7. [3.10-5]

The design criteria used for seismic qualification of both the original and replacement Class I electrical equipment and instrumentation are as follows:

A. All Class I electrical equipment must be capable of performing their Class I function under normal plant operating conditions and during and after a safe shutdown earthquake (SSE).

B. Primary pressure boundary devices must retain structural and pressure integrity during normal operation and during and after an SSE.

C. Non-Class I components whose failure may cause failure to a Class I component must maintain structural integrity during normal plant operation and during and after an SSE.

Specific criteria and qualification methods for individual components depend on the location and function of the component within a given system. Because devices at Quad Cities are located at different elevations and various locations, they were qualified in the original seismic qualification program for the worst possible earthquake loads, thus assuring generic qualification rather than location-specific qualification. Seismic qualification of replacement components and further discussion of this is provided in Section 3.10.2.

3.10.2 Methods and Procedures for Qualifying Electrical Equipment and Instrumentation

The methods and procedures for qualification of both original and replacement electrical equipment and instrumentation are provided in this subsection.

3.10.2.1 Seismic Qualification of Original Electrical Equipment and Instrumentation

Vibration tests and/or analyses were performed on instrumentation, devices, panels and racks supplied by General Electric in order to qualify them for use in critical safety applications. The tests were part of a program to demonstrate the acceptability of the Class I equipment for operation during the SSE. [3.10-6]

Tests and analyses were performed to prove the capability of the equipment to withstand seismic vibrations. The equipment was divided into four main classes: instruments and instrumentation and control devices; enclosures, panels and racks; primary pressure boundary devices; and metal clad switchgear.

The instruments and instrumentation and control devices were further subdivided into 72 generic types such as differential pressure transmitters, temperature elements, pressure switches, electrical relays, and level switches. Each generic type was operationally tested during vibration to well above the specified requirements. For the purpose of demonstrating seismic capability, sinusoidal vibrations with accelerations of 1.5g in two mutually perpendicular horizontal axes and 0.5g in the vertical axis were applied to each piece of equipment when attached to the vibrator in a manner similar to its mounting to the major structural members of the building. The instrument or device was considered to be acceptable if it demonstrated performance of its Class I function during the application of each of the above accelerations over the frequency range of 5 to 30 Hz.

The primary pressure boundary items, such as condensate chambers and temperature wells were at design pressure and temperature while being subjected to seismic Class I conditions.

The enclosures, panels and racks were also subdivided into generic types such as vertical boards, bench boards and local racks. This was possible due to the standard enclosure designs employed together with the resulting response accelerations due to resonance magnification. Enclosures and panels are covered in Section 3.10.3.

3.10.2.1.1 Testing Program

Certain Class I equipment such as panels containing ECCS electrical controls and instrumentation were not amenable to analytical solution. The Quad Cities panels are identical to panels used in an experimental vibrations program to determine the capabilities of mounted equipment to operate satisfactory under such conditions. The results of this test program were documented in a GE Topical Report and submitted to the AEC.

The bases for the criteria used to establish the seismic inputs for the experimental vibration program are as follows: [3.10-7]

A. The value of acceleration used for test purposes was selected to be greater than the maximum calculated floor acceleration at any Class I instrument location within the reactor building or the turbine building for an SSE.

B. The high end of the frequency range (5 - 33 Hz) was chosen to be high enough to cover predominant frequencies of all earthquake-induced floor vibrations. The low end was chosen to be close to the major portion of the seismic spectrum and low enough to cover the range of expected resonances in the instrumentation or equipment being qualified.

C. Comparison of test acceleration values with actual in-plant accelerations showed the test values to be conservative. The actual maximum floor accelerations calculated for any of the Class I instrument locations did not exceed 0.4 g horizontal considering an SSE. (See Figures 3.10-1 and 3.10-2). Amplification contributed by the instrument mounting depends on frequency and varies from unity at frequencies up to 10 Hz to a maximum of approximately 2 at frequencies above 26 Hz. Thus, actual acceleration at instrument location was not expected to exceed 0.4 g below 10 Hz.

D. Failure of specific instruments to qualify under the general 1.5 g specification were re-evaluated for specific instrument environment and function to determine actual capability to perform satisfactorily under SSE conditions in the plant. Such evaluations considered the floor accelerations calculated for the actual location of the instrument in question. Where malfunction was probable at the predicted amplified acceleration values for the instrument, then effects of failure on system action and safety were evaluated to determine whether the instrument was acceptable for its intended service.

3.10.2.1.1.1 Instrumentation and Control Devices

All instrumentation and control devices were tested in an operational condition. The instrumentation, for instance, was supplied with appropriate input signals and/or trip inputs and monitored with the trips set within 2% (upscale and downscale) of the levels. Relays were monitored in the energized and de-energized condition for both normally open and normally closed contacts. Pressure, level, and flow switches were vibrated while providing simulated input signals that approached set points within 2% of operational settings and the switch contacts monitored for false closure or opening (spurious trips). The instrument or device was mounted the way it is mounted in its actual application in the plant.

During seismic scans the devices were monitored for resonant frequencies using either accelerometers, strobe lights, or both. The accelerometers were connected to charge amplifiers which were used to drive a recorder for permanent recording of data. A meter output strobe light aided in detection of the resonant frequencies and response modes of the devices. The detection and exploration of the resonant frequencies was first made over the 5 — 33 Hz frequency range to detect possible weak points that could result in failure during subsequent endurance and higher acceleration runs.

Vibration endurance and maximum acceleration scans over the frequency range of 5 — 33 Hz were then conducted to subject the hardware to the maximum specified accelerations of 1.5 g horizontal in two perpendicular axes and 0.5 g vertical. Also each instrument or device was tested at 33 Hz at increasing amplitudes to the maximum acceleration without malfunction.

3.10.2.1.1.2 Primary Pressure Boundary Devices

Pressure retaining components that are part of reactor primary pressure boundary, the failure of which could cause the malfunction of an essential device or system, were tested and/or analyzed to show pressure and structural integrity under seismic conditions.

3.10.2.1.1.3 Metal-Clad Switchgear

The metal-clad switchgear utilized in the Quad Cities plant can easily withstand the seismic induced forces to which they might be subjected at Quad Cities, both from a structural and functional standpoint. The various switchgear have been subjected to analytical evaluations or vibration tests over a period of years. [3.10-8]

Typical of the Quad Cities equipment is the AKD-5 low-voltage switchgear and assembled Type AK breakers. This equipment was tested over the frequency range of 5 — 500 Hz in each of three directions and was also subjected to simultaneous horizontal and vertical accelerations by mounting the equipment at an angle on the test machine. Input accelerations of about 0.5 g were used to determine natural frequencies, and in addition, the breakers were individually subjected to much greater accelerations and shock tests than were performed on the switchgear. Results of these tests showed that:

A. AKD-5 switchgear is shock resistant to impacts producing accelerations up to 40 gs;

B. AKD switchgear is shock resistant to impacts producing accelerations up to 100 gs;

C. AK-2A-25 and AK-2A-50 breakers remain operable during shock at accelerations up to 15 gs;

D. AK-2-25 and AK-2-50 breakers remain operable during shock at accelerations up to 15 gs;

E. The AK-50 breaker lowest resonant frequency is 29 Hz and it operates successfully at this frequency at 5.0 g input; and

F. The AK-25 breaker lowest resonant frequency is 44 Hz and it operates successfully at this frequency at 3.0 g input.

The high voltage switchgear has been analyzed and the equipment and all components have natural frequencies of 22 Hz or greater. In addition, the analyses show that for an acceleration of about .4 g, the maximum building input acceleration, the highest stresses for combined static and seismic loads is about 7000 psi compared to an allowable of 27,000 psi. Furthermore, the breakers will not open or close falsely under acceleration shocks up to 3 gs.

The above discussion briefly summarizes results of tests and analyses on the typical type switchgear utilized in the Quad Cities installations. As shown, the highest period of natural vibration of any of the components is about .044 seconds. Hence, none of the equipment is in resonance with any of the significant vibration modes of the building, and the capability of the equipment to withstand accelerations is far in excess of those accelerations that could occur on the various building floors. It is therefore concluded that the switchgear will operate satisfactorily for both operating basis earthquake (OBE) and the SSE conditions.

3.10.2.2 Seismic Qualification of Replacement Electrical Equipment and Instrumentation

Seismic qualification of replacement equipment is demonstrated using one of the following three general methods. [3.10-9]

1. Predict the equipment's performance by analysis.
2. Test the equipment under simulated seismic conditions.
3. Qualify by combined test and analysis.

The choice of qualification method is based on practicality of the method for the type, size, shape and complexity of the equipment and reliability of the conclusion.

Seismic response spectrum curves have been generated for both the OBE and SSE at various plant locations for use in the seismic qualification of equipment as input loads. [3.10-10]

3.10.2.2.1 Qualification by Analysis

An equipment may be qualified by analysis if it is a relatively simple piece of equipment and can be approximated by a mathematical model. The most common methods of analysis are static and dynamic analyses. [3.10-11]

3.10.2.2.1.1 Static Analysis

If the equipment is determined to be rigid, then static analyses is performed using the maximum floor accelerations of the dynamic loads, its dead weight and other applicable static loads. For a simple structure such as a pipe stand or a cantilever bracket, a simple analysis such as hand calculation is sufficient to achieve required results. If the equipment is complex such as a control panel, it is analyzed using appropriate computer programs. The calculated stresses are compared against allowables for the material.

3.10.2.2.1.2 Dynamic Analysis

Flexible equipment are typically qualified by dynamic analysis. Dynamic analysis is performed using the finite element structural analysis technique using one of two methods: the response spectrum method or the time history method. Quad Cities equipment has been analyzed using both these methods. The structure is modeled using the appropriate material and geometric properties for elements (beams, plates, etc.) representing the actual equipment. The system stiffness and mass matrices are formed in the applicable program. The general equation of motion in the matrix form is given by:

$$M\left(\ddot{X}\right)+C\dot{X}+KX=0$$

where:

M = the mass matrix

X = the column vector of displacement relative to ground

C = the damping matrix

K = the stiffness matrix

Y = the column vector of ground acceleration

$\dot{}$ = the first derivative with respect to time

$\ddot{}$ = the second derivative with respect to time

Seismic analysis is performed using the above equation and other techniques to uncouple this equation.

3.10.2.2.2 Qualification by Test

If the equipment is flexible and too complex to be represented properly by structural elements, then the equipment is qualified by test. Testing is also performed where operability assurance is required which can not be established analytically under a seismic loading condition. Seismic tests are performed by subjecting the equipment to vibratory motion which conservatively simulates that postulated at the equipment mounting during an SSE. The tests procedures follow the recommendations of IEEE 344-1975.

3.10.2.2.3 Qualification by Combination of Test and Analysis

Some types of Class 1E equipment cannot be practically qualified by analysis or testing alone. This is because of the size of the equipment, its complexity, or the large number of similar configurations. Various techniques recommended by IEEE 344-1975 are used to seismically qualify equipment using both test and analysis. [3.10-12]

3.10.3 Methods and Procedures of Analysis or Testing of Supports of Electrical Equipment and Instrumentation

This subsection describes the methods used to seismically qualify items on which Class I electrical equipment and instrumentation are mounted.

3.10.3.1 Seismic Qualification of Supports for Original Electrical Equipment and Instrumentation

The original qualification was performed under the same test program described in Section 3.10.2.1.1.

3.10.3.1.1 Panels

A completed 10 1/2-foot relay panel similar in design to the Class I panels used in most BWR plants was tested. This panel was vibrated through the frequency range of 5 — 33 Hz with a steady-state low acceleration sinusoidal input. A resonant frequency occurred at 15 Hz resulting in an output acceleration of 13.5 gs normal to the front face at the center of the panel when normalized from a lower level to the 1.5 g input acceleration at the panel base. Installation of two truss type braces inside the panel reduced the acceleration to 6.0 gs and moved the corresponding resonant frequency to 23 Hz. Above this frequency the acceleration decreased to the base input at about 30 Hz. The preceding response was within the required limits for the panel under test because the devices mounted on the panel are all capable of higher acceleration rates over the test frequency range. [3.10-13]

3.10.3.1.2 Local Racks

A local rack type having the longest unsupported span and the greatest amount of weight was selected as the generic design out of the many specified. It was a 6-foot wide rack constructed of 3-inch steel channel. The testing indicated a resonance at 24 Hz with a peak acceleration of 13.8 gs when normalized from its lower input acceleration to an input of 1.5 gs at the base of the rack.

Two types of bracing were used to reduce the acceleration levels of this rack. First, two braces made from angle iron and spaced 2 feet apart were installed. This reduced the peak acceleration to 12 gs. Next, a single brace made of 4-inch steel channel was installed at the center of the rack. This brace reduced the peak acceleration to an acceptable level of 5.5 gs at 26 Hz.

3.10.3.2 Seismic Qualification of Supports for Replacement 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 are: battery racks, control consoles, cabinets, instrument racks, panels and pipe stands. Where possible supports are qualified by test with the equipment installed and operable. Otherwise, dummy mass is used to simulate equipment mass effects and dynamic coupling to the supports. In case of analysis, the stresses at all support points in parts like motor hold down bolts, base plate hold down bolts, support pads, pedestals and foundations, etc., are checked against the allowables of the applicable codes. [3.10-14]

If new and/or replacement equipment is mounted on an existing support, then the impact of the additional mass is evaluated. The existing support is re-analyzed to the requirement of the IEEE 344-1975 if there is significant impact on the dynamic characteristics of the support due to the replacement item.

3.10.4 Qualification Results

The subsection provides results of the Quad Cities seismic qualification program.

3.10.4.1 Qualification Results for Original Electrical Equipment and Instruments

The equipment subjected to the seismic tests described in Section 3.10.2.1 were of the same types and models as those purchased for use in the Quad Cities station, thus the components tested have seismic tolerance equal to the components being used in the Quad Cities station. [3.10-15]

Table 3.10-1 lists the Class I control panels by title and type and their maximum horizontal acceleration limits. The horizontal acceleration limit given for each panel corresponds to the acceleration limit of the instrument or device mounted in the panel having the lowest maximum acceleration capability without malfunction. The type of instrument or device that establishes that limit is listed opposite the maximum acceleration limit.

Table 3.10-2 lists the Class I local instrument racks and enclosures by title and type and maximum horizontal acceleration limits, as determined for Table 3.10-1.

Table 3.10-3 lists instruments and devices that were tested together with test results in terms of maximum usable acceleration.

Table 3.10-4 shows the calculated floor accelerations (see Figures 3.10-1 and 3.10-2) at the various locations of instruments that did not meet the generic 1.5 g level. This evaluation criteria takes into account appropriate worst case amplification contributions from the supporting structures. The accelerations applied to the instruments in question are given in Table 3.10-4. The following is a description of results for these instruments.

3.10.4.1.1 Reactor Level Switch

The switch units that provide initiation of the emergency core cooling systems (high pressure coolant injection (HPCI), core spray, etc.) are normally open, capillary mercury magnetic switches that close upon reactor low coolant level (lower than scram level) to initiate the emergency core cooling systems. The switches remain open when the reactor coolant level is above the trip setting. [3.10-16]

During the vibration test, at all frequencies from 5 to 33 Hz and accelerations greater than 0.5 g the switches all changed state, i.e., the normally closed switches opened and the normally open switches closed while the vibrations were sustained. No damage to the switches occurred and all switches tested returned to their normal state when the vibrations ceased. Since the switches are located in the plant at points where the required g level will not exceed 0.5 g, the switches are considered qualified and acceptable.

3.10.4.1.2 Level Switch (Condensate Storage Tank Level

The level switches are mercury magnetic switches that remain stable out to an acceleration level of 0.5 g at all test frequencies.

The switches are mounted near the bottom of the tank close to ground level where they are expected to see less than 1 g acceleration during an SSE.

The use of the mercury switches in the condensate storage tank level application does not compromise plant safety because the switch at worst condition will merely switch HPCI and reactor core isolation cooling (RCIC) suction from the condensate storage tank to the suppression pool without loss of coolant flow.

3.10.4.1.3 Pressure Switch (Main Steam Line Flow)

The main steam line flow switches are differential pressure activated mercury magnetic switches. The switches initiate main steam line isolation valve closure and subsequent reactor scram upon excess flow in the main steam lines.

The test showed that the switches have excellent stability at all test frequencies out to accelerations of 11 g in the vertical and one horizontal X plane, but becomes resonant in the horizontal Z plane normal to the face of the instrument (and mounting panel) at approximately 32 Hz. The switches become unstable at acceleration of 1 g or greater at 32 Hz or greater. The switches are stable at greater than 1.5 g below the resonant frequency. Additional structural members were installed in the mounting panels to assure that the seismic tolerance of the switches will not be exceeded at frequencies up to 30 Hz maximum and 1.5 g maximum applied at the base of the panel as specified in the acceptance criteria. The switches are therefore considered to be acceptable for the application.

3.10.4.2 Qualification Results for Replacement Electrical Equipment and Instrumentation

Qualification of the replacement components are performed by using both analytical methods and/or testing, as identified in the preceding subsection, in accordance with the requirements of IEEE 344-1975. [3.10-17]

Review of test and analysis reports filed for such qualification indicate that the equipment and instrumentation meet or exceed the required design intent. The extensive documentation indicate that the equipment and instrumentation will maintain their structural and functional integrity and, therefore, their capability to perform the required safety function during and after an SSE.

3.10.5 References:

1. “Plant Specific Safety Evaluation for USI A-46 Program Implementation at Quad Cities Nuclear Power Station, Units 1 and 2”, U. S. Nuclear Regulatory Commission, September 21, 1999.

2. Generic Implementation Procedure (GIP), Revision 2 as corrected on February 14, 1992, Seismic Qualification Utility Group (SQUG)

3. Letter from M. A. Jackson (CECo) to Dr. T. Murley (NRC), September 21, 1992, Response to Supplement 1 to Generic Letter (GL) 87-02, “Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors”, SQUG Resolution of USI A-46

4. Letter from C.P. Patel (NRC) to T.K. Kovach (CECo), November 20, 1992, “Evaluation of the Dresden Nuclear Power Station, Units 2 and 3, Quad Cities Nuclear Power Station, Units 1 and 2, and Zion Nuclear Power Station, Units 1 and 2, 120-Day Response to Supplement No. 1 to Generic Letter 87-02.

Table 3.10-1

CONTROL ROOM PANELS

Identification		Description	
Name	Type	Limiting Part	Max. Horizontal Acceleration Normal to Front Face at Panel Without Failure
Reactor and containment cooling	Bench board	Controller	5 gs
Reactor cleanup and circulation	Bench board	HFA relays	7 gs
Startup neutron monitor	3-section panel	IRM	3 gs
Power range monitor	5-section panel	APRM	1.5 gs
Protection system Ch. A	Vertical board	HFA relays	7 gs
Protection system Ch. B	Vertical board	HFA relays	7 gs
Process instr. aux. cleanup	Single section panel	GE/MAC instr.	3 — 11 gs
RHR and core spray Ch. B	Vertical board	HFA relays	7 gs
RHR and croe spray Ch. B	Vertical board	HFA relays	7 gs
HPCI relays	Vertical board	HFA relays	7 gs
Inboard isol. valve relays	Hoffman encl.	HFA relays	7 gs
Outboard isol. valve relays	Hoffman encl.	HFA relays	7 gs
Steam leak detector A relays	Vertical board	HFA relays	7 gs
Steam leak detector B relays	Vertical board	HFA relays	7 gs

Table 3.10-2

LOCAL INSTRUMENT RACKS AND ENCLOSURES

IDENTIFICATION	DESCRIPTION		
Name	Type	Limiting Part	Max. Horizontal Acceleration Normal to Front Face at Panel Without Failure
LPCI/cont./core spray	Local rack	Flow transmitter	2 gs
Reactor instrument and prot. rack A	Local rack	Pressure transmitter	2 gs
Ractor instrument and prot. rack B	Local rack	Pressure transmitter	2 gs
SRM/IRM preamp rack A	Hoffman enclosure	IRM preamp	8.5 gs
HPCI instrument rack	Local rack	Flow transmitter	2 gs
Recirculation pump instrument rack A	Local rack	Flow transmitter	2 gs
Recirculation pump instrument rack B	Local rack	Flow transmitter	2 gs
SRM/IRM reamp rack B	Hoffman enclosure	IRM preamp	8.5 gs
Solenoid fuse panels	Hoffman enclosure	Fuse	15 gs

Table 3.10-3

SUMMARY OF RESULTS - CLASS I EQUIPMENT SEISMIC TEST

Description	Maximum Usable g Level
Voltage preamplifier	8.5
TIP ball valve	25
IRM dectector	> 1.5 (Maximum not determined)
Local rack (typical)	0.5 in vertical and 1.5 in horizontal
Reactor level switch	0.5 Note 1
Temperature control switch	12
Contactor (CR 1050)	12
Indicator and trip unit	15
PRM fixed incore detectors	> 1.5 (Maximum not determined)
TIP Shear valve assembly	10
Timer	9
Temperature switch	4
Temperature switch	5
Pressure transmitter	10
Relay (HFA) (initiation logic)	4.75

Table 3.10-3 (continued)

SUMMARY OF RESULTS - CLASS I EQUIPMENT SEISMIC TEST

Description	Maximum Usable g Level
Flow switch (RHR minimum flow bypass)	4
Pressure switch	11
Flow switch (standby liquid flow)	15
Flow switch (HPCI minimum flow bypass)	4
Fuse	15
Flow converter	15
Flow auxiliary unit	11
Source range monitor	3
Intermediate range monitor (dc)	0.5 in vertical and 1.5 in horizontal
Power supply (20 Vdc)	0.5 in vertical and 1.5 in horizontal
Log. radiation monitor	3
Intermediate range monitor	3
Senser converter	15
Pressure switch (reactor pressure) (scram) (core spray and LPCI valve open permissive)	15
Temperature element	15

Table 3.10-3 (continued)

SUMMARY OF RESULTS - CLASS I EQUIPMENT SEISMIC TEST

Description	Maximum Usable g Level
Level switch (condensate storage tank level)	0.5 Note 1
Pressure switch drywell pressure (scram)	15
Pressure switch (main steam line flow)	1.0 Note 1
Pressure switch drywell pressure (core cooling initiate)	15
Pressure switch	2
Relay (CR 120A)	12
Relay (CR 2820)	25
Switch, SBM (manual start-stop, etc.)	25
Relay (CR 120K)	25
Relay time delay (CR 120KT)	12
IRM trip auxiliary	12
Scram solenoid fuse panel	10
Channel B RHR and core spray	0.5 in vertical and 1.5 in horizontal
PRM system	0.5 in vertical and 1.5 in horizontal
IRM range switch	8.5

Table 3.10-3 (continued)

SUMMARY OF RESULTS - CLASS I EQUIPMENT SEISMIC TEST

Description	Maximum Usable g Level
Gamma chamber	0.5 in vertical and 1.5 in horizontal
Controller	5
Manual loading station	2
Millivolt converter	3
Pressure transmitter	2
Flow transmitter	2
Pressure transmitter	12
Dual alarm	5
Proportional amplifier (flow summer)	3
Square root converter	11
Power supply	11
LPRM	0.5 in vertical and 1.5 in horizontal
APRM	0.5 in vertical and 1.5 in horizontal
ICPS	0.5 in vertical and 1.5 in horizontal
RBM	0.5 in vertical and 1.5 in horizontal
Selector switch, thermocouple	25
Switch, (oil-tight)	20

Table 3.10-3 (continued)

SUMMARY OF RESULTS - CLASS I EQUIPMENT SEISMIC TEST

Note 1: The maximum anticipated g levels at the instrument location are enveloped by the tested g levels. Also see Table 3.10-4.

Table 3.10-4

FLOOR ACCELERATIONS AT INSTRUMENT LOCATIONS

Instrument	Elevation (Location)	Floor Accel. (Max.)	Instrument Support Trans-missability	Max g On Instrument Case	Max g For Correct Operation
Reactor Level Switch	623 ft. (RB)	<0.4g	<2	<0.8 Note 1	0.5
Condensate Storage Tank Level Switch	595 ft. (TB)	0.24	1.0	0.24	0.5
Main Steam Line Flow Switch	554 ft. (RB)	0.24	< 2	<0.48	1.0

Note 1: For additional detail see Section 3.10.4.1.1.

3.11 ENVIRONMENTAL QUALIFICATION OF ELECTRICAL EQUIPMENT

The environmental qualification (EQ) of electrical equipment is performed in accordance with the guidelines of NRC Office of Inspection and Enforcement Bulletin (IEB) 79-01B, and the requirements of 10 CFR 50.49. The EQ program for Quad Cities, which was submitted in the Response to IEB 79-01B, was approved by the NRC. That program, as it has evolved and is currently being implemented, is described in the following sections.[1],[3.11-1]

3.11.1 Equipment Identification and Environmental Conditions

Equipment within the scope of the EQ program includes safety-related electrical equipment, nonsafety-related electrical equipment, and post-accident monitoring equipment as defined by paragraphs (b)(1), (b)(2), and (b)(3), respectively, of 10 CFR 50.49. Methods used to identify this equipment, as well as the methods to determine the EQ program environmental conditions, are discussed in the following subsections.

3.11.1.1 Identification of Safety-Related Electrical Equipment Requiring Environmental Qualification

The safety-related electrical 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 electrical equipment. Therefore, not all safety-related electrical equipment must be environmentally qualified. [3.11-2]

Only the equipment's ability to perform its safety functions (i.e., those functions delineated in items (i) through (iii) of 10 CFR 50.49 (b)(1)) must be considered in environmental qualification. The equipment's capability to perform nonsafety-related functions need not be assured.

The methodology used to identify electrical equipment designated as EQ-related is as follows:

A. All design basis events such as loss-of-coolant accident (LOCA), main steam line breaks (MSLB) inside containment, and high energy line breaks (HELBs) outside containment were reviewed. [3.11-3]

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 six functions considered for accident mitigation were:

   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 which must remain functional in these systems was identified by review of system descriptions and appropriate drawings (piping and instrumentation drawings (P&IDs), schematics, electrical single-line diagrams and control logic diagrams). System/component failure analyses were performed to identify the electrical equipment which requires 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 mitigate the consequences of an accident in order for the system to accomplish its design basis safety function. Several other systems only require 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 were 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 that they are required to perform a safety function; these items were also judged outside the scope of 10 CFR 50.49.

F. For electrical equipment designated as EQ-related, 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 station's work control system data base. The station's work control system data base 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. [3.11-4]

The methodology used to identify safety-related electrical equipment designated as EQ in the station's work control system data base is in full compliance with the requirements of NRC IEB 79-01B Supplements 1 and 2 and 10 CFR 50.49. Therefore, the station's work control system data base 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 impact the accomplishment of safety functions. An evaluation of the possibility of failure of nonsafety-related equipment in a manner detrimental to safety equipment used a combination of methods which are summarized below: [3.11-5]

A. Safety-related electric equipment as defined in paragraph (b)(1) of 10 CFR 50.49 were identified. See 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 the safety system operation, systems interaction, and the operation of equipment within each safety system. This failure analysis identified all safety-related and non safety-related auxiliary systems and equipment that are necessary for the required operation of the safety-related system or equipment. This effort included a review of the plant safety analyses, Technical Specifications, Emergency Operating procedures, P&IDs, schematics, wiring diagrams, electrical-line diagrams, and control logic diagrams.

C. Based on the preceding failure analysis, nonsafety-related electrical equipment having a failure mode under postulated environmental conditions which prevent accomplishment of safety functions are designated as EQ in the station's work control system data base.

The review methodology is judged to adequately identify electrical equipment within the scope of 10 CFR 50.49(b)(2).

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 considered by EGC to be classified as Regulatory Guide 1.97, Revision 2, Category 1 or Category 2 items which are located in harsh post-accident areas are judged to be within the scope of paragraph (b)(3) of 10 CFR 50.49 and are included in the environmental qualification program. These items are designated as EQ in the station's work control system data base. [3.11-6]

UFSAR Section 7.5 provides additional information regarding Regulatory Guide 1.97 compliance.

3.11.1.4 Environmental Conditions

The following sections describe the zones and selection of environmental parameters for Quad Cities. It must be emphasized that in each case the harsh and normal environmental conditions represent conservative bounding conditions for these zones. When environmental conditions are required for a specific location, a more detailed analysis may be performed to establish the environmental conditions for the location. Therefore, the qualification environments specified in the qualification records may not agree with the zone parameters identified in the zone tables. In such cases, unique calculations have been performed to justify these conditions and are part of the environmental qualification records.

The plant has been divided into forty distinct zones. Each of these has been assigned a unique zone number. Figure 3.11-1, Sheets 1 through 10 (Drawing M-4A) depict the location and boundaries of these EQ zones within the plant, the temperature, pressure, humidity, radiation parameters for each zone under normal conditions, and a postulated design basis accident (HELB and LOCA, as applicable). [3.11-7]

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: [3.11-8]

A. Temperatures above 120°F,

B. Total radiation exposure greater than 5 x 10^4 rads, or

C. Pressure transient resulting from a LOCA or HELB inside the drywell, the pressure suppression pool, and the main steam tunnel.

NRC IEB 79-01B defines the postulated accident conditions to be addressed as the LOCA/HELB inside the containment and the HELB outside the containment.

For the LOCA/HELB inside containment, worst-case environmental conditions are established by the LOCA resulting from a double-ended recirculation line break.

The postulated HELBs outside the drywell were determined in the Special Report Number 12, "Analysis of Effects of Pipe Break Outside Primary Containment."[2] The HELBs considered occurred in the following lines:

A. Main steam,

B. Reactor feedwater,

C. High-pressure coolant injection (HPCI) (steam line),

D. Reactor water cleanup, and

E. Reactor core isolation cooling (RCIC) (steam line).

In support of the response to IE Bulletin 79-01B, studies were performed to establish equipment-integrated radiation doses as a result of the postulated accidents. [3.11-9]

3.11.1.4.2 Mild Post-Accident Areas

A mild environment is defined as meeting all of the following criteria: [3.11-10]

A. 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 will result only in minor changes in pressure in the mild pressure areas.

Exelon Generation Company's position, with respect to areas where the temperature does not exceed 120°F due to a DBA, is that these are mild temperature areas 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 temperature of equipment represents no significant change from the normal temperature for equipment located in these areas. For all equipment located in these areas, the mild temperature environment 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 will be 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. In addition, operating experience does not indicate that a common-mode failure of safety-related equipment resulting from mild temperature environments is a problem. For these reasons, and because 10 CFR 50.49 does not require qualification to mild environments, no additional evaluations or documentation is necessary to ensure that this equipment will perform its safety function.

3.11.2 Qualification Tests and Methodology

Commonwealth Edison Company's approach to achieve EQ of pertinent electrical equipment is summarized as follows: [3.11-11]

A. Equipment located in mild-temperature and mild-radiation environments was not included within the scope of the NRC SER in accordance with 10 CFR 50.49. No action by CECo was required.

B. Qualification analysis or qualification testing (or a combination of both) was performed to ensure that equipment located in harsh-temperature and 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, qualification testing, or by replacement with a fully qualified component.

D. Equipment located in harsh-temperature and harsh-radiation environments was qualified by testing or by replacement with a qualified component.

IE Bulletin 79-01B and 10 CFR 50.49 require consideration of equipment aging due to material degradation occurring during normal plant life due to temperature and radiation effects. Electrical equipment having materials susceptible to significant age related degradation have been identified. A qualified (designated) life has been established for each equipment type with requisite replacement or component refurbishment schedules. Various methods were employed in establishing the qualified life for equipment such as: use of available qualification test data on similar or actual components or equipment to support a conservative equivalent life extrapolation of the enveloping temperature test profile using Arrhenius techniques; contact with vendors to obtain bills of material, material information, and technical data to identify age sensitive materials; review and engineering evaluation of industry references and technical literature to determine material radiation threshold and thermal withstand capabilities; and engineering analyses to establish a reasonable qualified life and justified replacement schedule. Calculations, assumptions, technical data and references were 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. [3.11-12]

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

Electrical equipment determined to be within the scope of 10 CFR 50.49 are identified in the station's work control system data base. The results of the environmental qualification determination for each of these items is in the extensive EQ file and EQ Binders, either created and maintained specifically for Quad Cities Station or for generic CECo nuclear plant applications. The 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. [3.11-13]

The existing maintenance and surveillance programs are used to specifically address the maintenance and surveillance requirements of environmental qualification (e.g., required maintenance resulting from use of components and parts with limited qualified life). These current programs are as follows: [3.11-14]

A. Like-for-like parts are used to maintain presently installed qualified components whether these components are qualified to the Division of Operating Reactors (DOR) Guidelines or to NUREG 0588, Category I or Category II. When identical parts are not available, an engineering analysis is 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 equipment qualified to NUREG 0588, Category I. Sound reasons to the contrary may preclude this upgrading practice when deemed necessary on a case-by-case basis. Guidance for determining sound reasons have been provided by the NRC in Generic Letter 82-09 and Reg Guide 1.89, Revision 1.

3.11.4 Loss of Ventilation

Where necessary, plant areas are served by appropriate HVAC systems to protect equipment from extreme environmental conditions and to maintain compartment temperatures below the 10 CFR 50.49 qualification temperature (as reported in Figure 3.11-1, Sheets 1 through 10; M-4A) of components required for safe shutdown of the plant. For further discussion of such systems, refer to the following sections:

- Control room, cable spreading room, auxiliary electric equipment room, and computer room — Section 9.4.1
- Battery room, computer room — Section 9.4.4
- Diesel generator rooms — Section 9.4.5
- HPCI room, Residual Heat Removal (RHR) corner rooms, Core spray rooms — Section 6.3.2
- RCIC room — Section 5.4.6

In determining the normal temperature parameters for environmental zones (reflected in Figure 3.11-1, Sheets 1 through 10; M-4A) the evaluation included the effects of normal plant operation, loss of HVAC, or operation of equipment required for post-accident plant recovery. Where comparatively high values for normal temperature appear, these result from conditions other than direct exposure to a LOCA or HELB.
[3.11-15]

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 Quad Cities. Demineralized water containment spray is used and considered in the evaluation. [3.11-16]

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 environment was based on the GE document "Radiation Source Information for NUREG 0578 Implementation Computer Run"[3][4][5].

The introduction of SPC fuel (ATRIUM-9B) did not invalidate the results of the study 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 2957 MWt and the use of fuel types GE14, Westinghouse Optima2, and AREVA (now Framatome) ATRIUM 10XM with a 24-month fuel cycle. [6][7] The radiation environments for normal service and post-accident conditions depicted in Figure 3.11-1 reflect the uprated core. The methodology discussed below to develop the environment 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. Inspections have been performed in the Unit 2 drywell, confirming that all equipment and cable is enclosed in inorganic materials. Beta doses have therefore not been considered.

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 need be considered. Each safety-related component was located with respect to the piping systems containing post-LOCA radioactive fluids. The integrated dose was established 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 calculate a total dose.

3.11.6 References

1. Response to IEB 79-01B for Quad Cities Nuclear Power Station Units 1 and 2, Commonwealth Edison Company, Revision 6, June 1986.

2. "Analysis of Effects of Pipe Break Outside Primary Containment," Quad Cities Station Units 1 and 2, Special Report 12, Revision 1, February 1975.

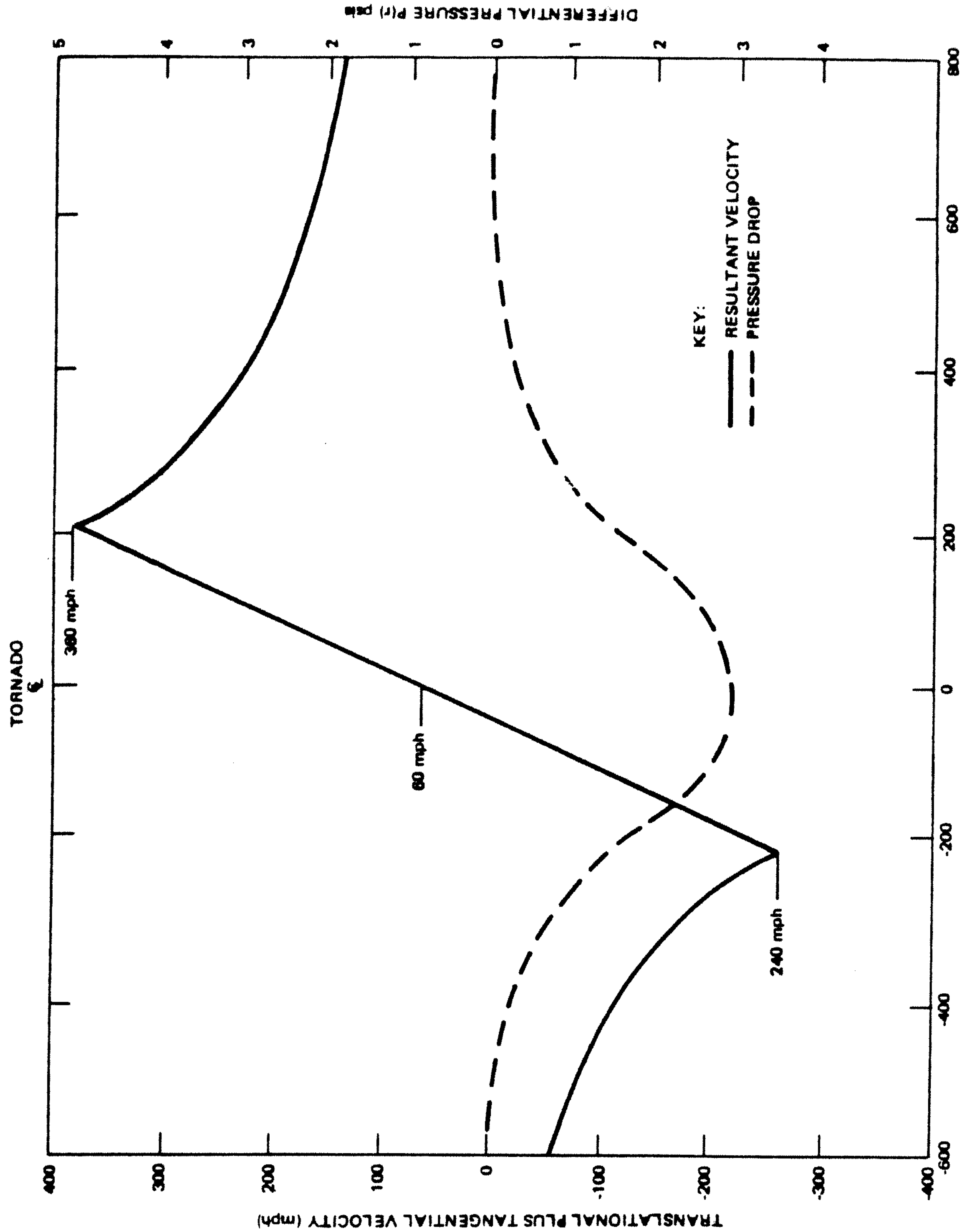
3. General Electric Document, Radiation Source Information for NUREG 0578 Implementation Computer Run, SNUMB 7007S, November 1979.

4. "Response to IE Bulletin 79-01B, Post-LOCA/HELB Radiation Exposure Levels Received by ESF System Components for Quad Cities Nuclear Power Station, Units 1 and 2, Commonwealth Edison Company, Docket Numbers 50-254 and 50-265" (Bechtel Radiation Study), Specification 13524-069-N201, Rev. 1, 4/10/85.

5. "BWR Owners' Group NUREG-0578 Implementation: Analyses and Positions for Plant-Unique Submittals," NEDO-24782, August 1980.

6. General Electric Document GE-NE-A22-00103-64-01, “Dresden and Quad Cities Asset Enhancement Program, Task TO 802: Radiation Sources and Fission Products,” August 2000.

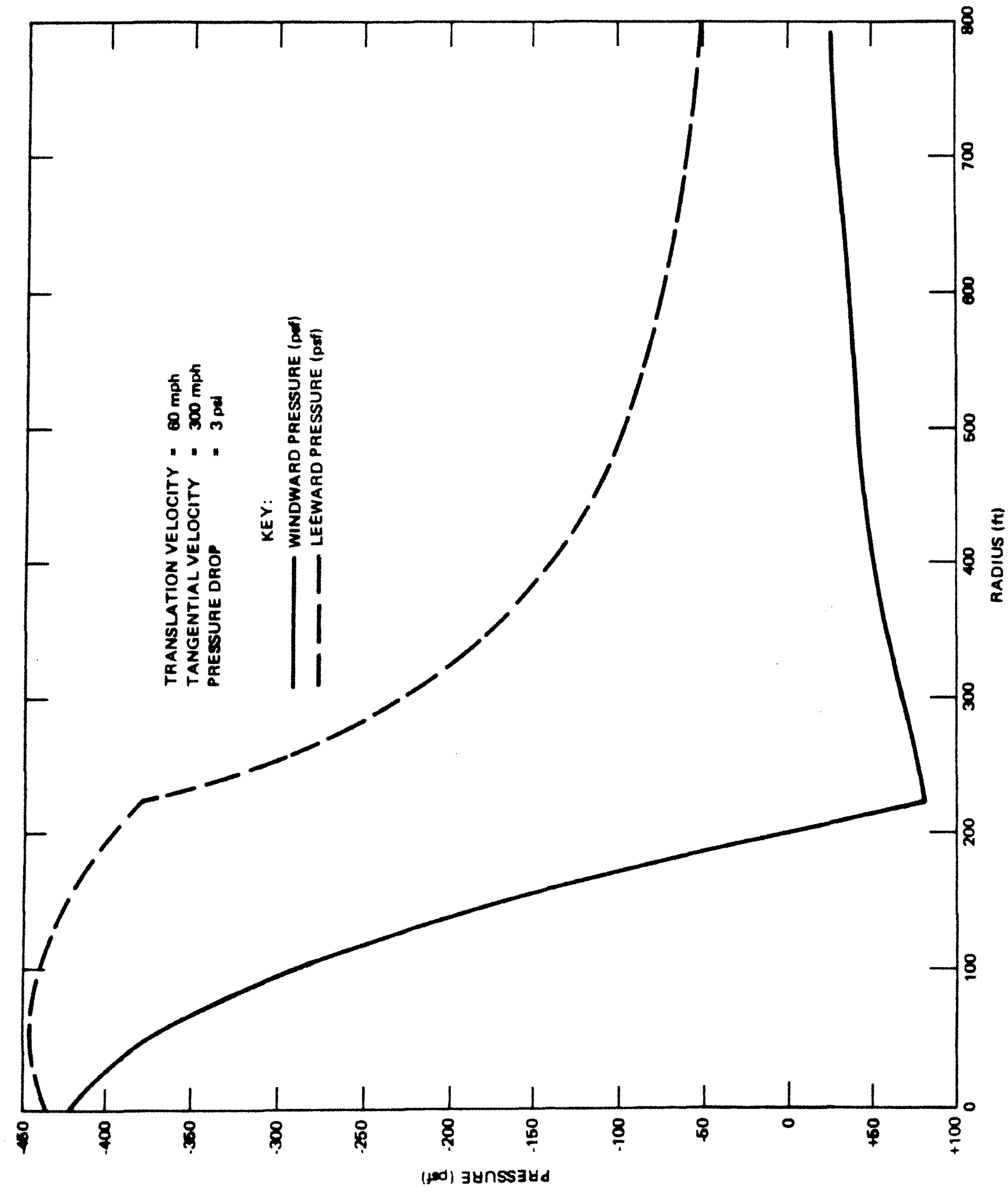
7. QDC-0000-N-1022, Rev. 1, “Dose and Dose Rate Scaling Factors to Evaluate Impact of EPU on Radiological Equipment Qualification and Vital Access”



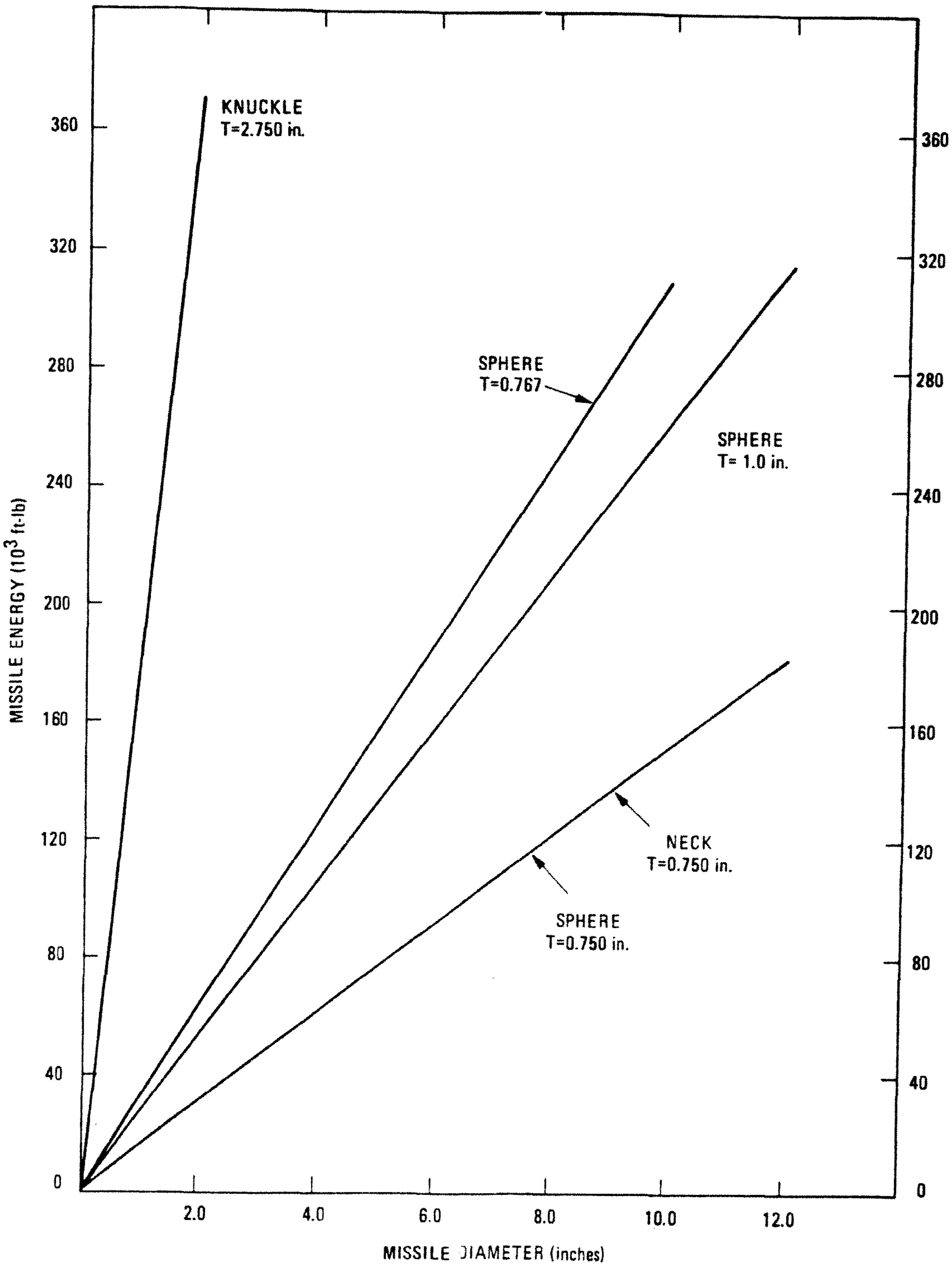
QUAD CITIES STATION
UNITS 1 & 2

VARIATION OF DIFFERENTIAL PRESSURE AND TANGENTIAL PLUS TRANSLATIONAL VELOCITY AS A FUNCTION OF DISTANCE FROM CENTER OF TORNADO

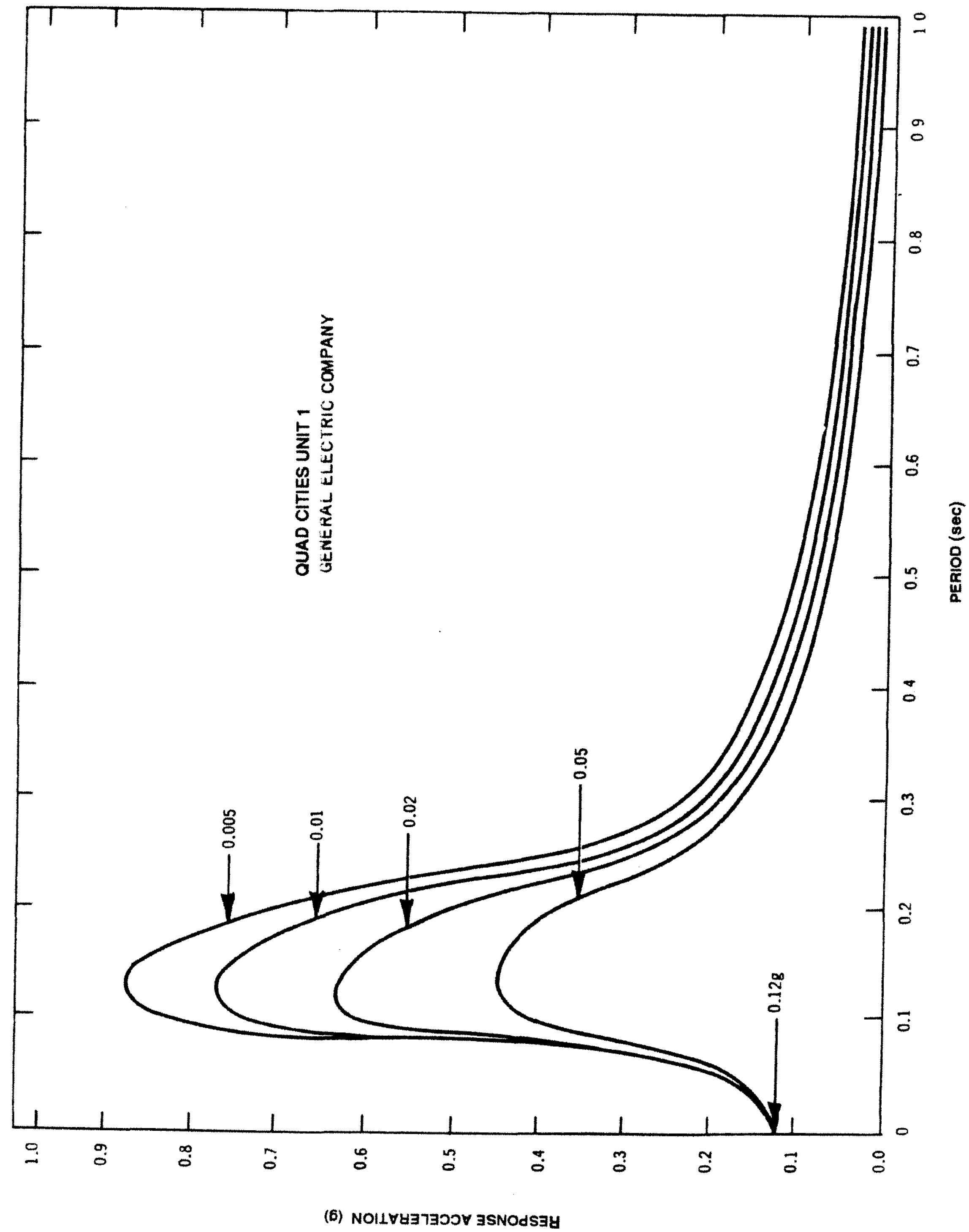
FIGURE 3.3-1



QUAD CITIES STATION UNITS 1 & 2
RESULTANT SURFACE PRESSURES (WINDWARD AND LEEWARD) FOR DESIGN BASIS TORNADO
FIGURE 3.3-2



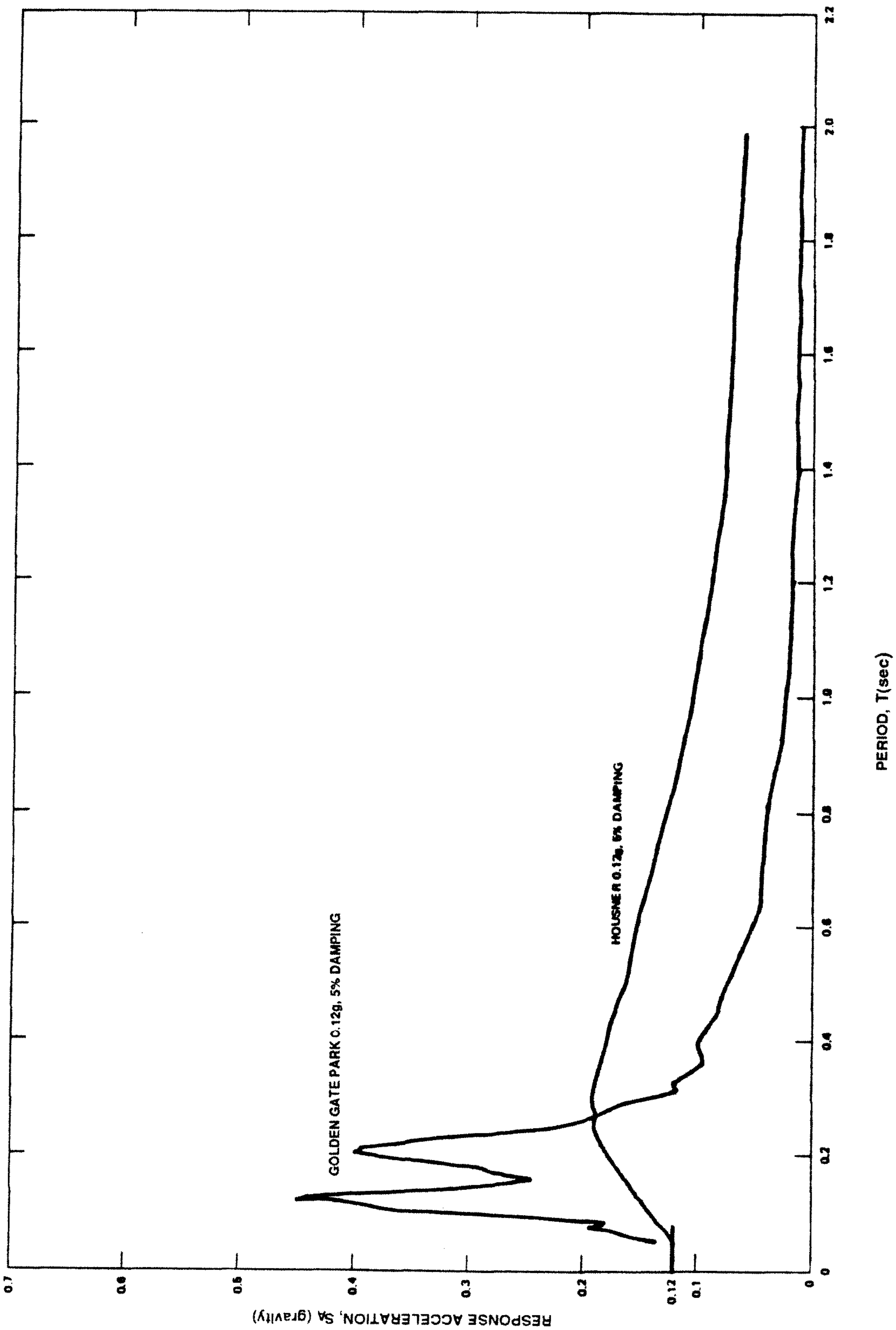
QUAD CITIES STATION UNITS 1 & 2
ENERGY NECESSARY TO PENETRATE DRYWELL CONTAINMENT
FIGURE 3.5-1



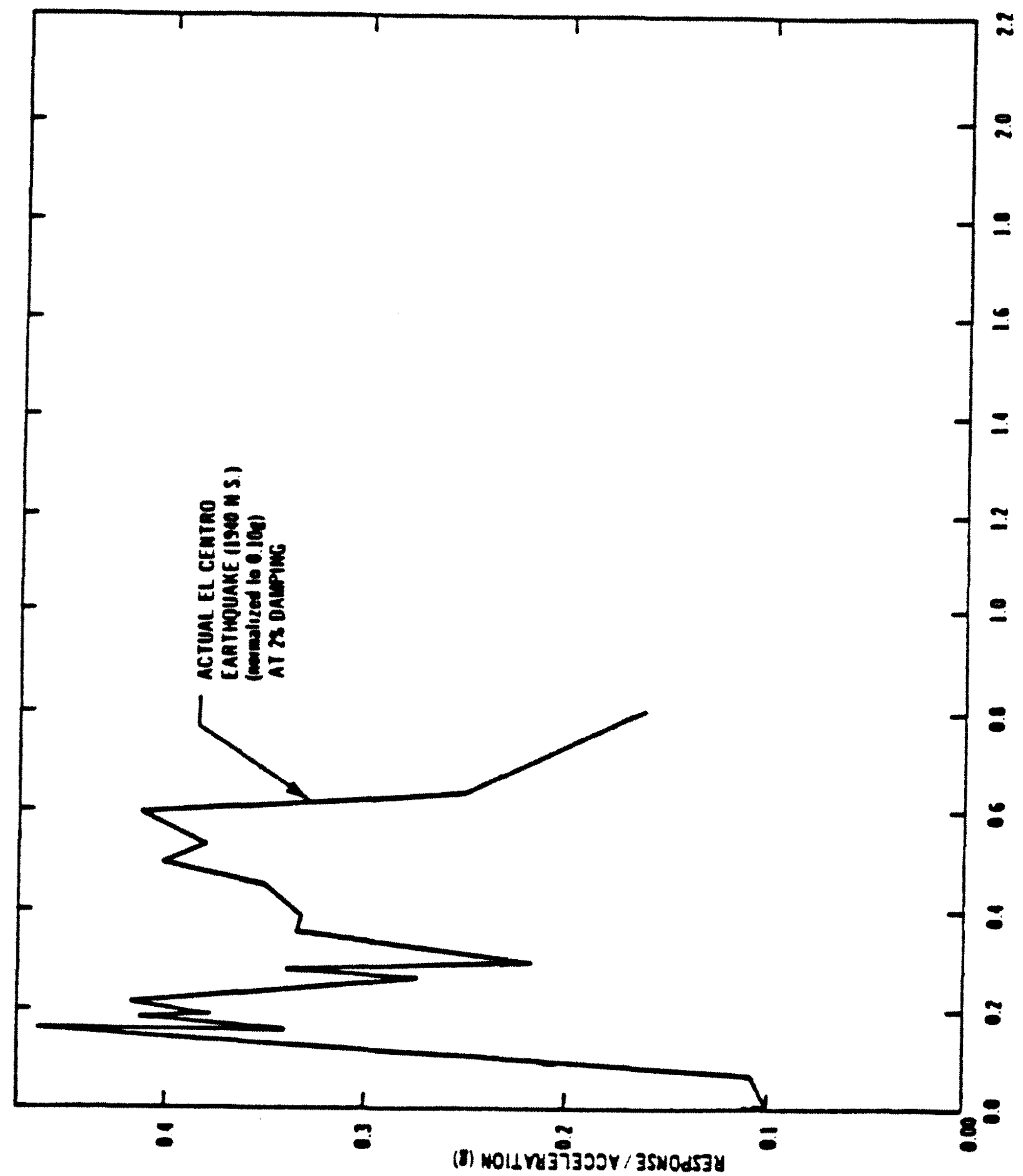
QUAD CITIES STATION
UNITS 1 & 2

RESPONSE ACCELERATION SPECTRA

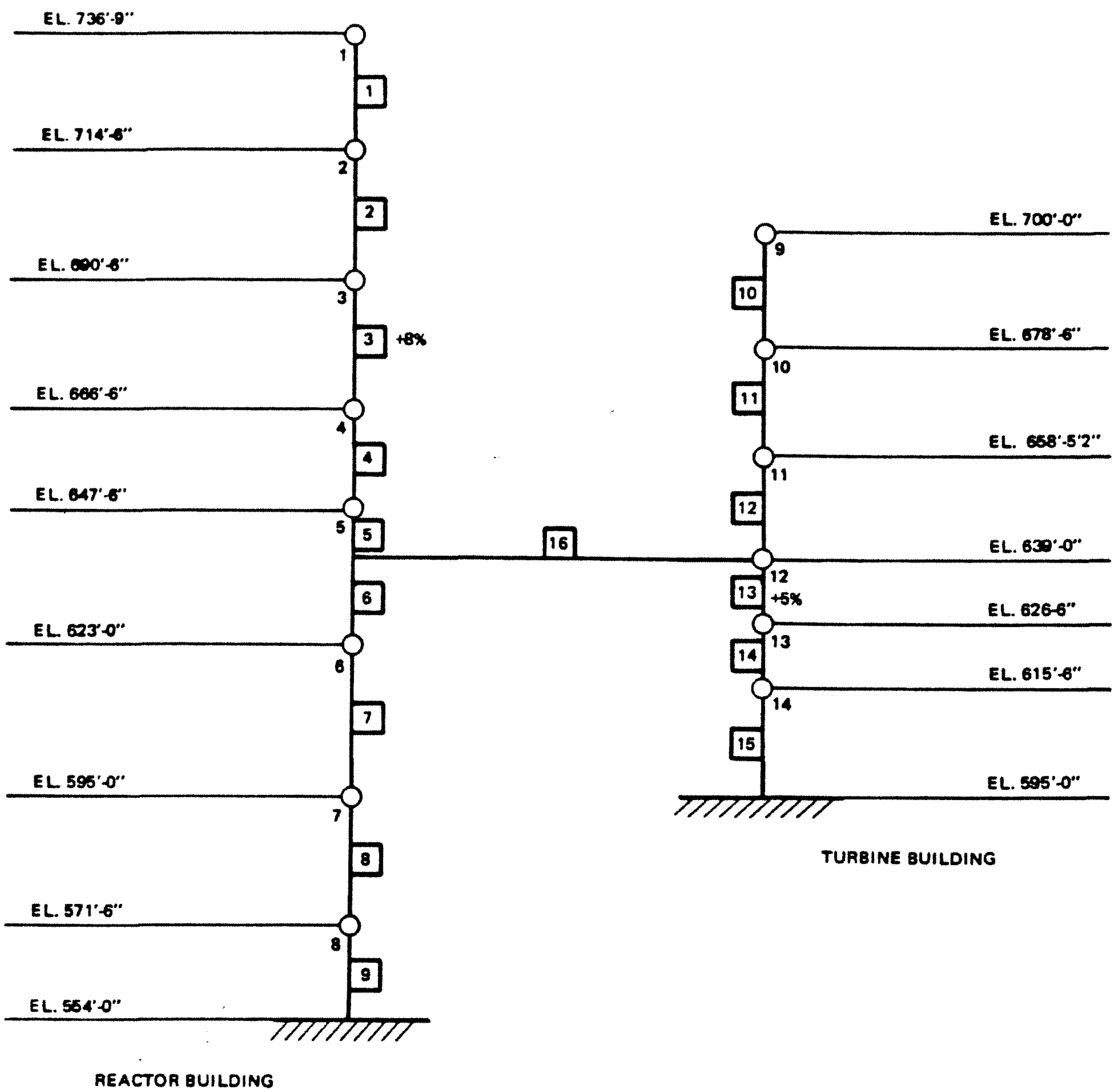
FIGURE 3.7-1



QUAD CITIES STATION UNITS 1 & 2
COMPARISON OF GOLDEN GATE PARK AND HOUSNER GROUND RESPONSE SPECTRA
FIGURE 3.7-2



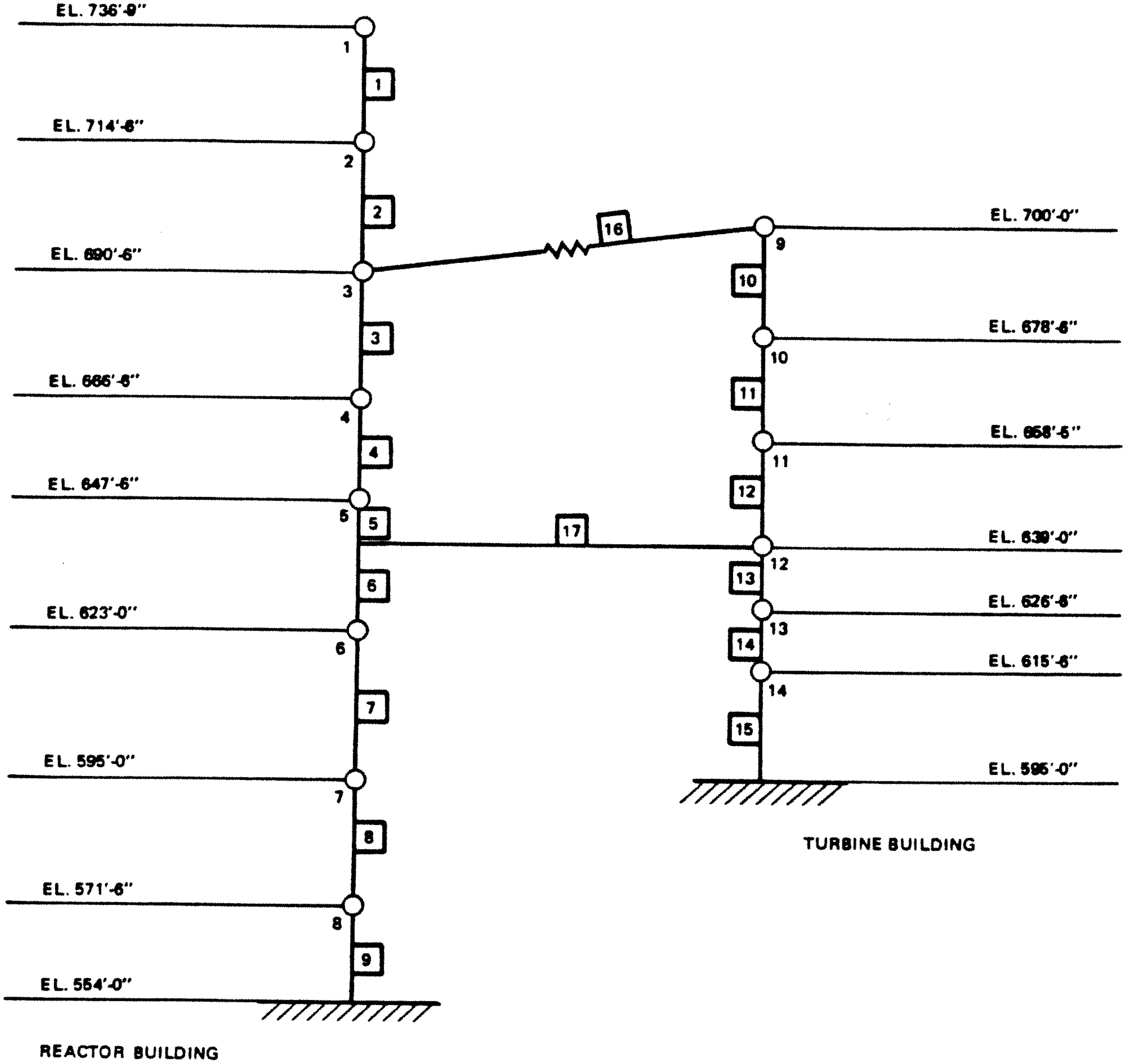
QUAD CITIES STATION UNITS 1 & 2
RESPONSE SPECTRUM CURVE FOR EL CENTRO EARTHQUAKE
FIGURE 3.7-3



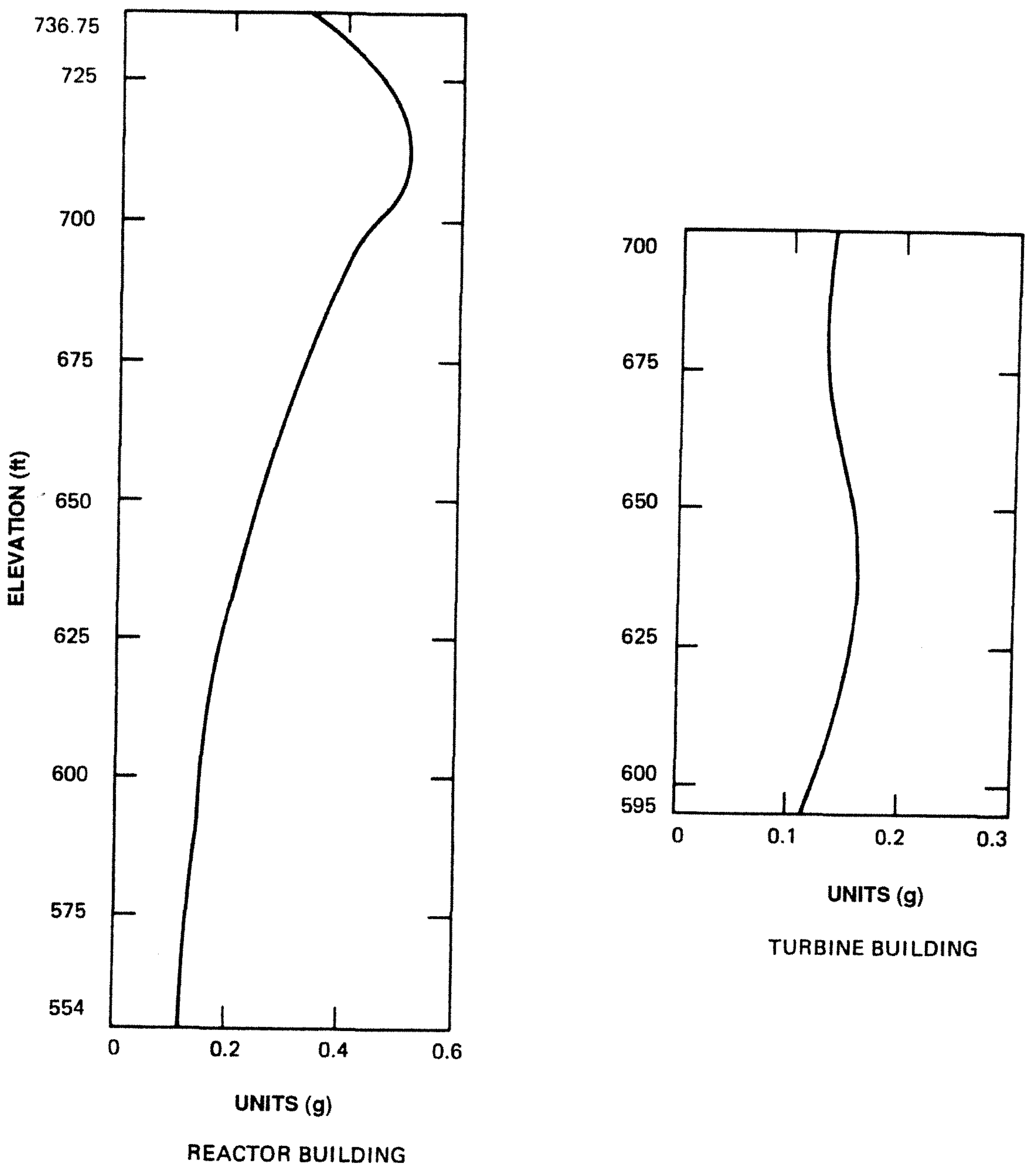
QUAD CITIES STATION
UNITS 1 & 2

REACTOR TURBINE BUILDING MATHEMATICAL MODEL EARTHQUAKE IN N-S DIRECTION

FIGURE 3.7-4



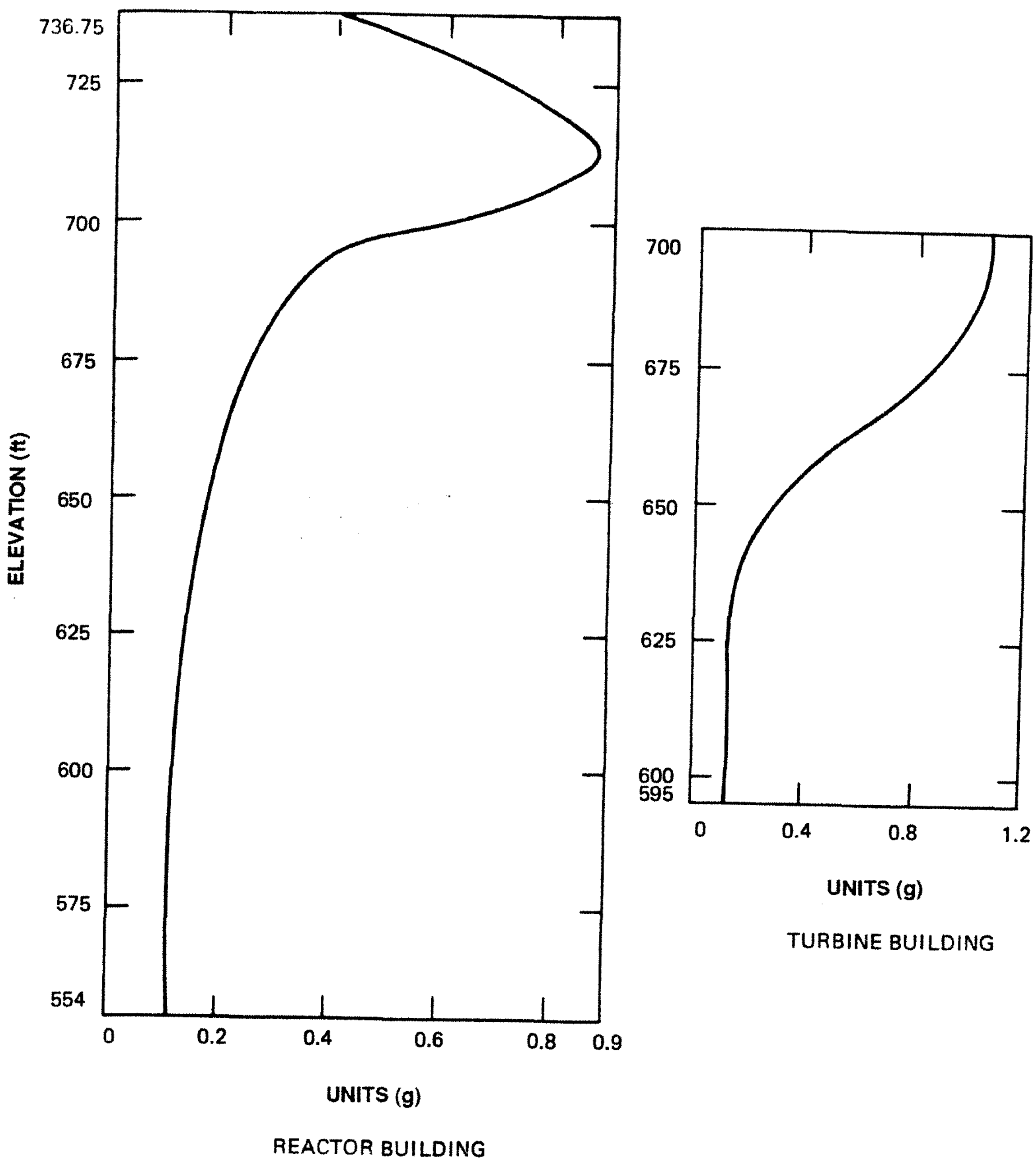
QUAD CITIES STATION UNITS 1 & 2
REACTOR TURBINE BUILDING MATHEMATICAL MODEL EARTHQUAKE IN E-W DIRECTION
FIGURE 3.7-5



QUAD CITIES STATION
UNITS 1 & 2

SEISMIC ACCELERATION DIAGRAM, REACTOR AND TURBINE BUILDINGS - EARTHQUAKE IN N-S DIRECTION

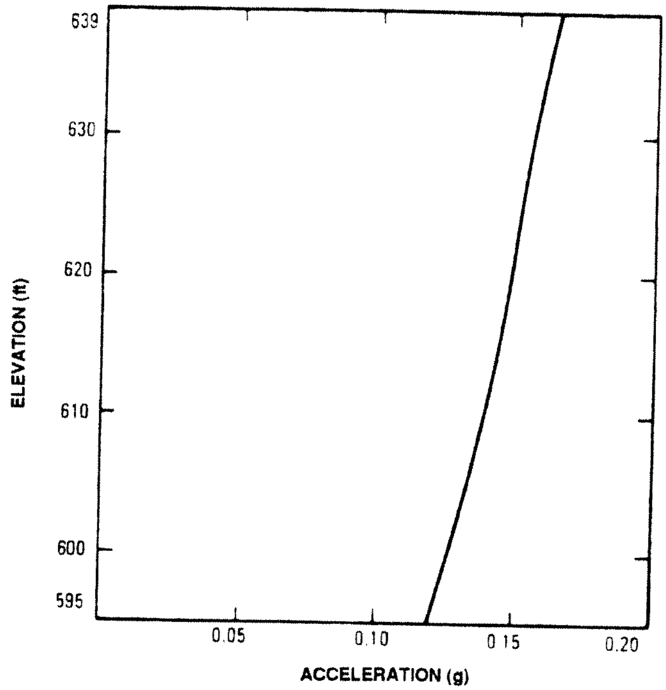
FIGURE 3.7-6



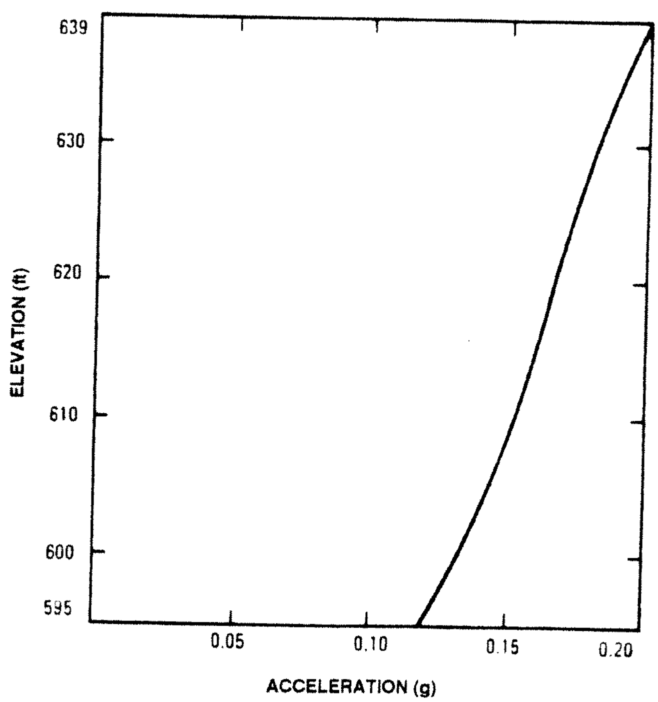
QUAD CITIES STATION
UNITS 1 & 2

SEISMIC ACCELERATION DIAGRAM, REACTOR AND TURBINE BUILDINGS - EARTHQUAKE IN E-W DIRECTION

FIGURE 3.7-7

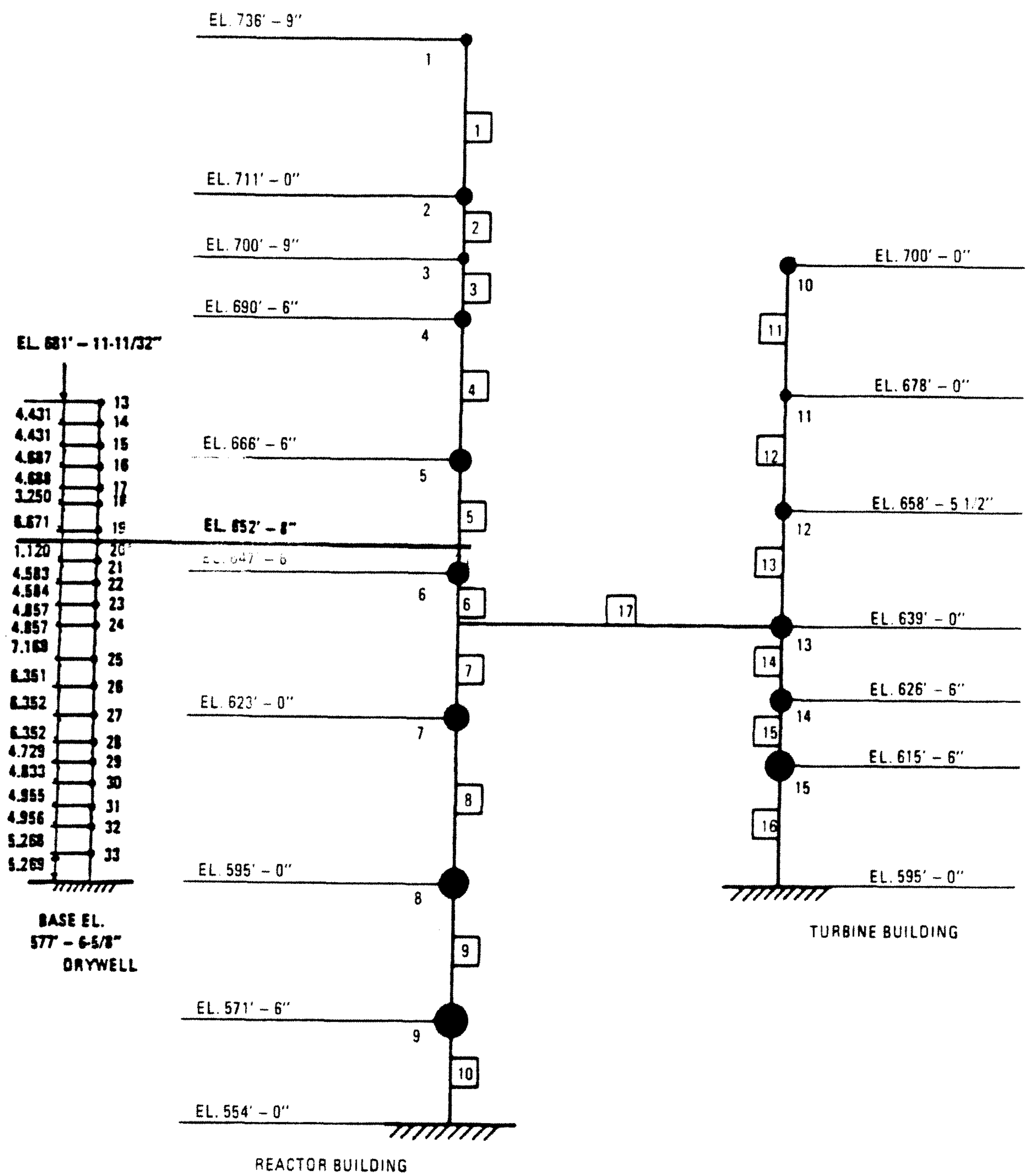


N-S DIRECTION



E–W DIRECTION

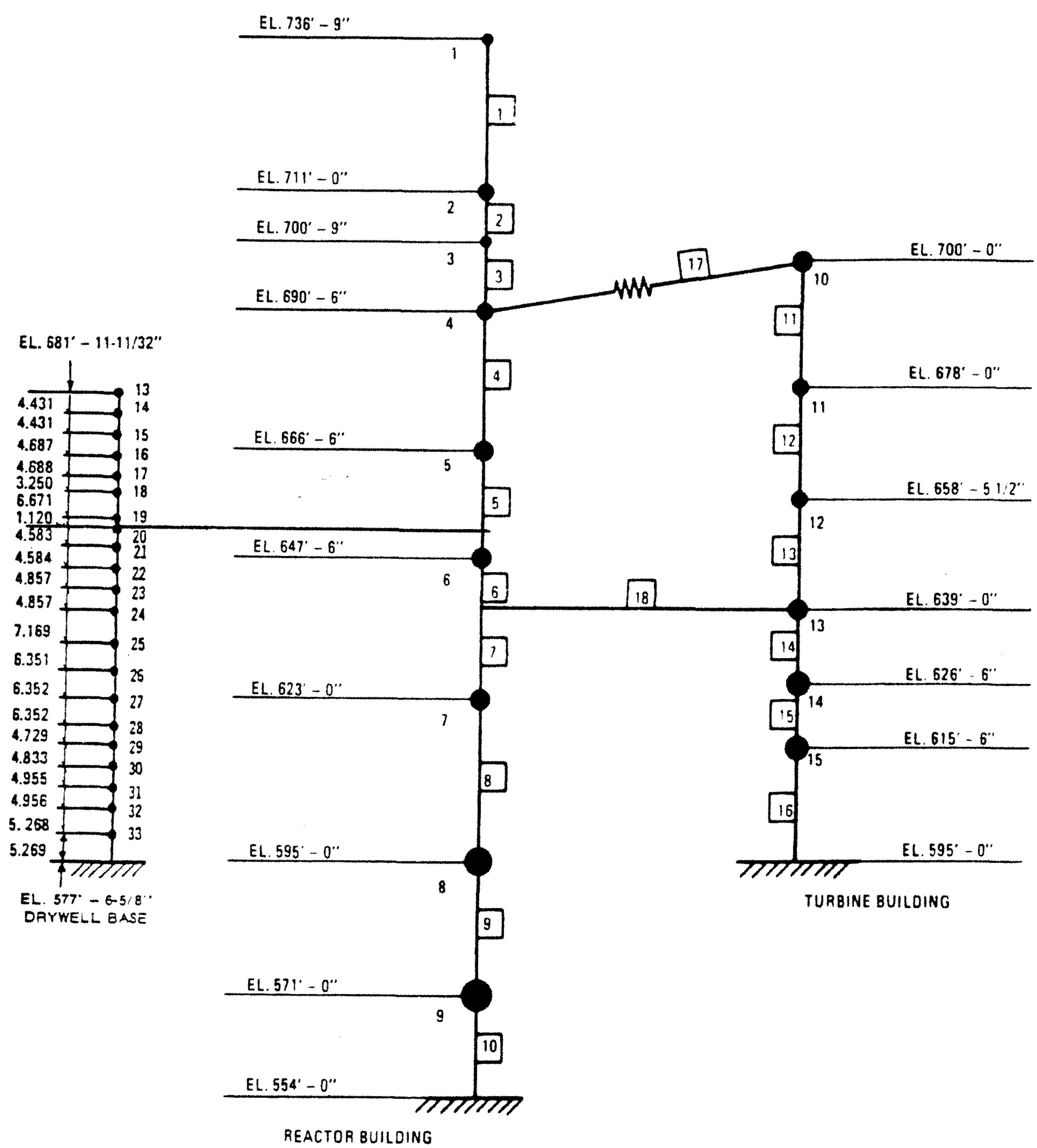
QUAD CITIES STATION UNITS 1 & 2
SEISMIC ACCELERATION DIAGRAMS CONTROL ROOM
FIGURE 3.7-8



QUAD CITIES STATION
UNITS 1 & 2

MATHEMATICAL MODEL FOR DRYWELL
ANALYSIS IN N-S DIRECTION

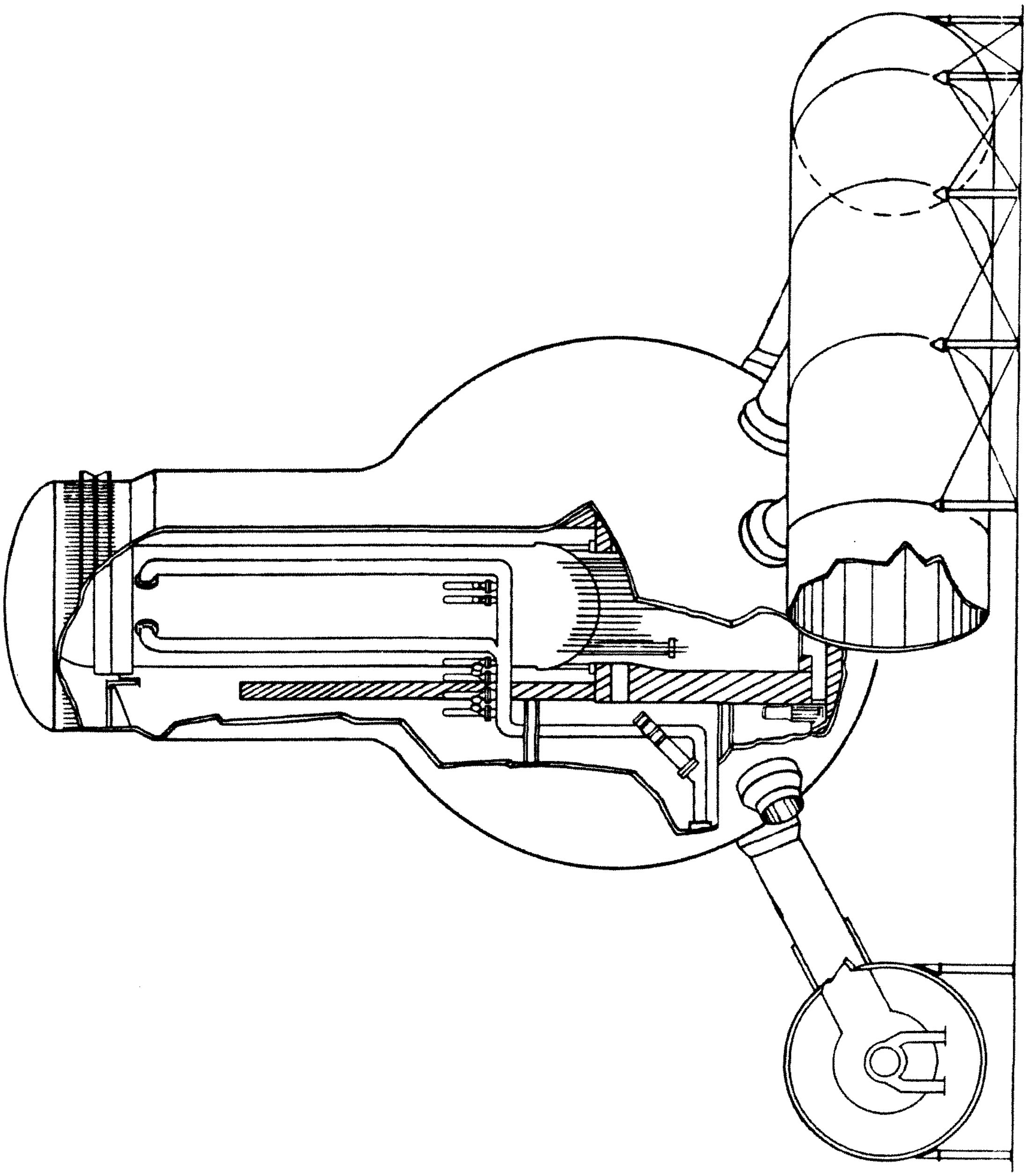
FIGURE 3.7-9



QUAD CITIES STATION
UNITS 1 & 2

MATHEMATICAL MODEL FOR DRYWELL ANALYSIS IN E-W DIRECTION

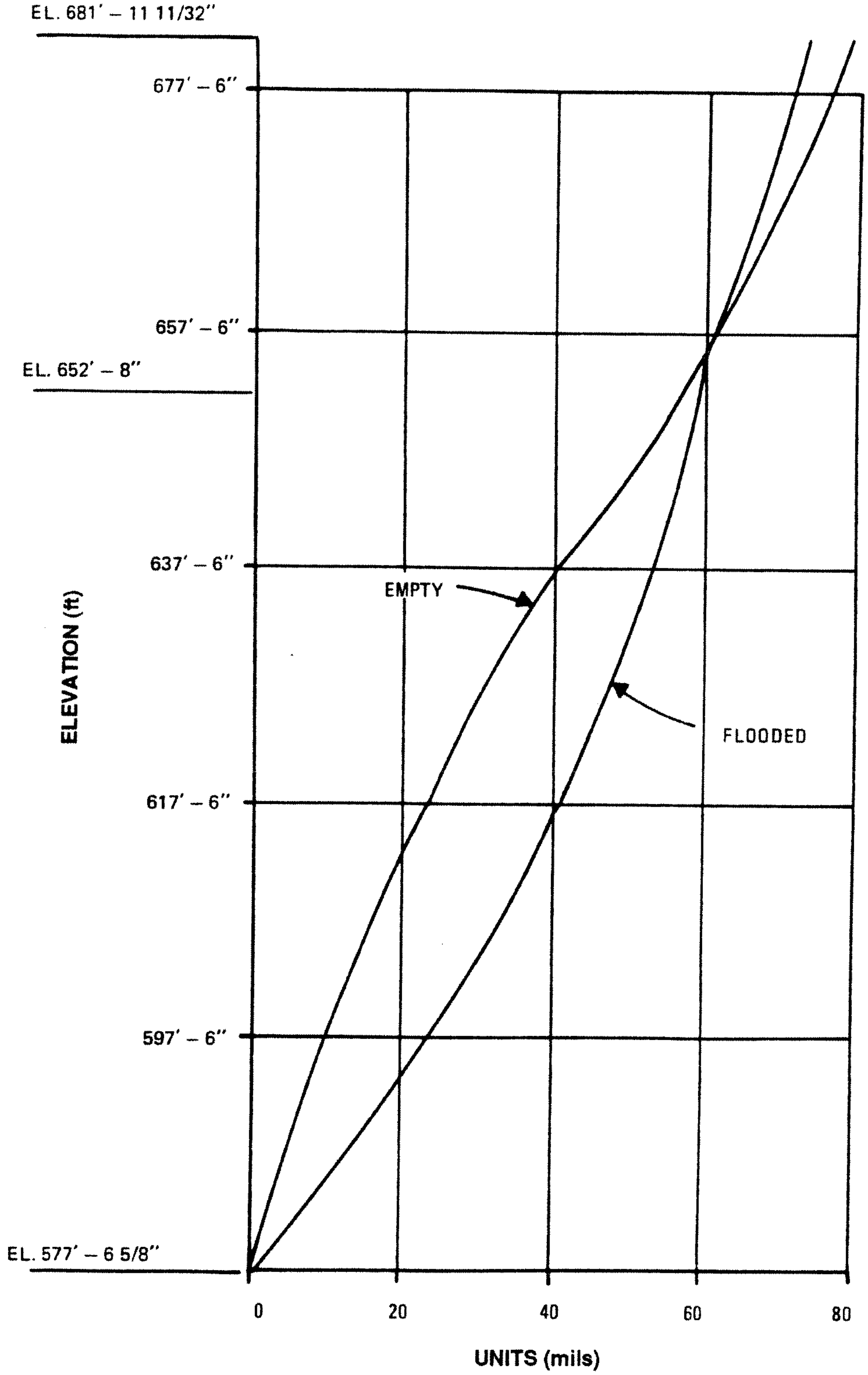
FIGURE 3.7-10



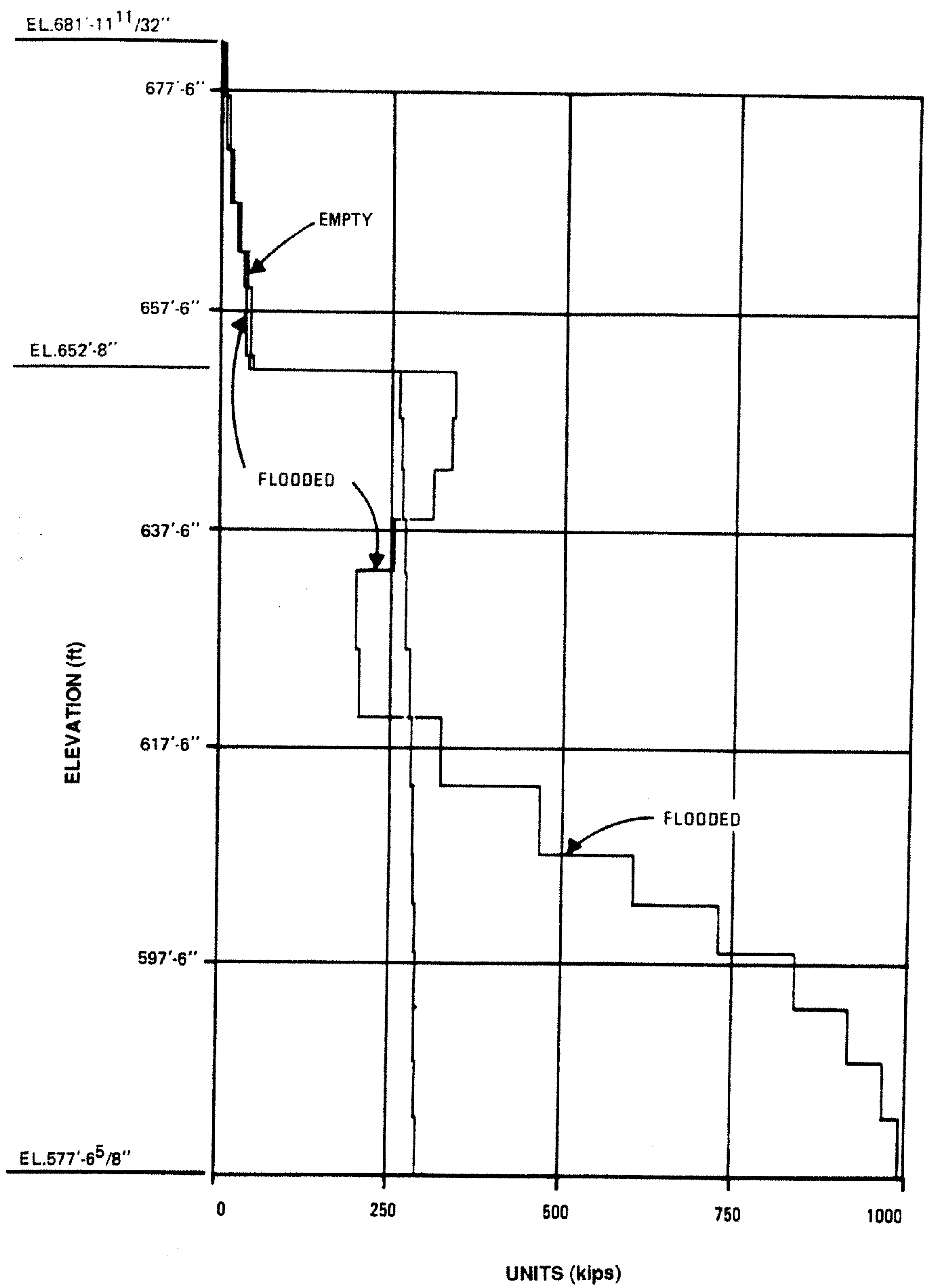
QUAD CITIES STATION
UNITS 1 & 2

PRESSURE SUPPRESSION CONTAINMENT SYSTEM

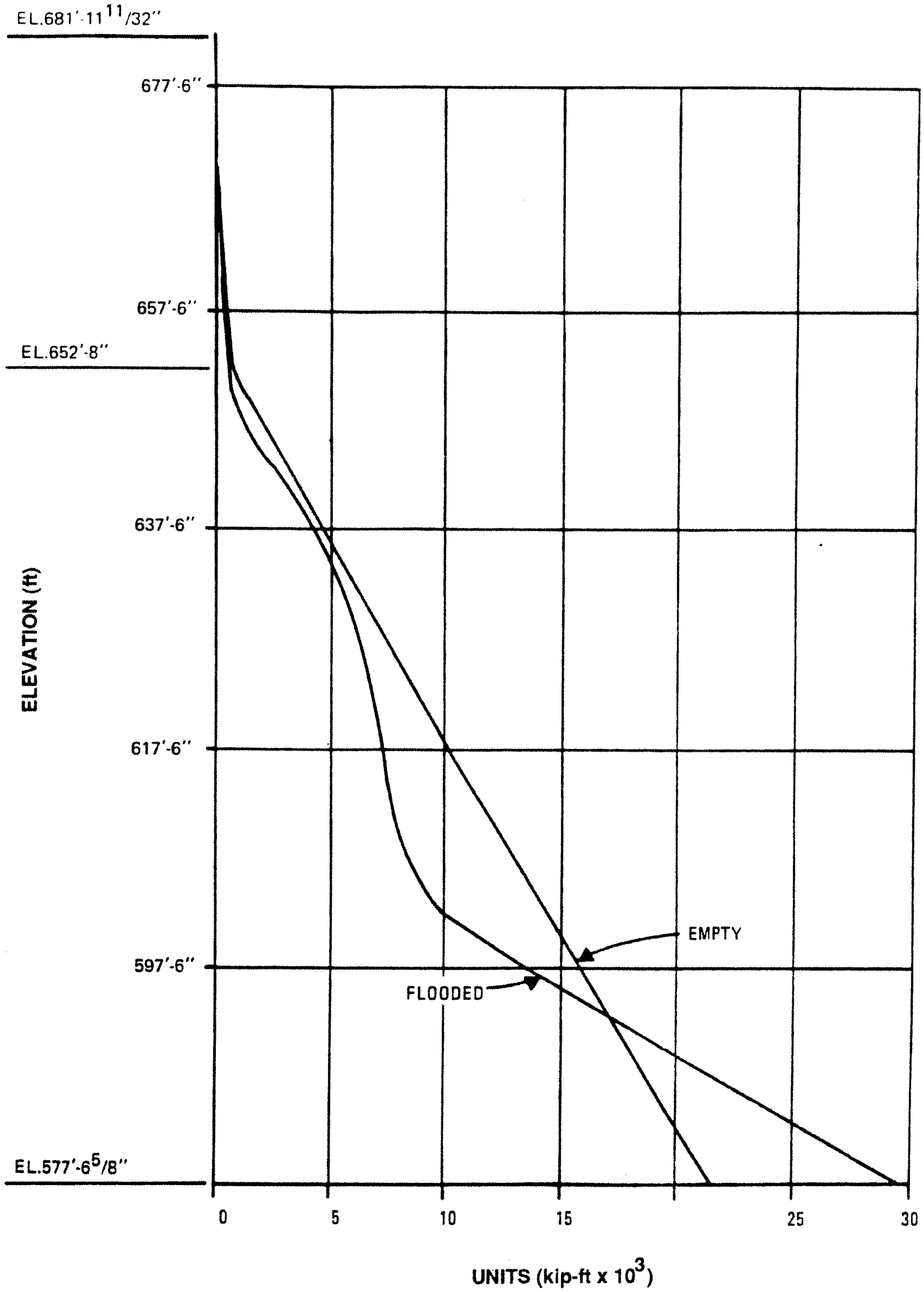
FIGURE 3.8-1



QUAD CITIES STATION UNITS 1 & 2
DRYWELL OBE DISPLACEMENT DIAGRAM (N-S DIRECTION)
FIGURE 3.8-2



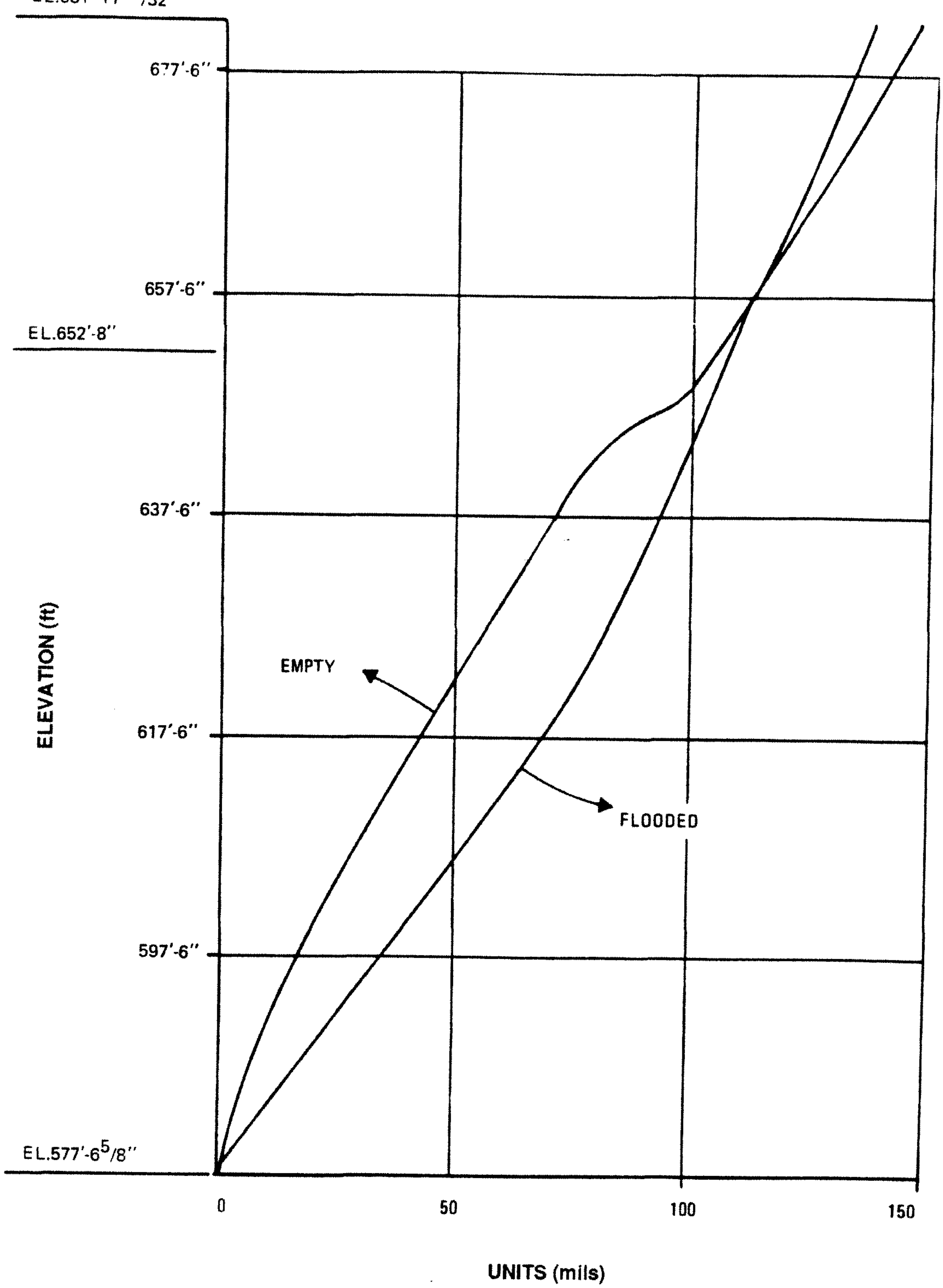
QUAD CITIES STATION UNITS 1 & 2
DRYWELL OBE SHEAR DIAGRAM (N-S DIRECTION)
FIGURE 3.8-3



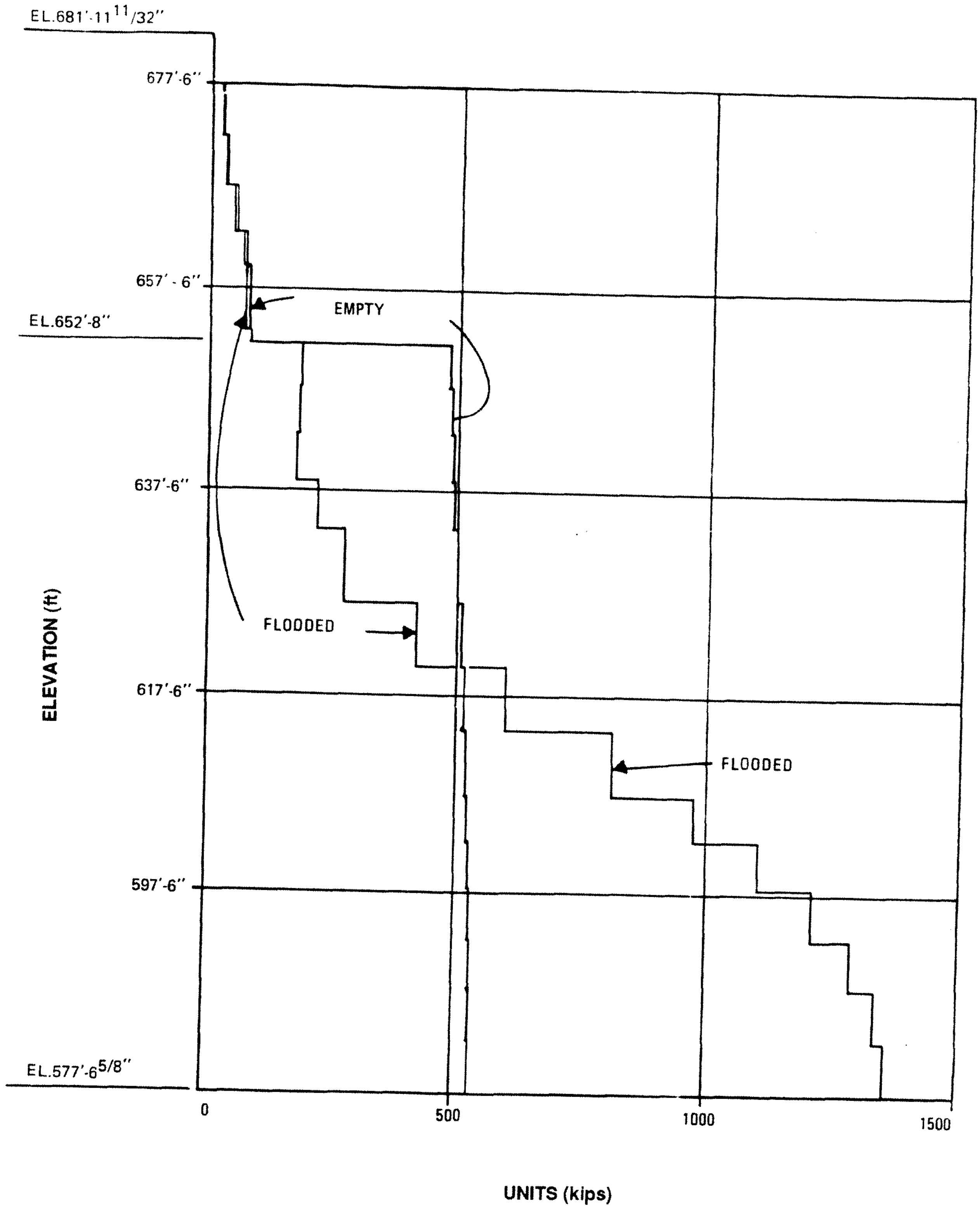
QUAD CITIES STATION
UNITS 1 & 2

DRYWELL OBE MOMENT DIAGRAM
(N-S DIRECTION)

FIGURE 3.8-4



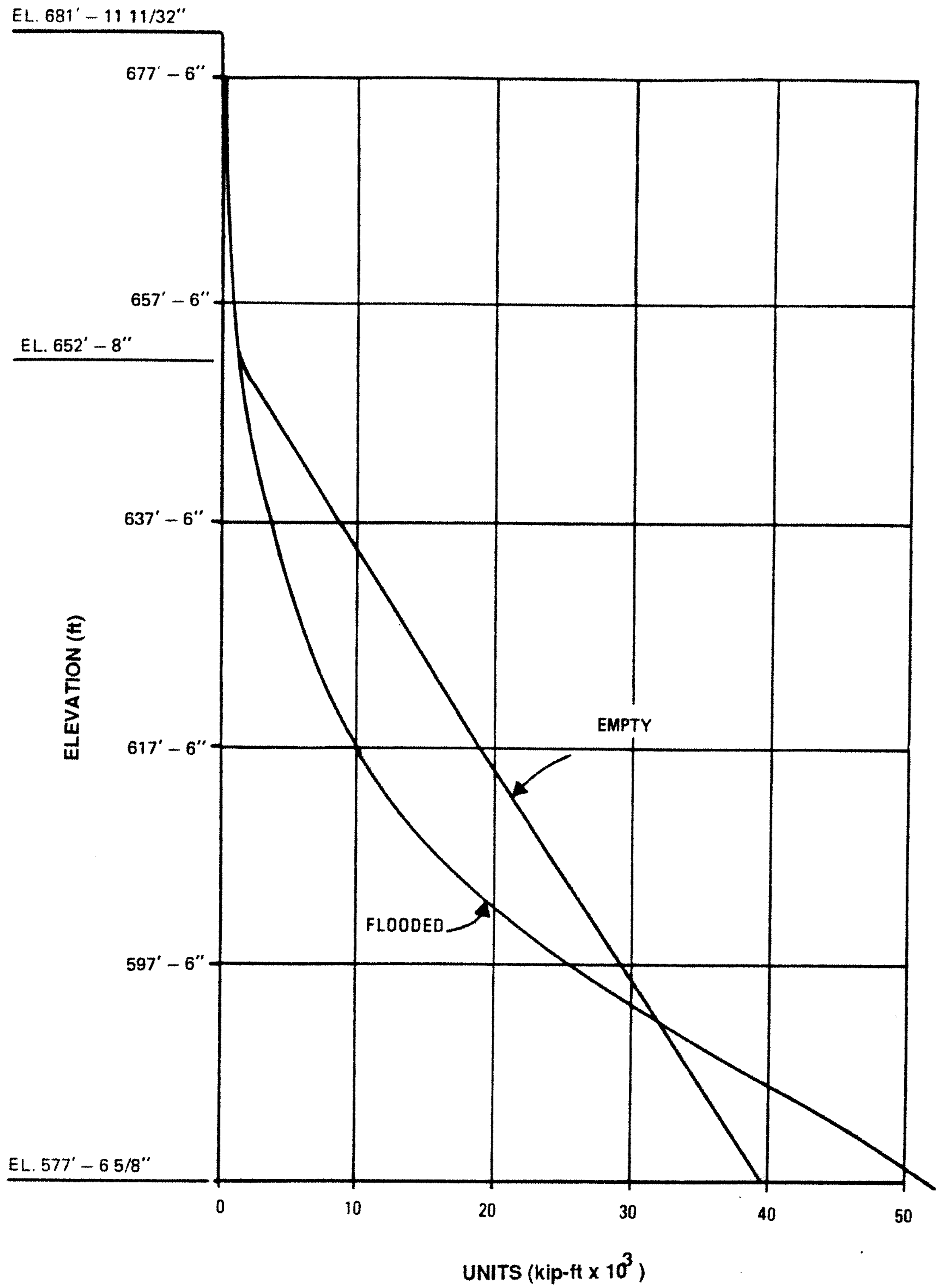
QUAD CITIES STATION UNITS 1 & 2
DRYWELL OBE DISPLACEMENT DIAGRAM (E-W DIRECTION)
FIGURE 3.8-5



QUAD CITIES STATION
UNITS 1 & 2

DRYWELL OBE SHEAR DIAGRAM
(E-W DIRECTION)

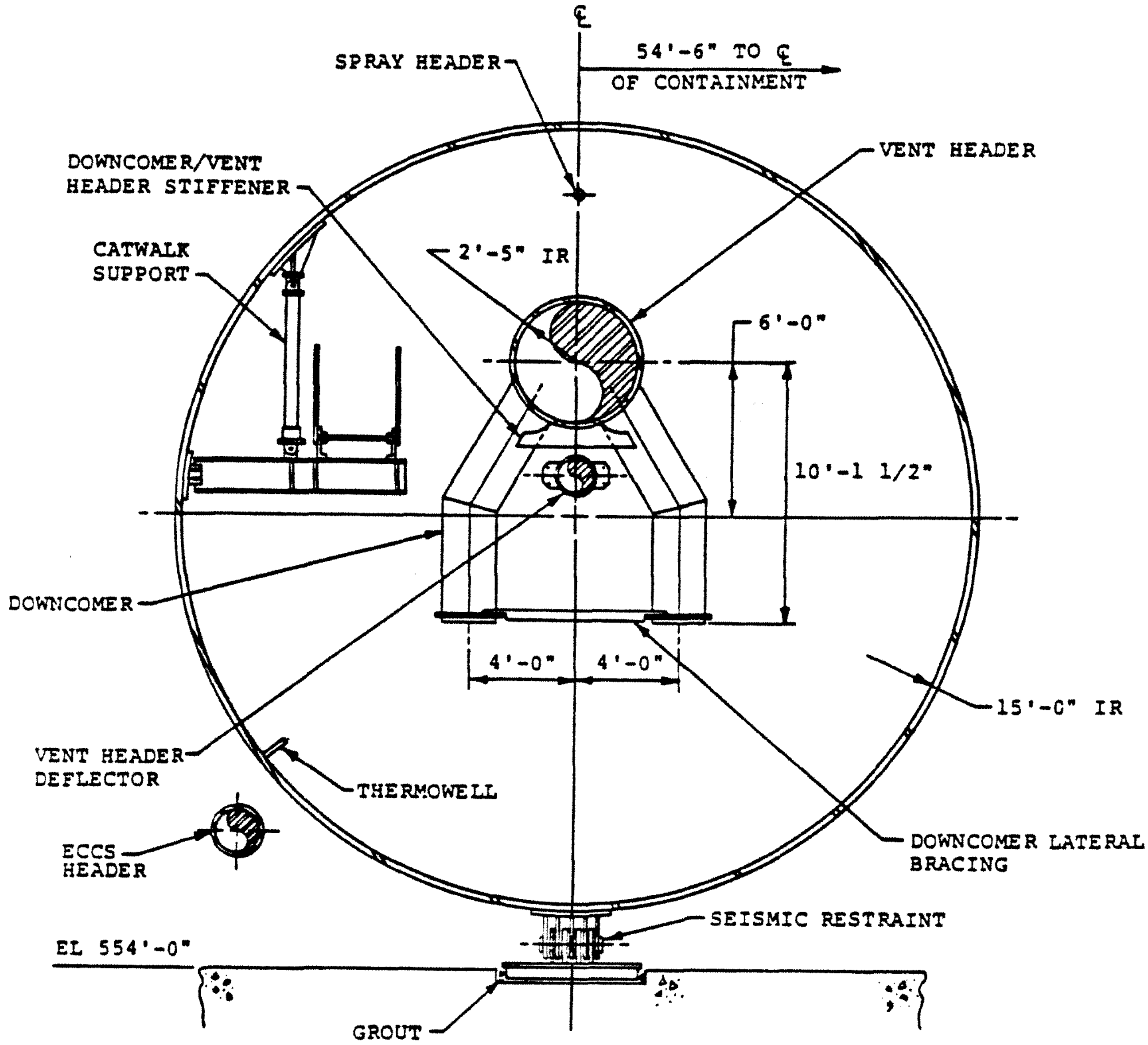
FIGURE 3.8-6



QUAD CITIES STATION
UNITS 1 & 2

DRYWELL OBE MOMENT DIAGRAM
(E-W DIRECTION)

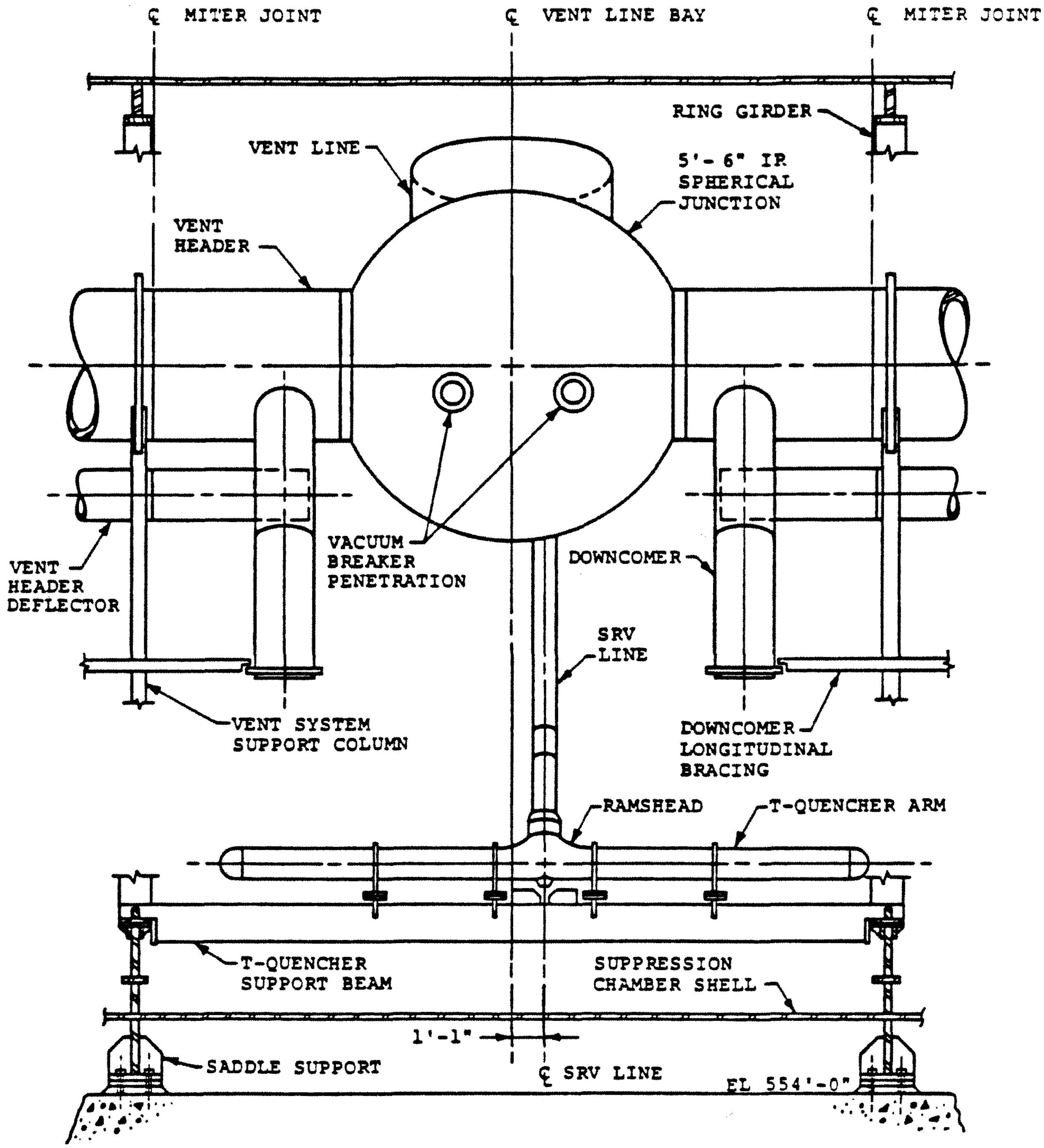
FIGURE 3.8-7



QUAD CITIES STATION
UNITS 1 & 2

SUPPRESSION CHAMBER SECTION
MIDBAY NON-VENT LINE BAY

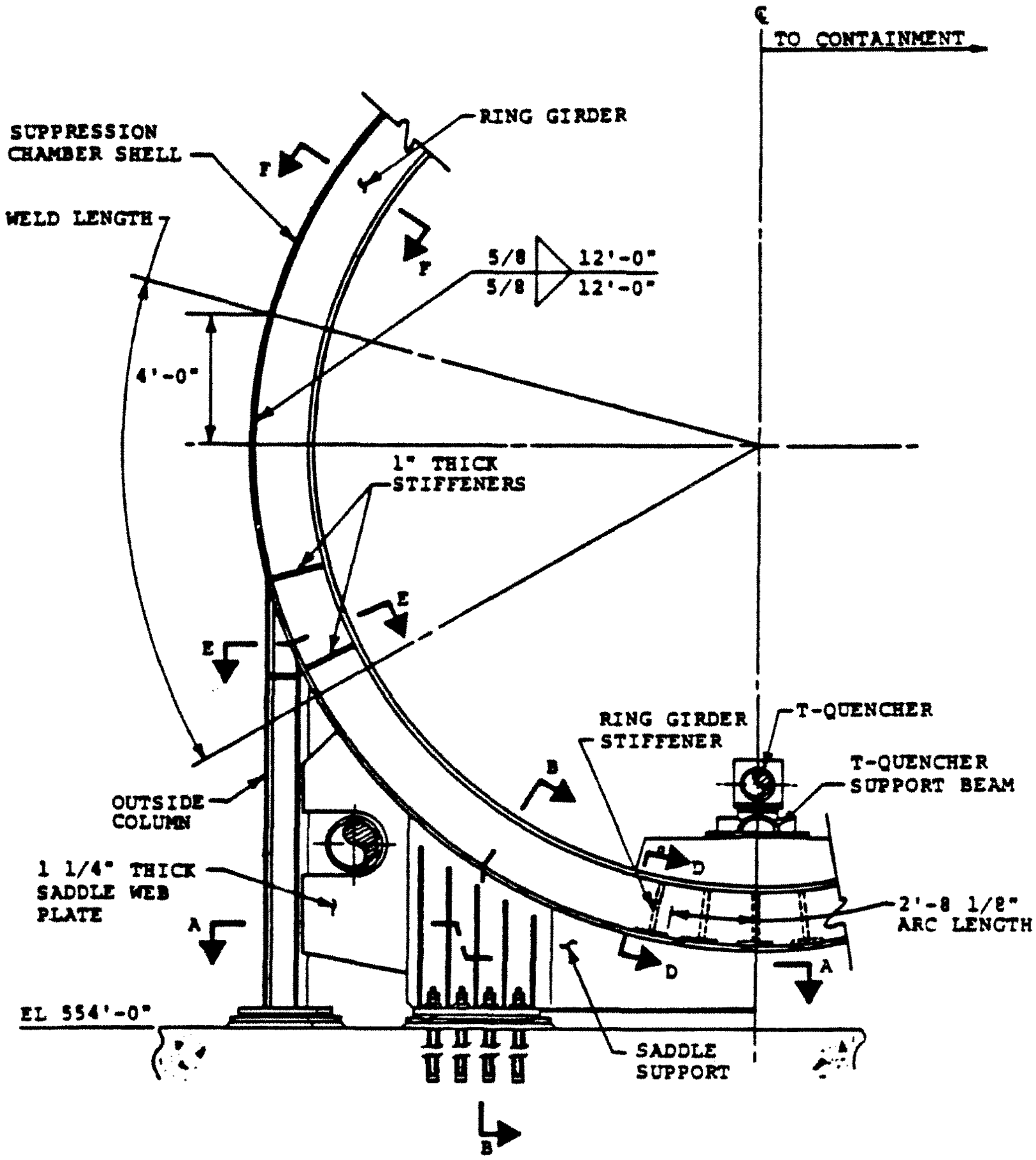
FIGURE 3.8-8



QUAD CITIES STATION
UNITS 1 & 2

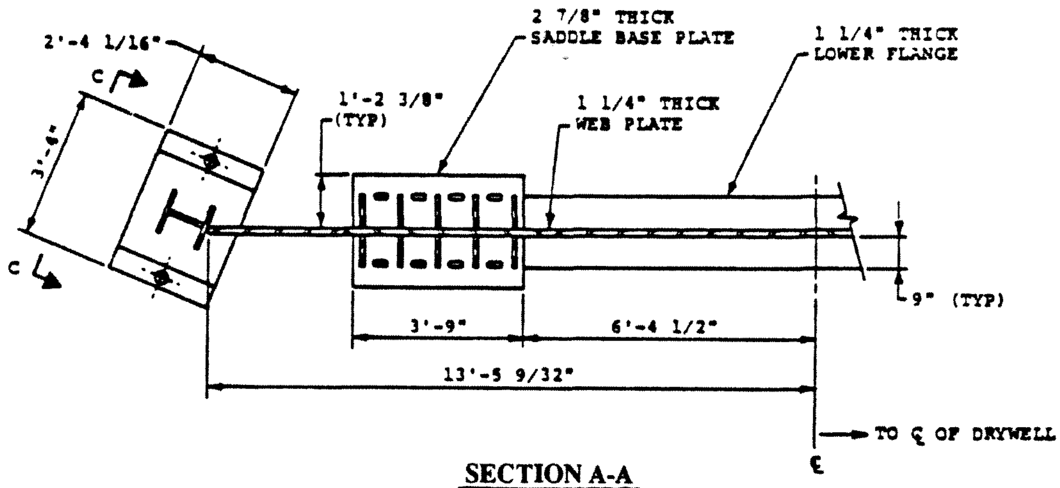
DEVELOPED VIEW OF SUPPRESSION CHAMBER SEGMENT

FIGURE 3.8-9

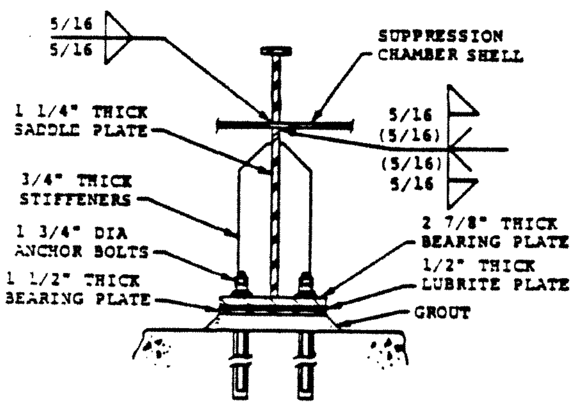


NOTE:
SEE FIGURES 3.8-11 & 3.8-12 FOR SECTIONS AND VIEWS

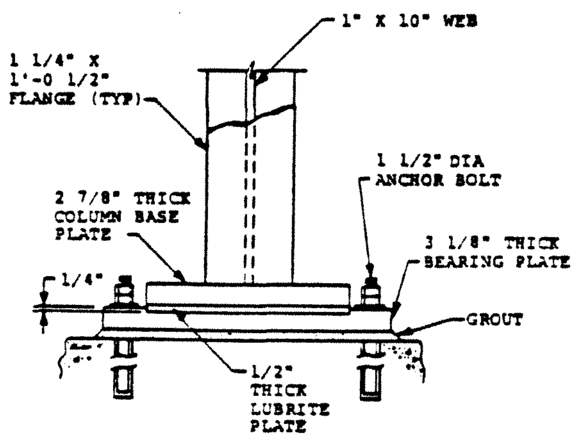
QUAD CITIES STATION UNITS 1 & 2
SUPPRESSION CHAMBER RING GIRDER AND VERTICAL SUPPORTS - PARTIAL ELEVATION VIEW
FIGURE 3.8-10



SECTION A-A
FIGURE 3.8-10



SECTION B-B
FIGURE 3.8-10

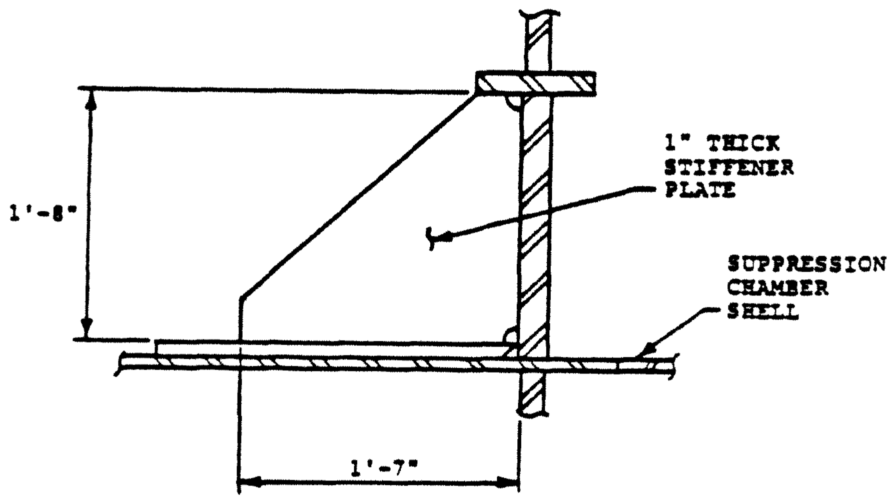


VIEW C-C

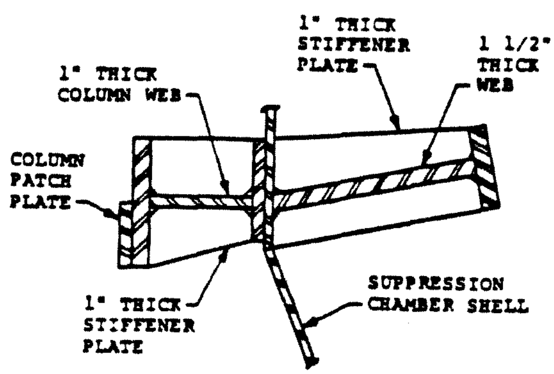
QUAD CITIES STATION
UNITS 1 & 2

SUPPRESSION CHAMBER VERTICAL SUPPORT
BASE PLATES - PARTIAL PLAN VIEW AND DETAILS

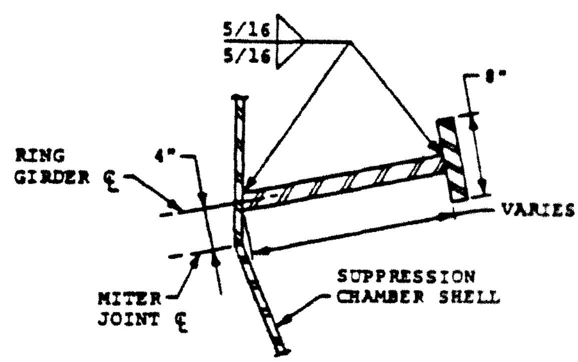
FIGURE 3.8-11



SECTION D-D
FIGURE 3.8-10

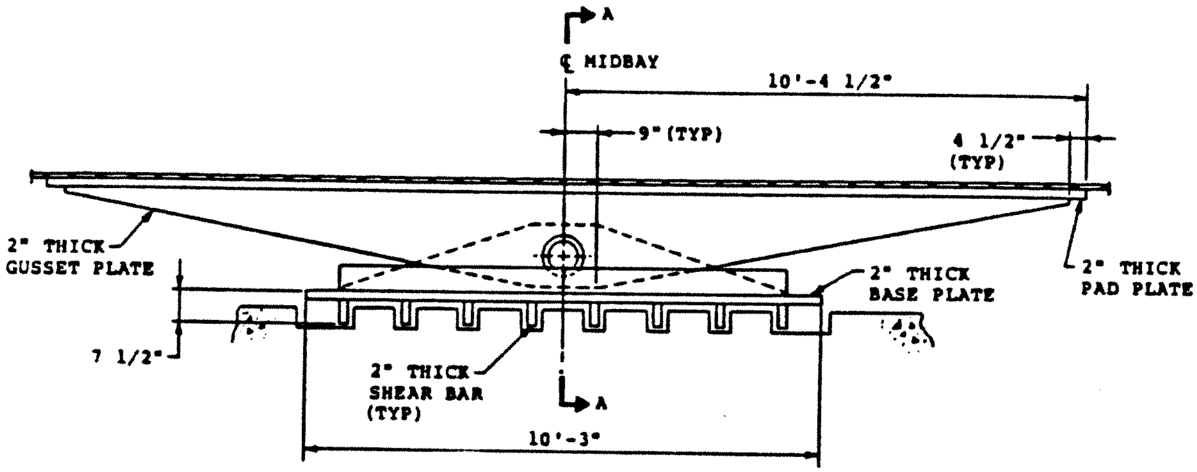


SECTION E-E
FIGURE 3.8-10

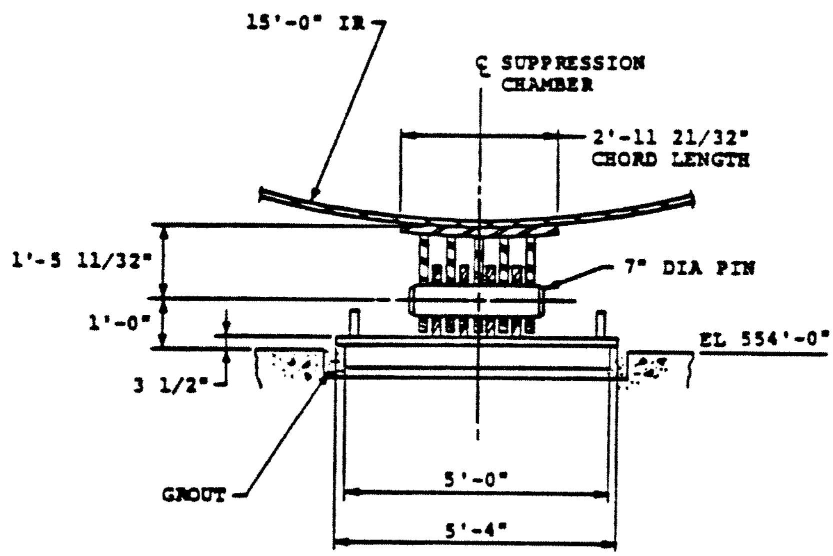


SECTION F-F
FIGURE 3.8-10

QUAD CITIES STATION UNITS 1 & 2
SUPPRESSION CHAMBER RING GIRDER AND COLUMN CONNECTION DETAILS
FIGURE 3.8-12

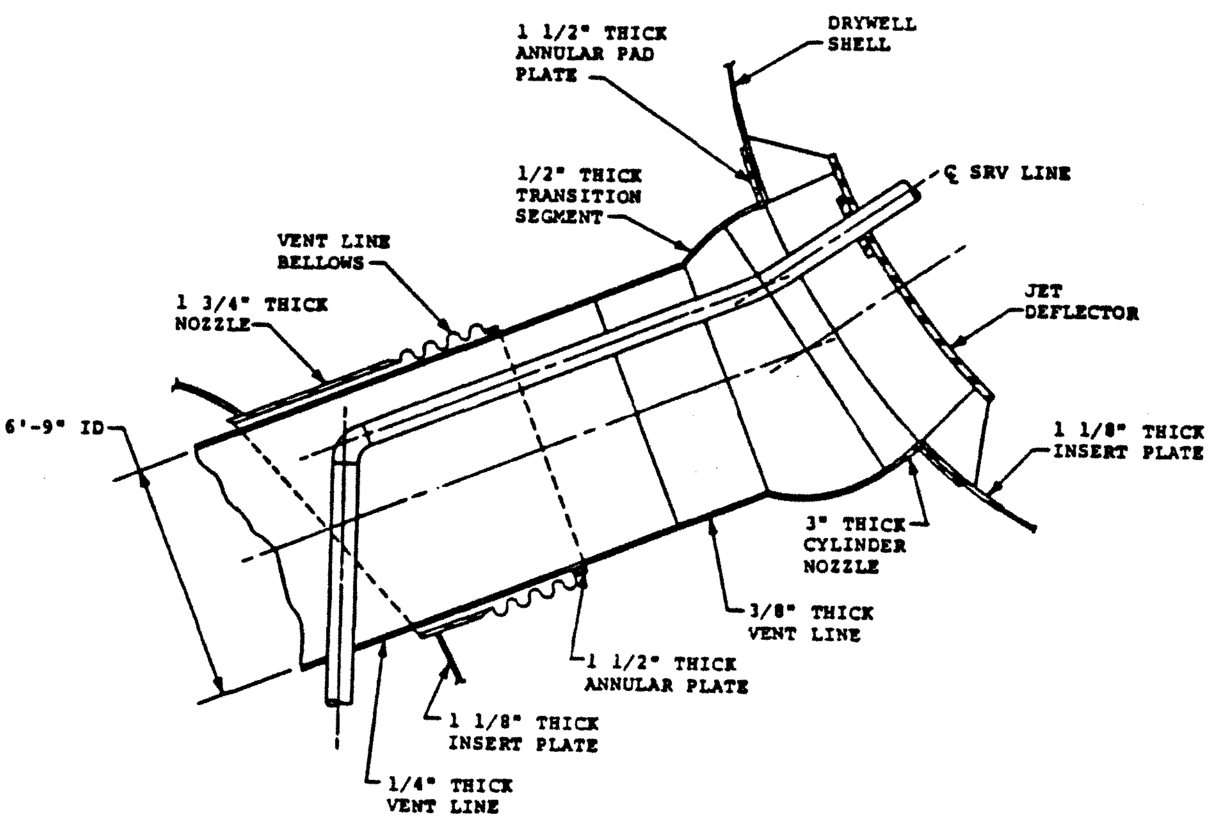


ELEVATION VIEW



SECTION A-A

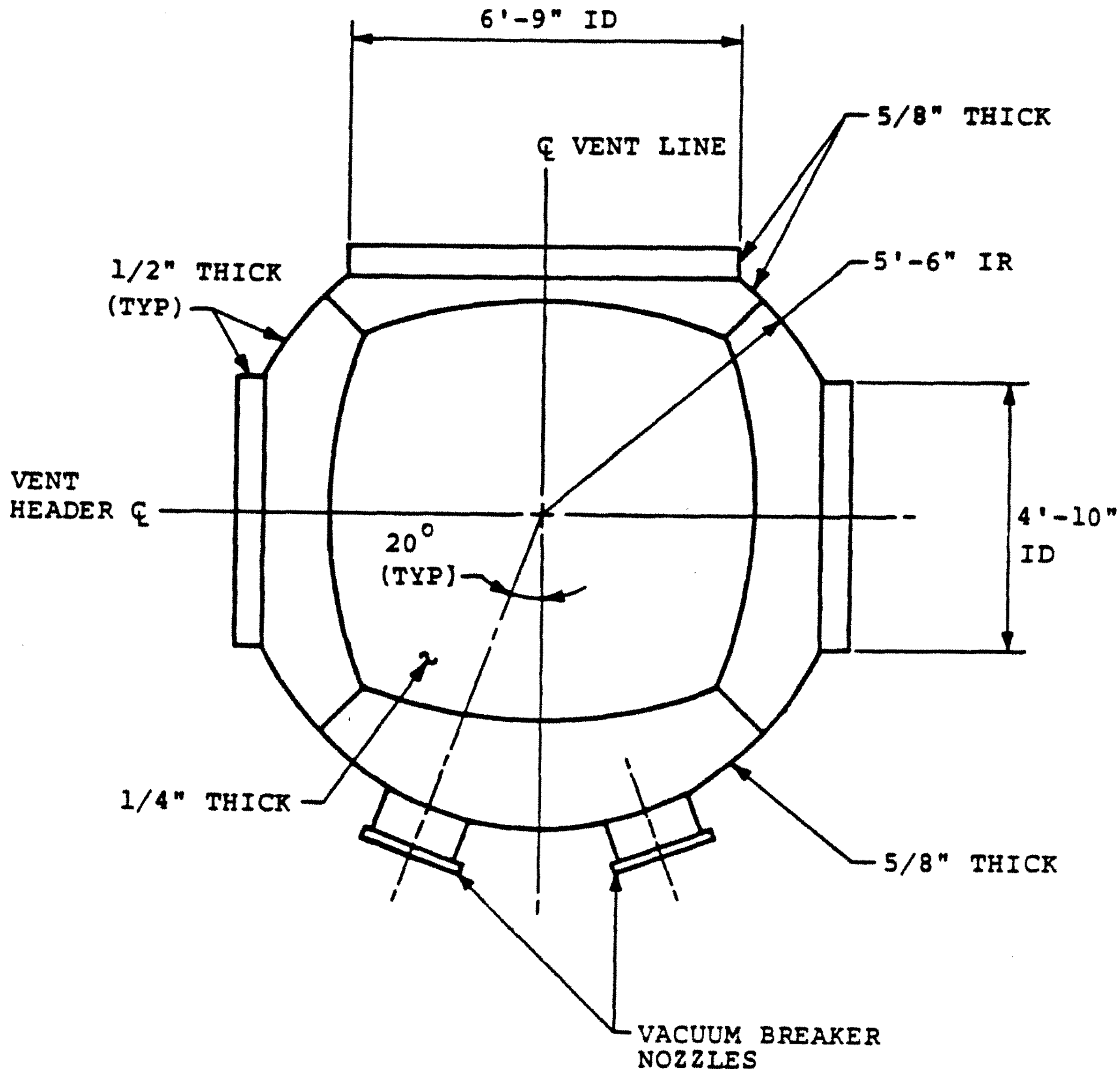
QUAD CITIES STATION UNITS 1 & 2
SUPPRESSION CHAMBER SEISMIC RESTRAINT
FIGURE 3.8-13



QUAD CITIES STATION
UNITS 1 & 2

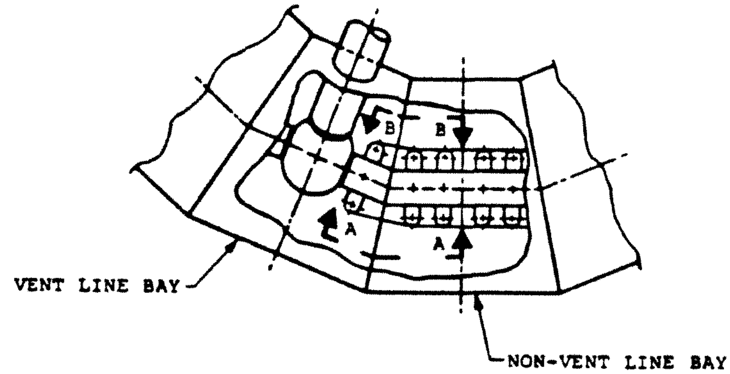
VENT LINE DETAILS - UPPER END

FIGURE 3.8-14

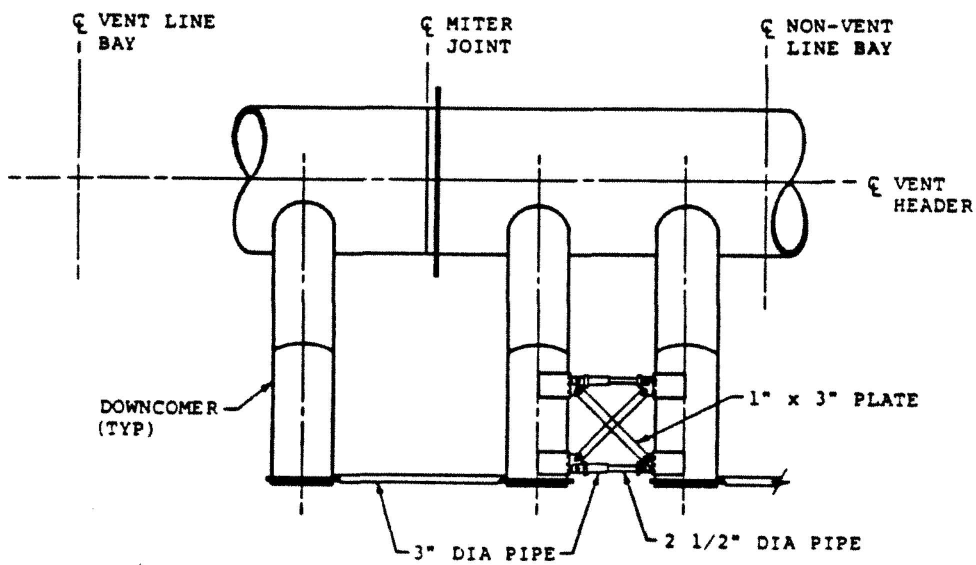


NOTE:
VACUUM BREAKERS NOT SHOWN FOR CLARITY

QUAD CITIES STATION UNITS 1 & 2
VENT LINE - VENT HEADER SPHERICAL JUNCTION
FIGURE 3.8-15



PARTIAL PLAN VIEW OF SUPPRESSION CHAMBER



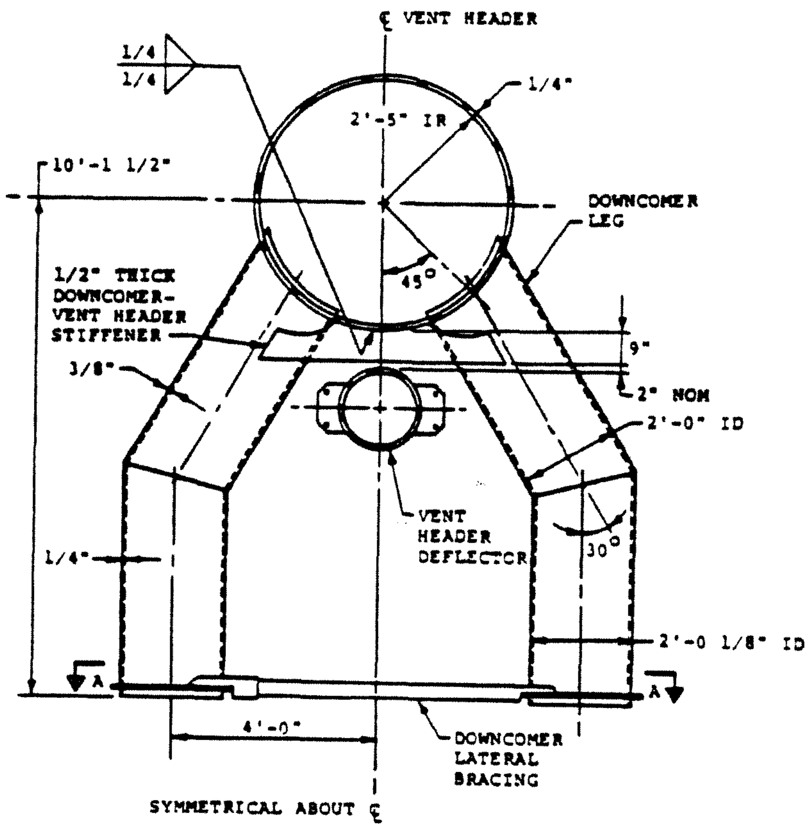
VIEW A-A

VIEW B-B (OPPOSITE HAND)

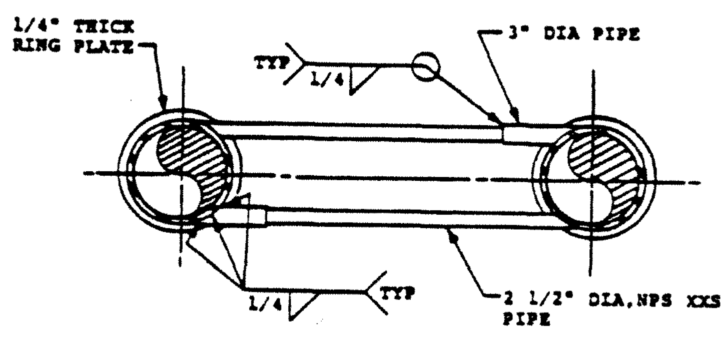
NOTE:

VENT HEADER DEFLECTOR AND VENT HEADER COLUMNS NOT SHOWN FOR CLARITY

QUAD CITIES STATION UNITS 1 & 2
DEVELOPED VIEW OF DOWNCOMER LONGITUDINAL BRACING SYSTEM
FIGURE 3.8-16

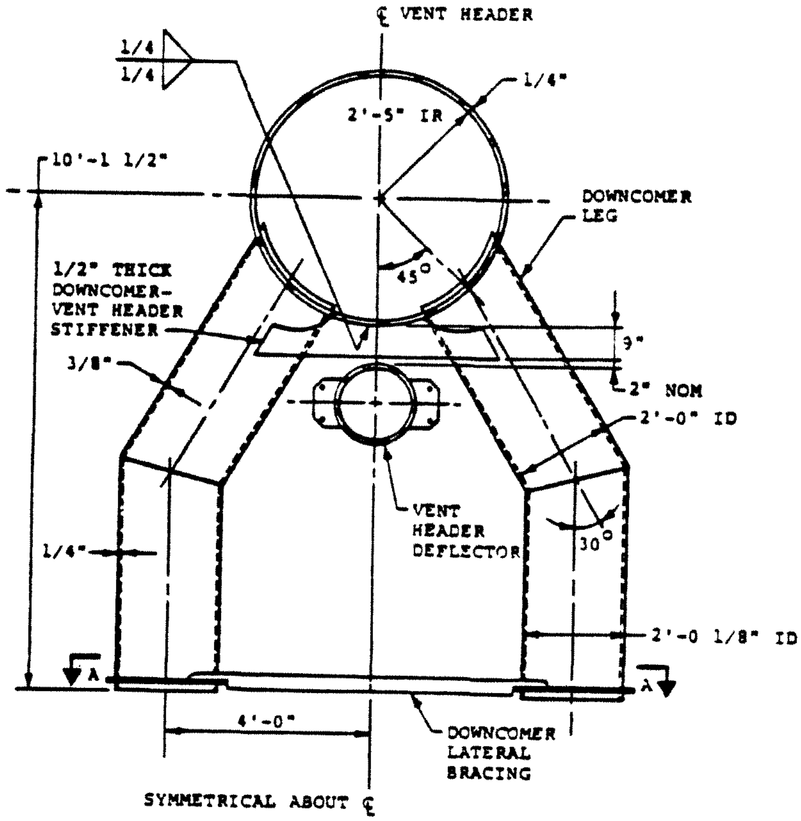


ELEVATION VIEW

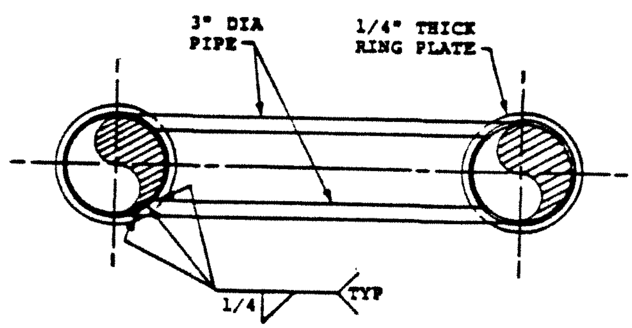


SECTION A-A

QUAD CITIES STATION UNITS 1 & 2
DOWNCOMER-TO-VENT HEADER INTERSECTION DETAILS - UNIT 1
FIGURE 3.8-17

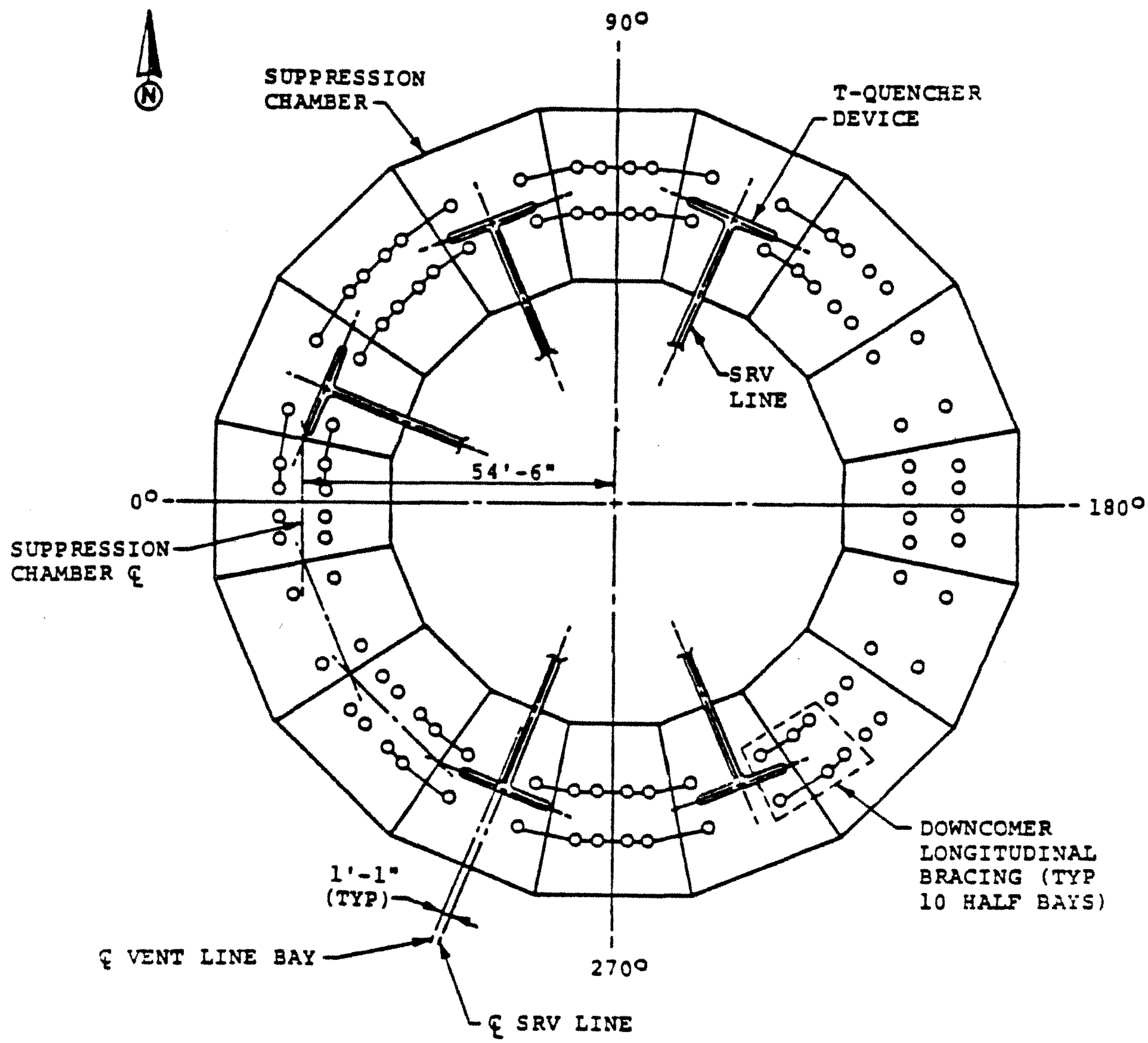


ELEVATION VIEW



SECTION A-A

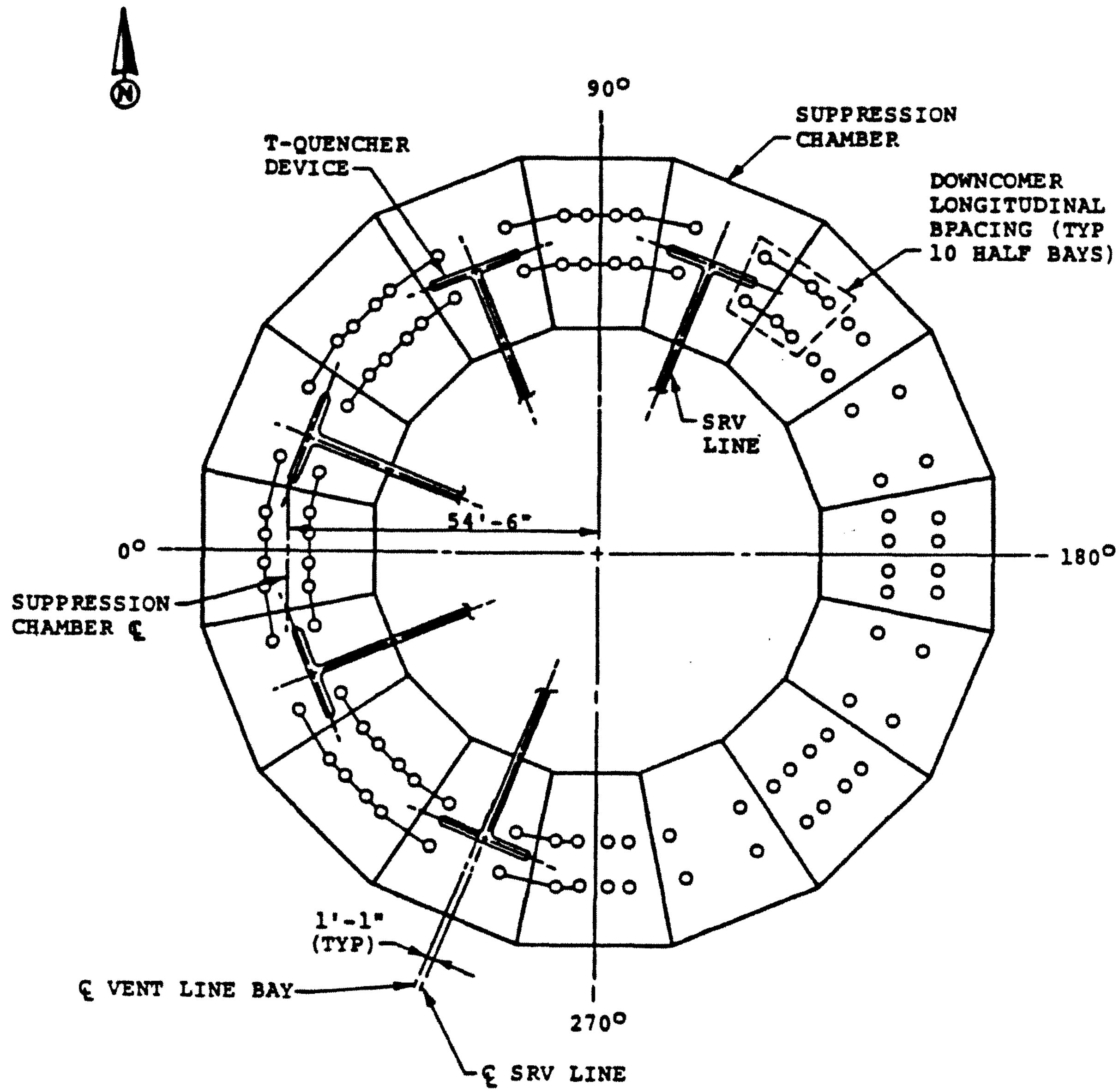
QUAD CITIES STATION UNITS 1 & 2
DOWNCOMER-TO-VENT HEADER INTERSECTION DETAILS - UNIT 2
FIGURE 3.8-18



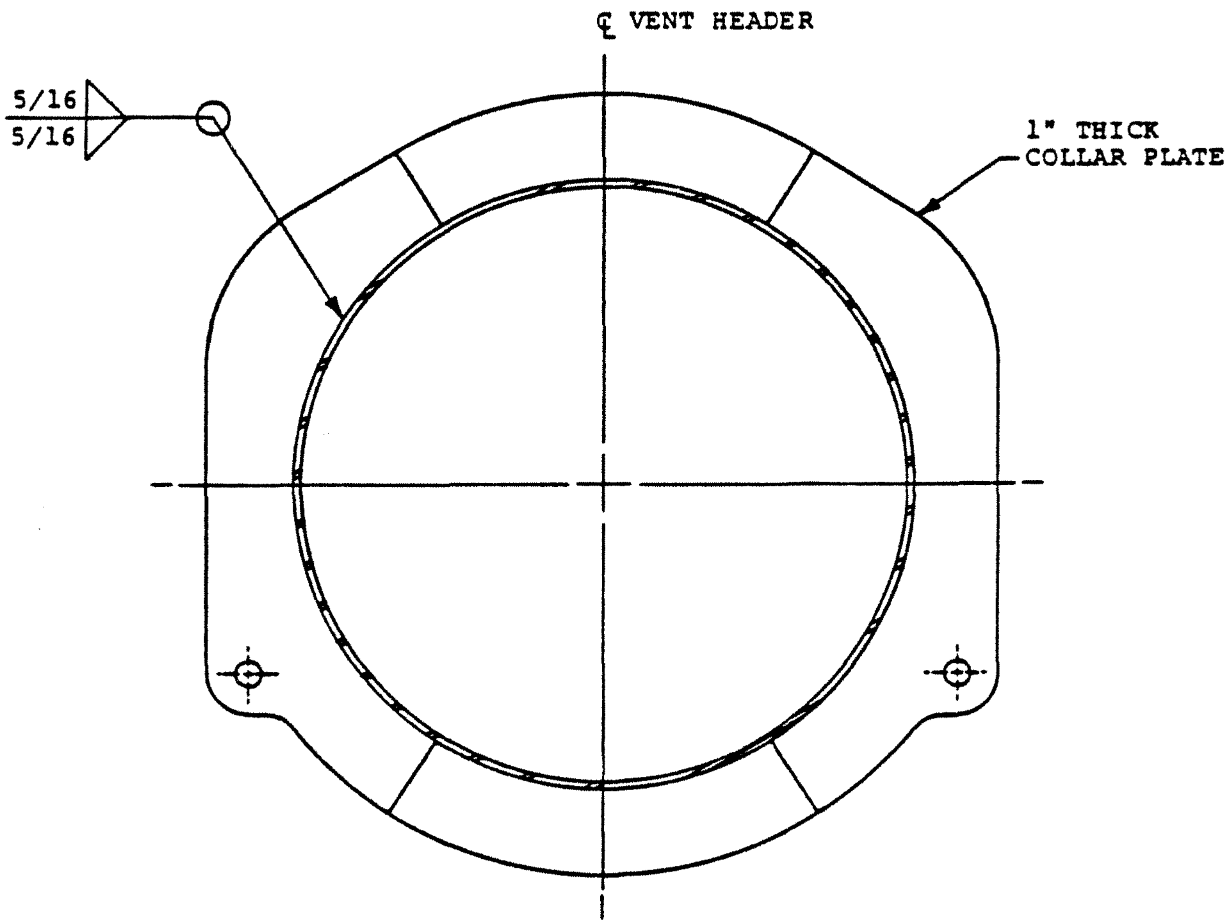
QUAD CITIES STATION
UNITS 1 & 2

DOWNCOMER LONGITUDINAL BRACING SYSTEM CONFIGURATION - UNIT 1

FIGURE 3.8-19

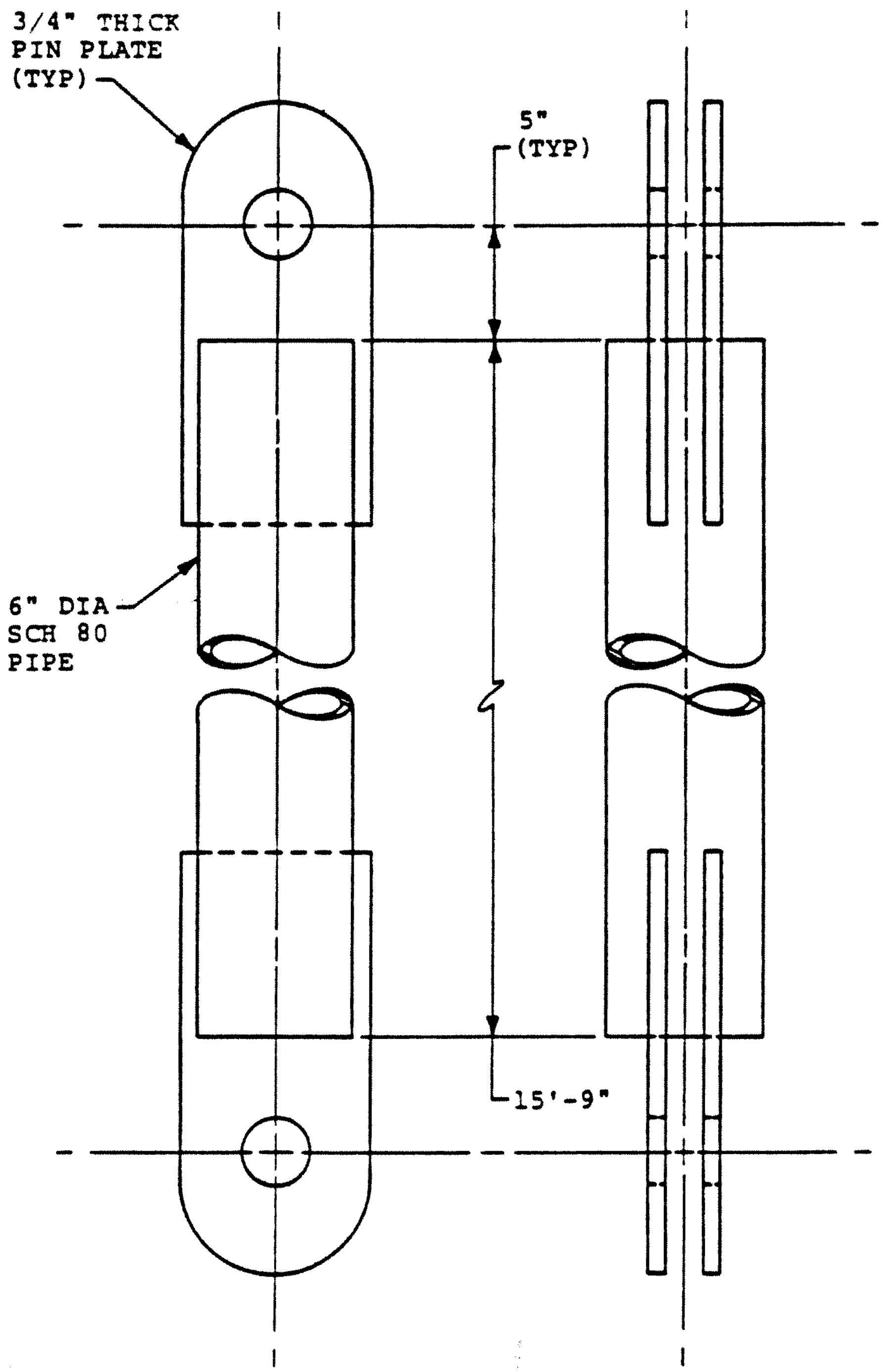


QUAD CITIES STATION UNITS 1 & 2
DOWNCOMER LONGITUDINAL BRACING SYSTEM CONFIGURATION - UNIT 2
FIGURE 3.8-20



SECTION THROUGH VENT HEADER
SUPPORT COLLAR

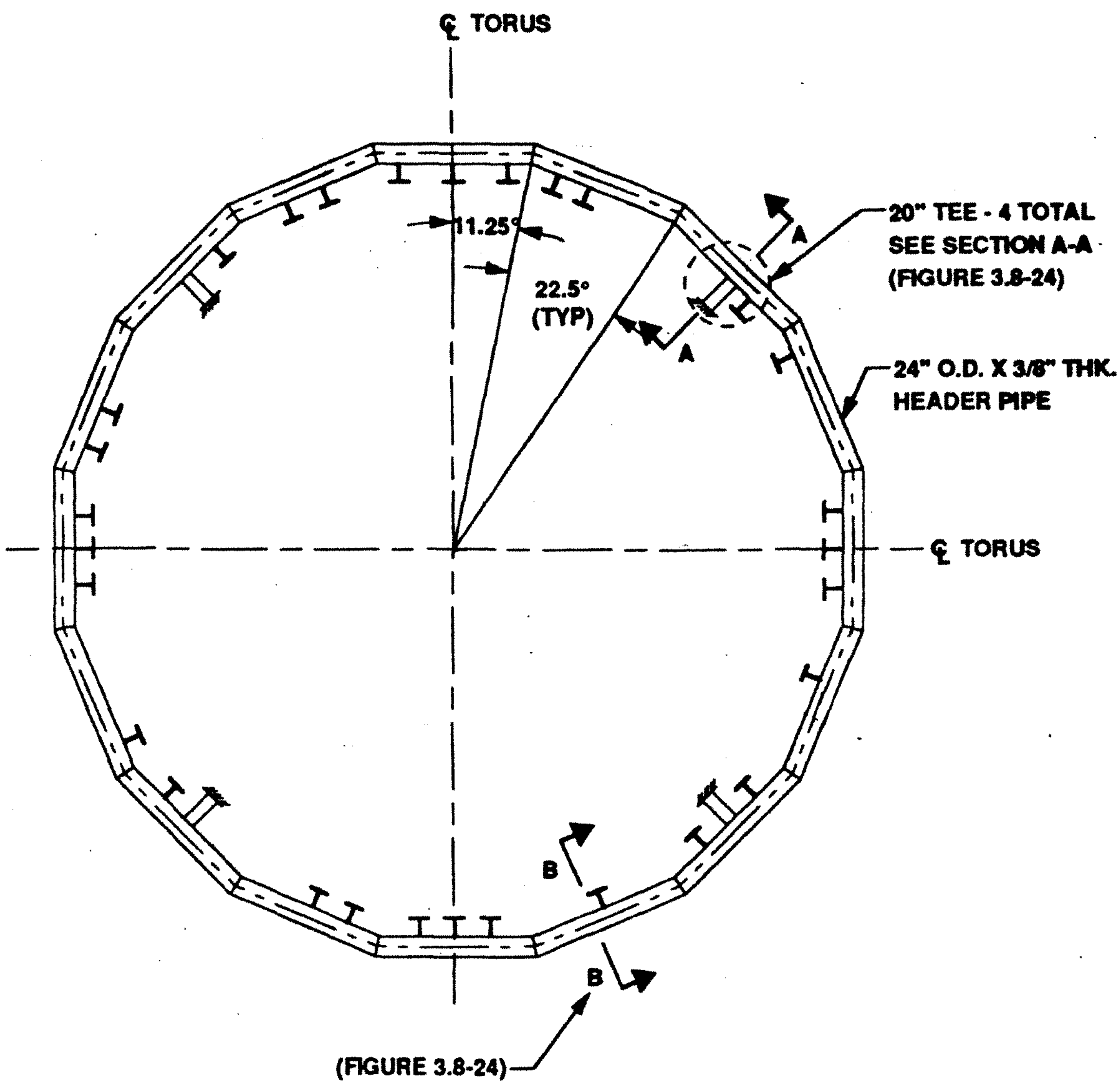
QUAD CITIES STATION UNITS 1 & 2
VENT HEADER SUPPORT COLLAR PLATE DETAILS
FIGURE 3.8-21



QUAD CITIES STATION
UNITS 1 & 2

VENT SYSTEM SUPPORT COLUMN DETAILS

FIGURE 3.8-22

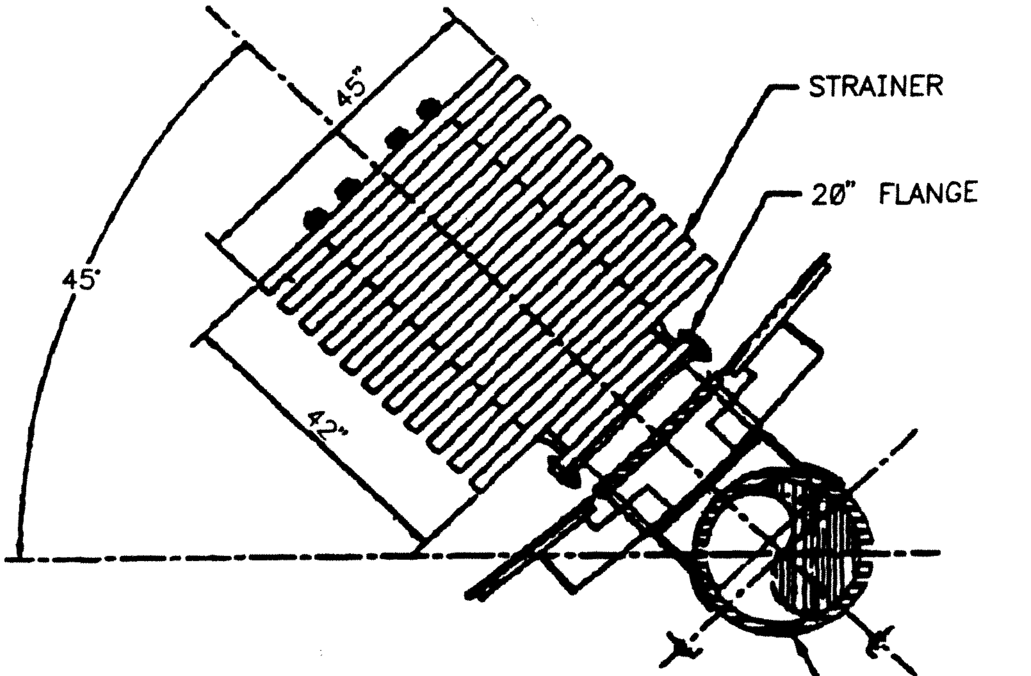


QUAD CITIES STATION
UNITS 1 & 2

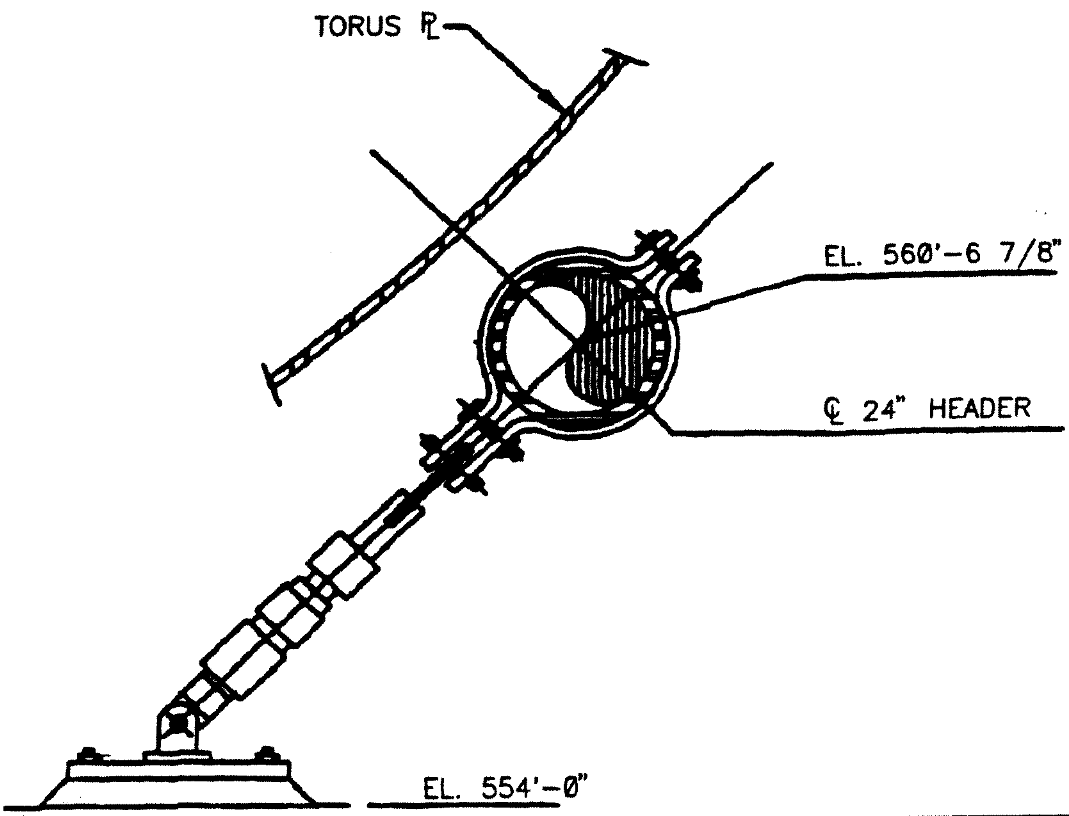
SUPPRESSION CHAMBER RING HEADER
GENERAL PLAN AND LOCATION OF SUPPORTS

FIGURE 3.8-23

REVISION 4, APRIL 1997

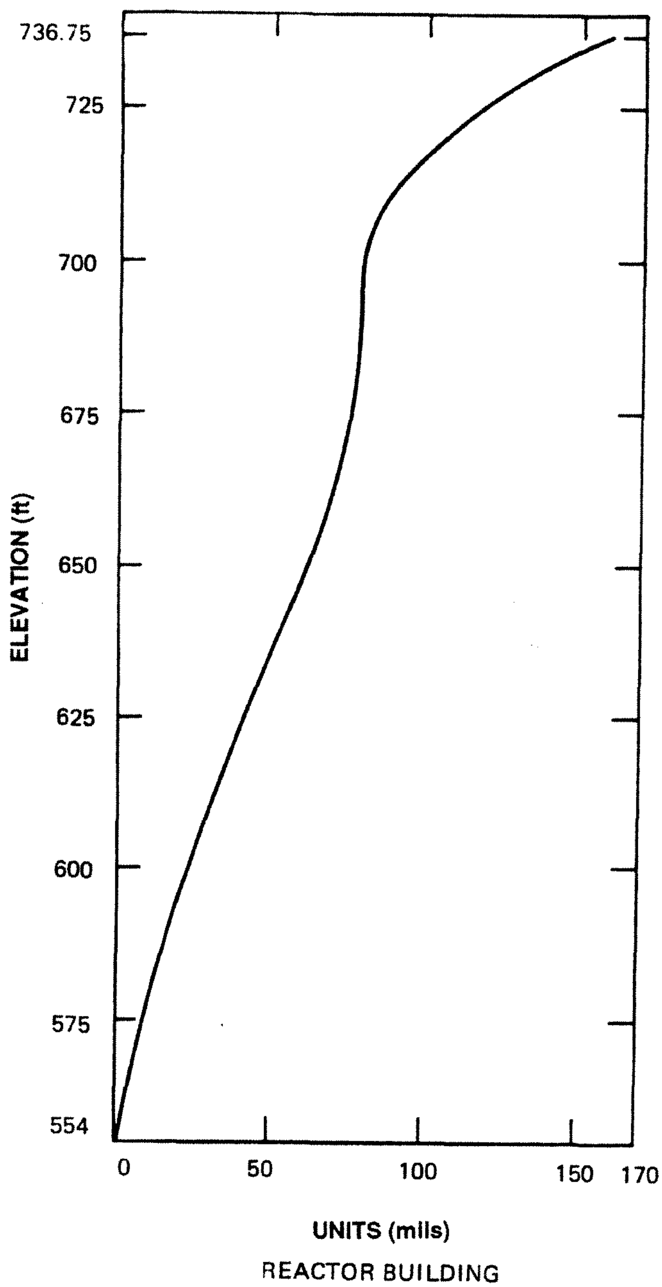


SECTION A-A
(FIGURE 3.8-23)



SECTION B-B
(FIGURE 3.8-23)

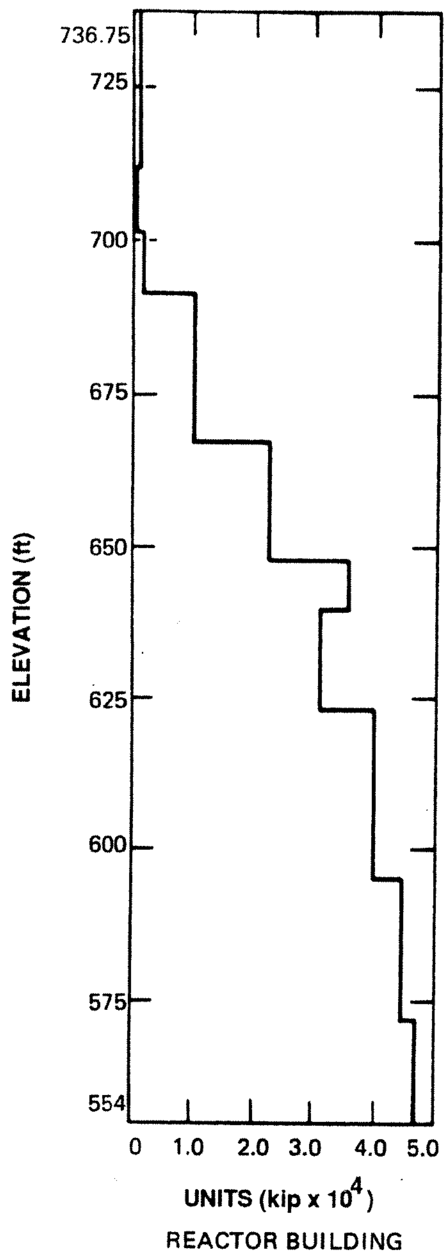
QUAD CITIES STATION UNITS 1 & 2
SUPPRESSION CHAMBER RING HEADER SECTIONS
FIGURE 3.8-24 REVISION 5, JUNE 1999

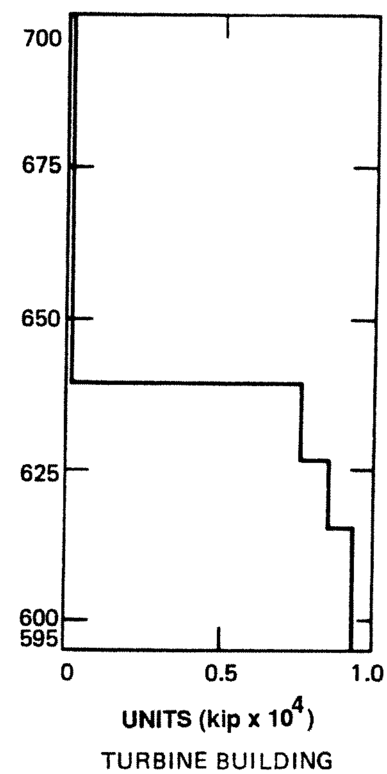


QUAD CITIES STATION
UNITS 1 & 2

REACTOR AND TURBINE BUILDINGS OBE
DISPLACEMENT DIAGRAM (N-S DIRECTION)

FIGURE 3.8-25

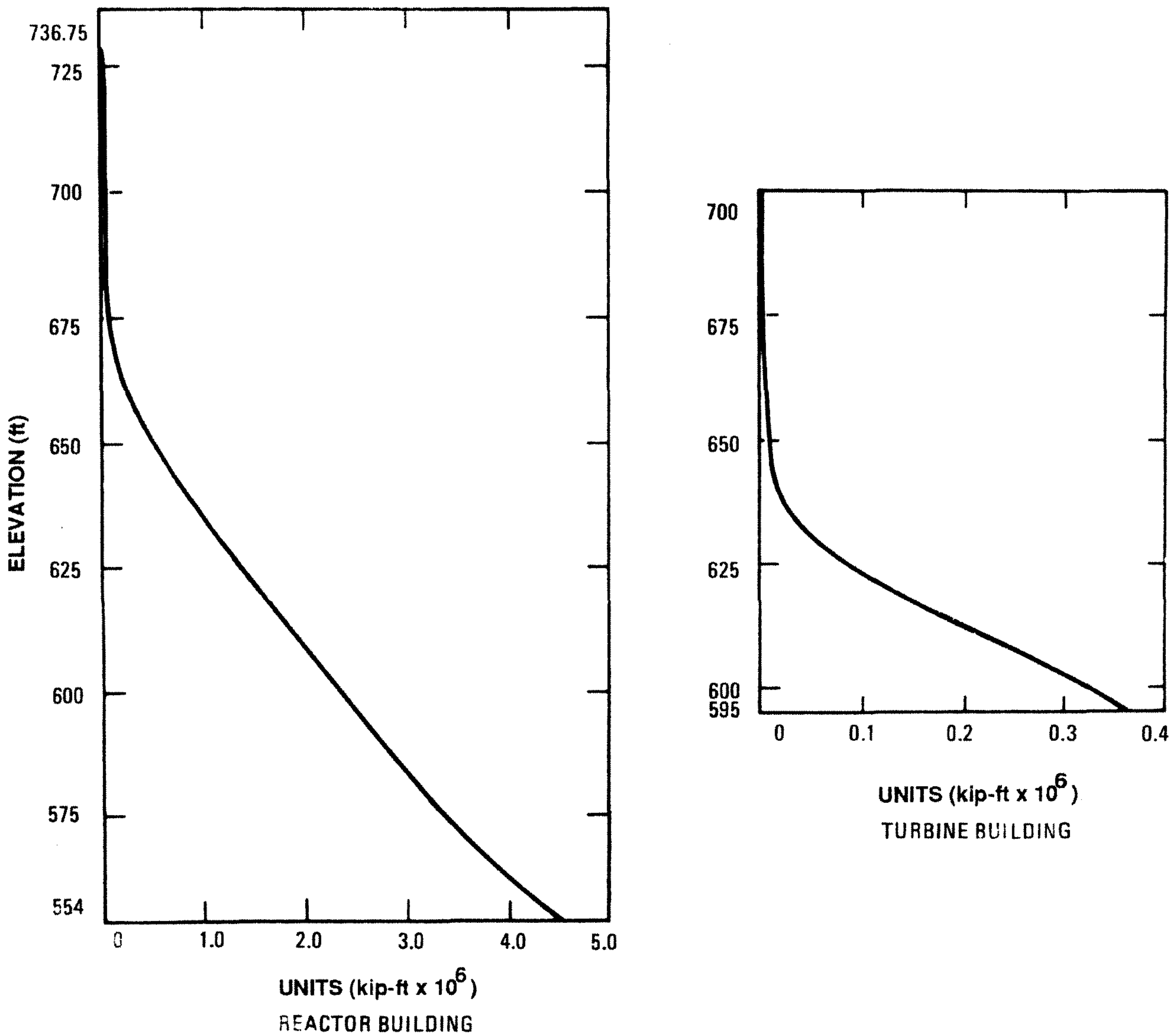




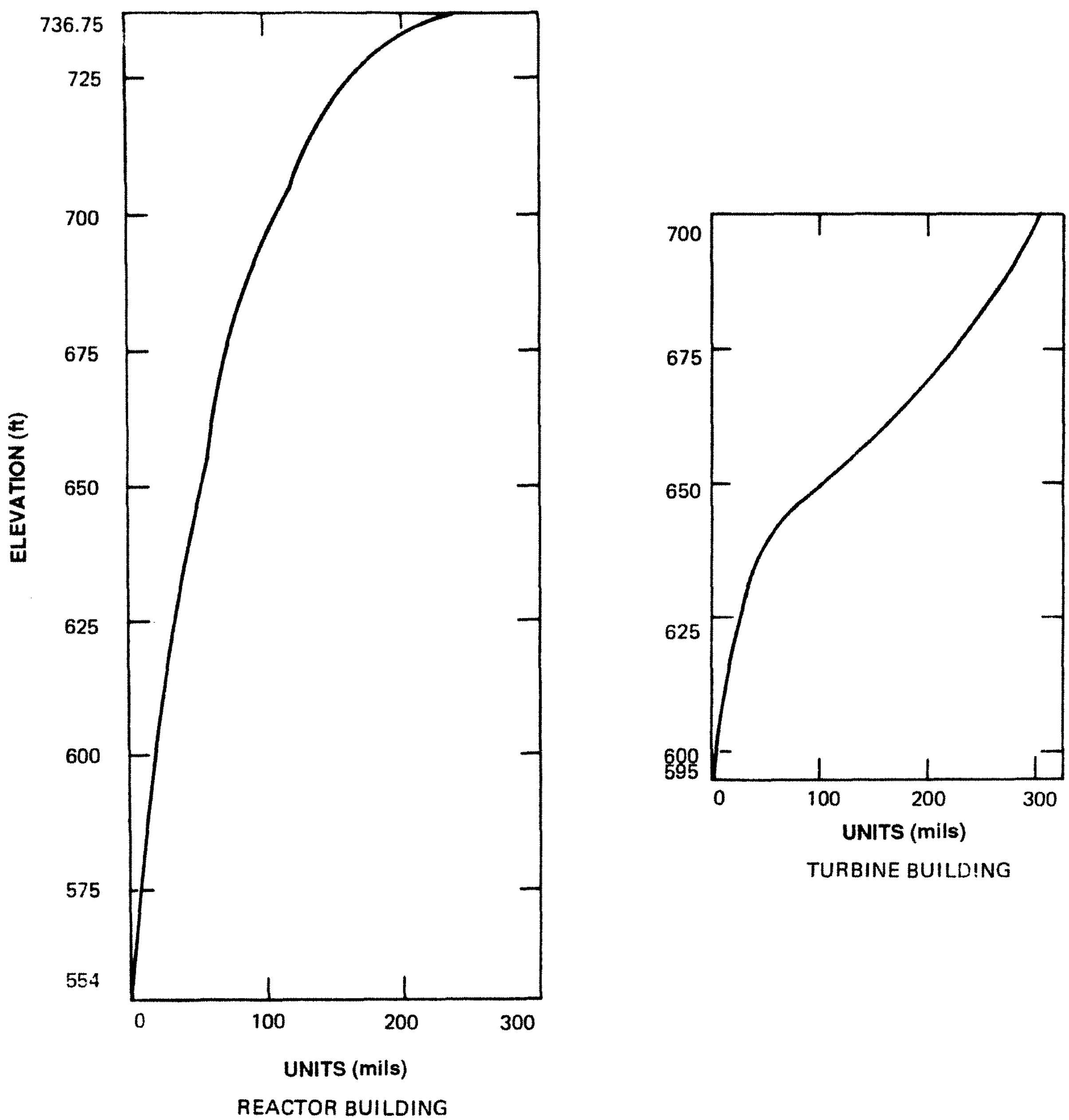
QUAD CITIES STATION
UNITS 1 & 2

REACTOR AND TURBINE BUILDINGS OBE
SHEAR DIAGRAM (N-S DIRECTION)

FIGURE 3.8-26



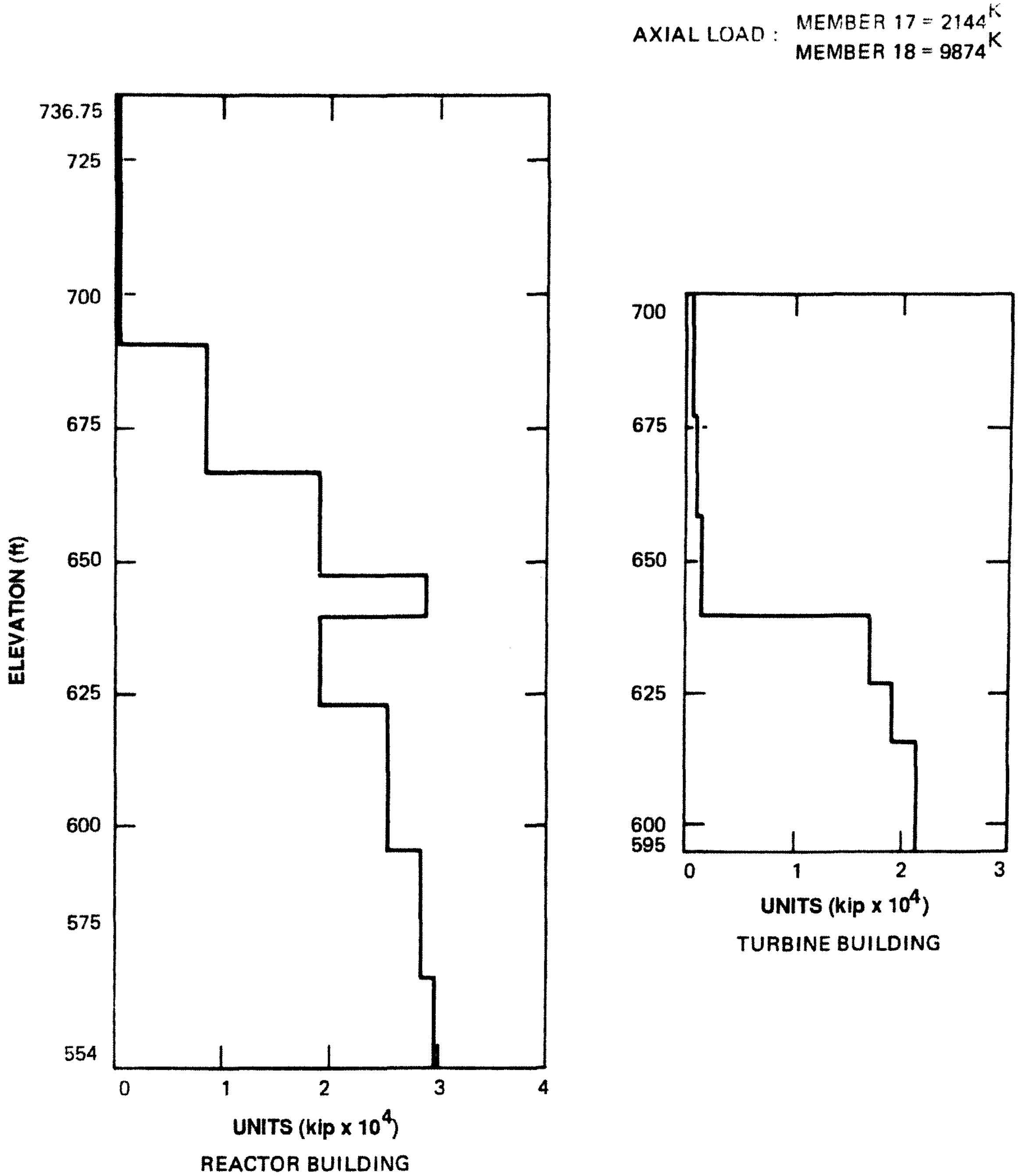
QUAD CITIES STATION UNITS 1 & 2
REACTOR AND TURBINE BUILDINGS OBE MOMENT DIAGRAM (N-S DIRECTION)
FIGURE 3.8-27



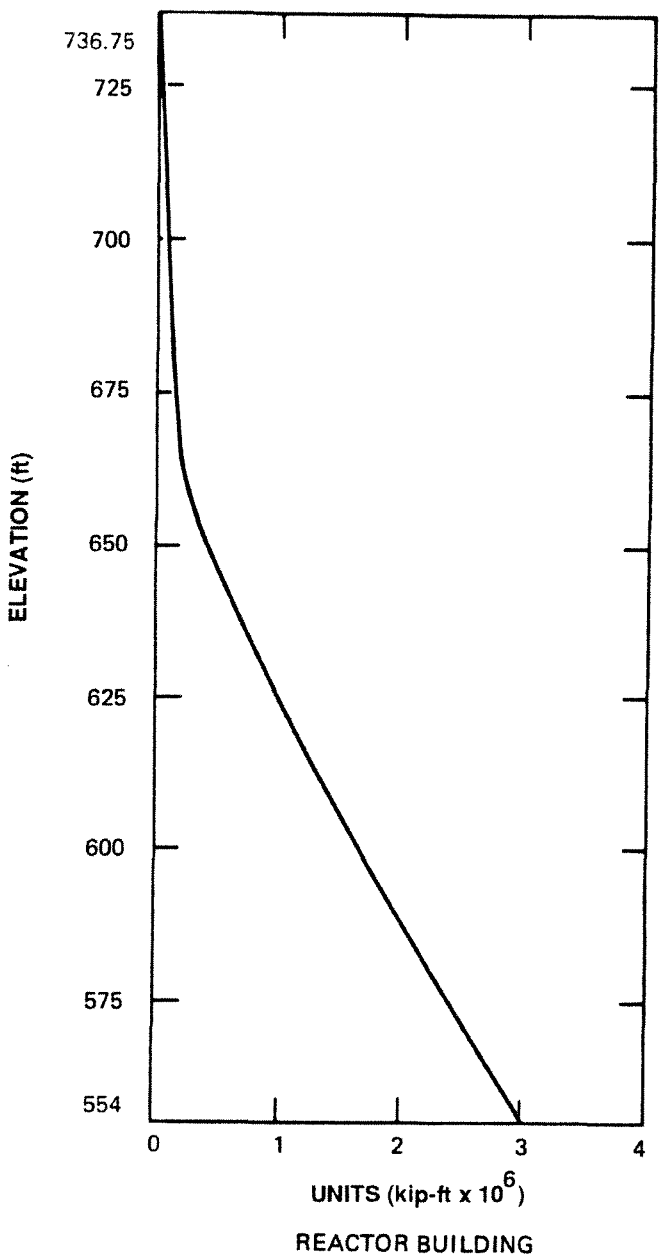
QUAD CITIES STATION
UNITS 1 & 2

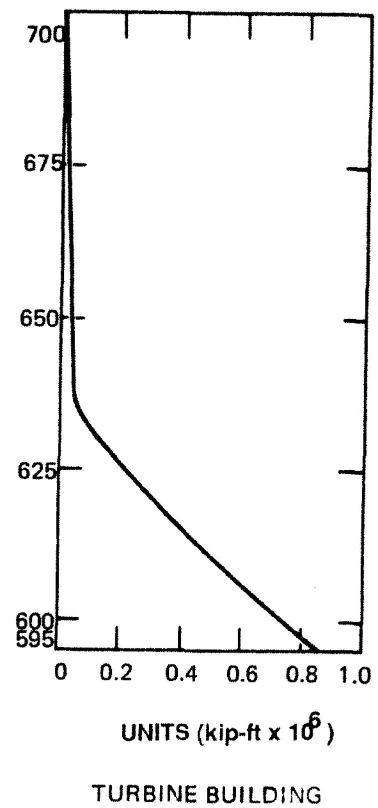
REACTOR AND TURBINE BUILDINGS OBE DISPLACEMENT DIAGRAM (E-W DIRECTION)

FIGURE 3.8-28

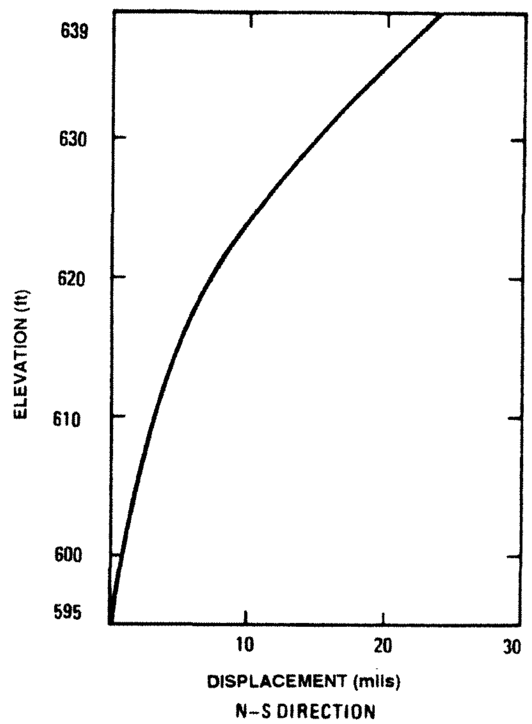


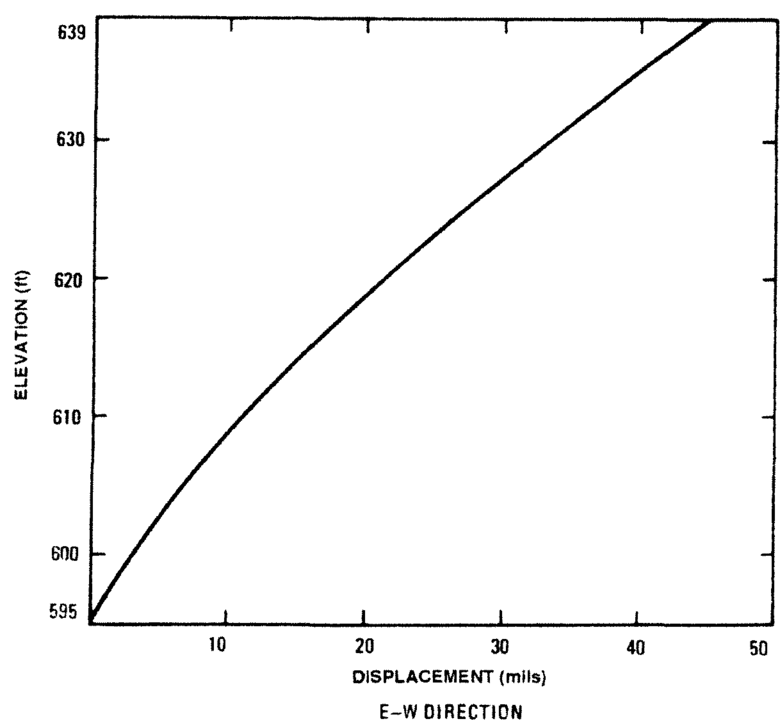
QUAD CITIES STATION UNITS 1 & 2
REACTOR AND TURBINE BUILDINGS OBE SHEAR DIAGRAM (E-W DIRECTION)
FIGURE 3.8-29





QUAD CITIES STATION UNITS 1 & 2
REACTOR AND TURBINE BUILDINGS OBE MOMENT DIAGRAM (E-W DIRECTION)
FIGURE 3.8-30

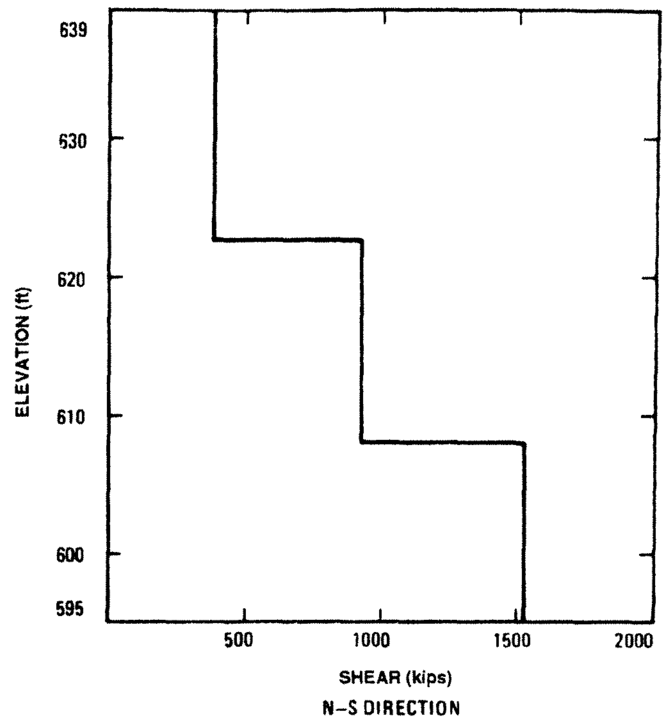


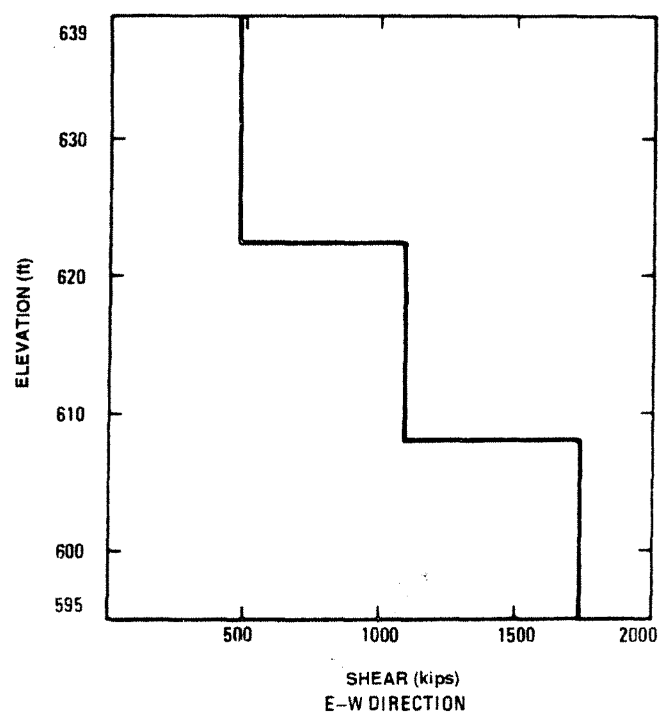


QUAD CITIES STATION
UNITS 1 & 2

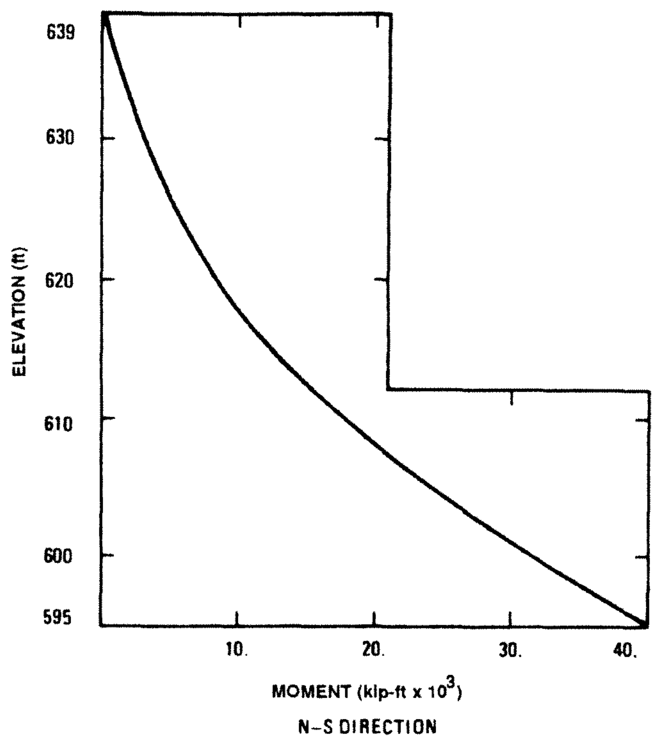
CONTROL ROOM OBE DISPLACEMENT DIAGRAMS

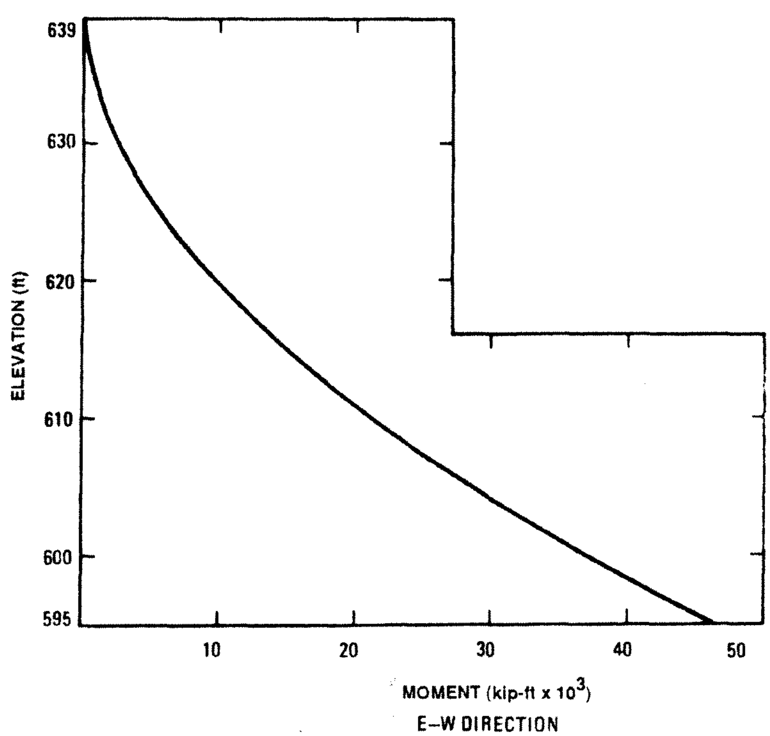
FIGURE 3.8-31





QUAD CITIES STATION UNITS 1 & 2
CONTROL ROOM OBE SHEAR DIAGRAMS
FIGURE 3.8-32

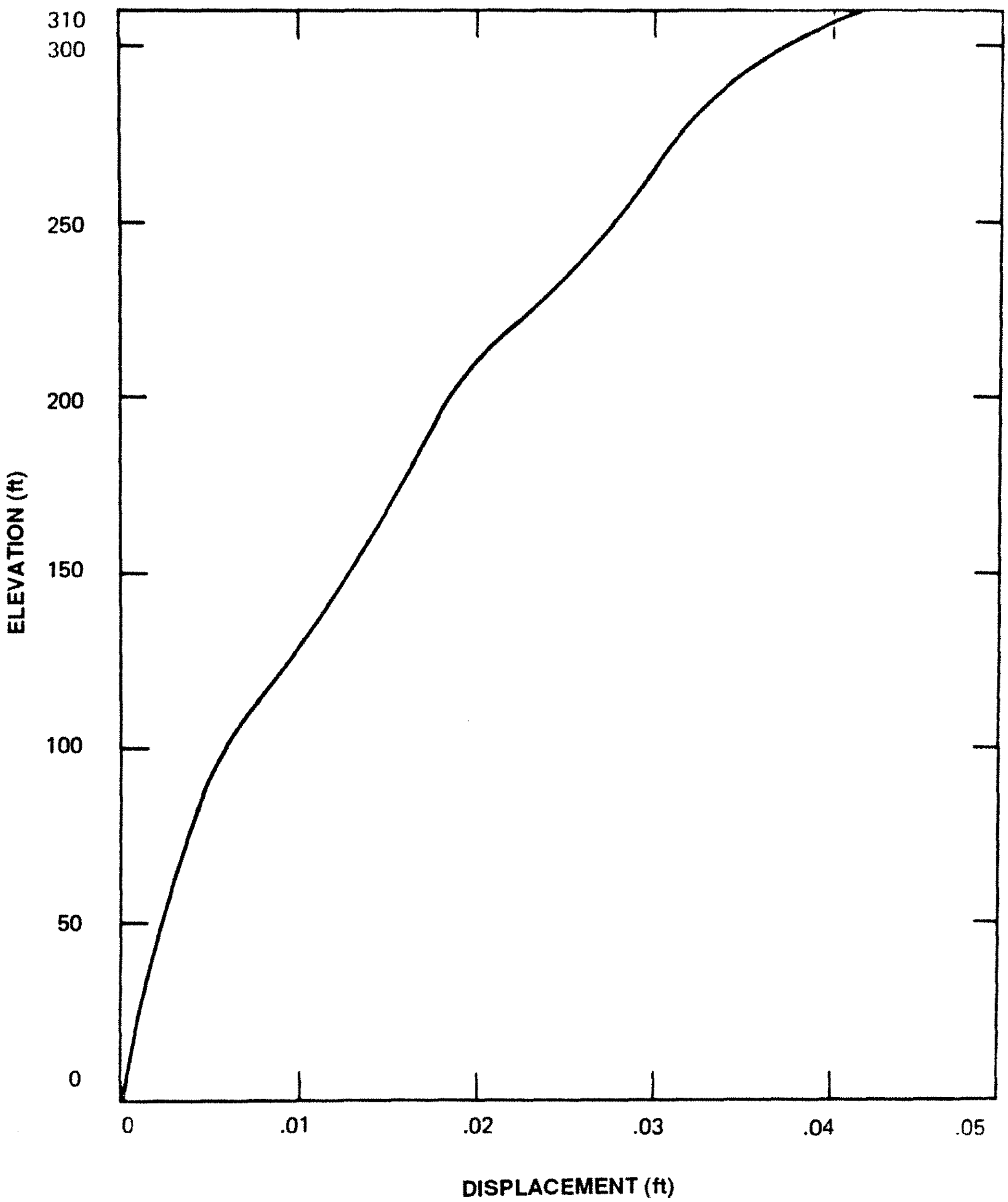




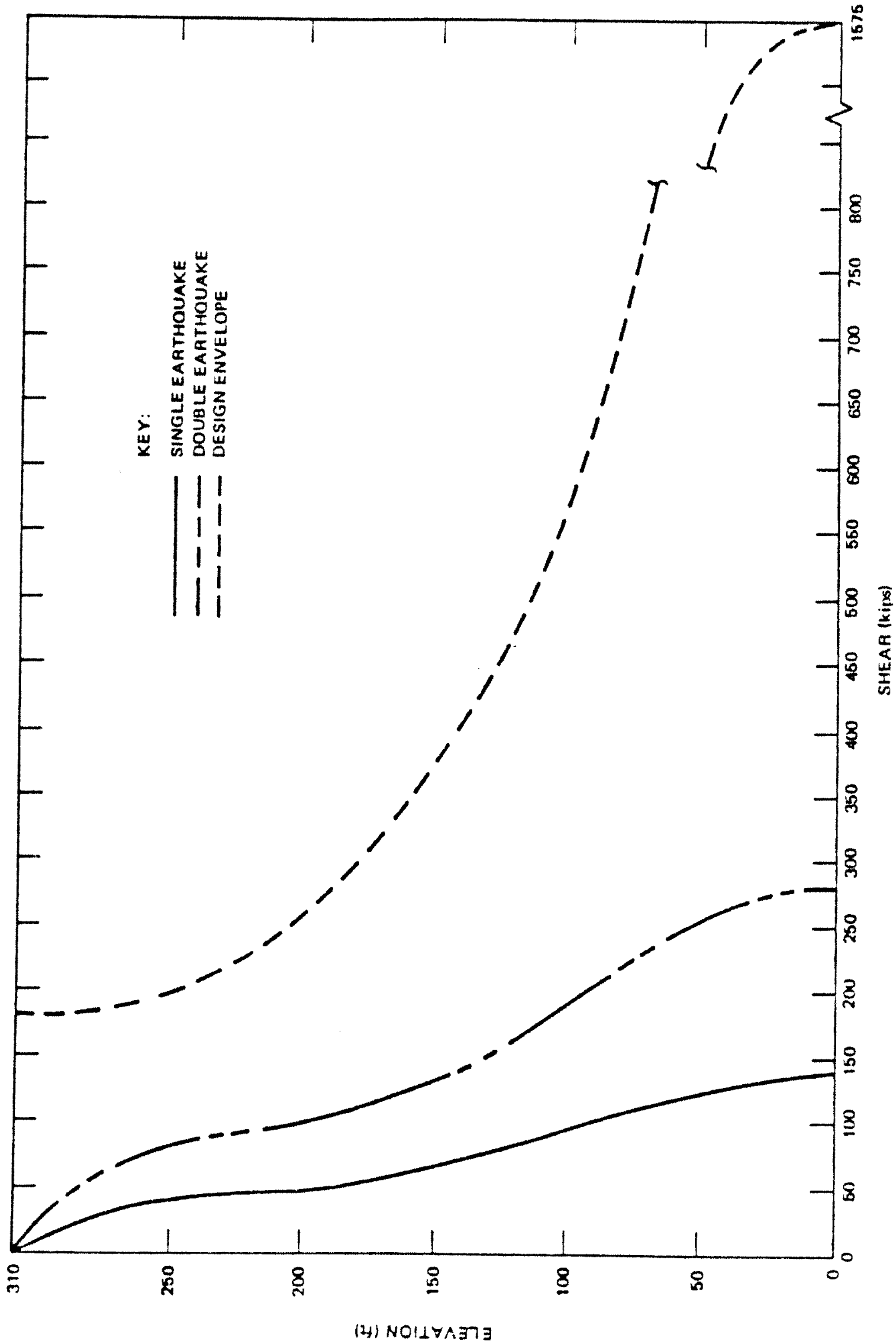
QUAD CITIES STATION
UNITS 1 & 2

CONTROL ROOM OBE MOMENT DIAGRAMS

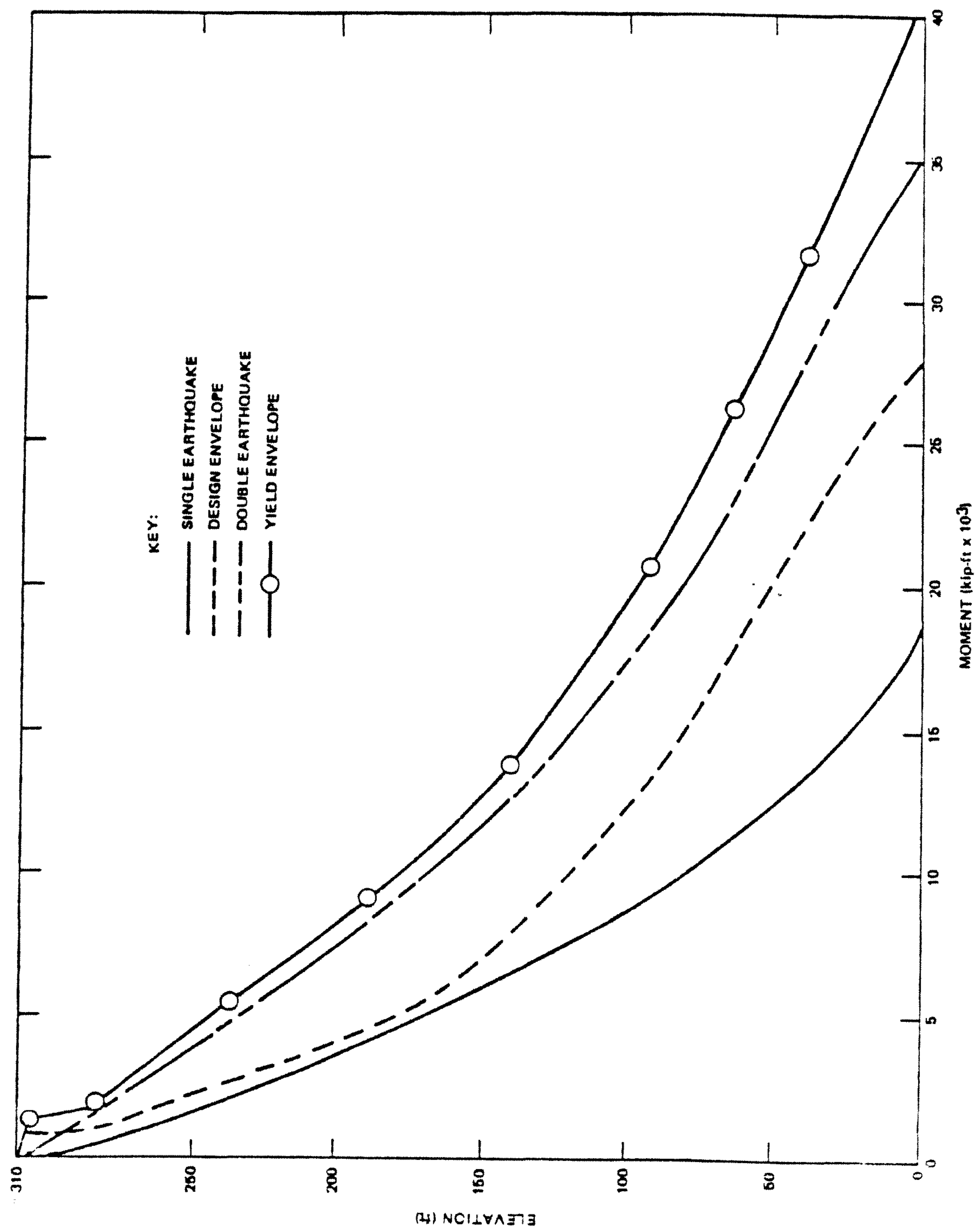
FIGURE 3.8-33



QUAD CITIES STATION UNITS 1 & 2
VENTILATION STACK OBE DISPLACEMENT DIAGRAM
FIGURE 3.8-34



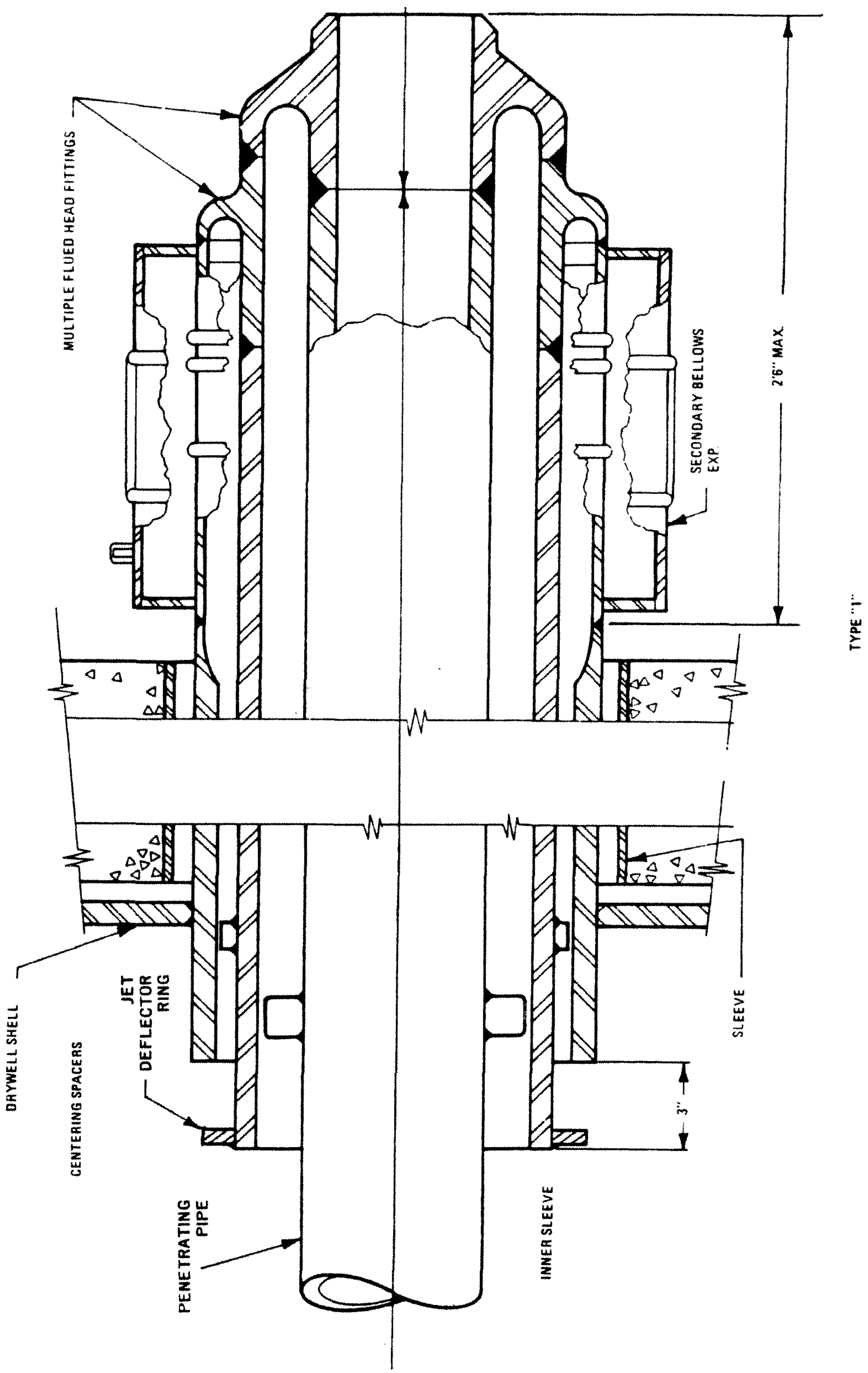
QUAD CITIES STATION UNITS 1 & 2
VENTILATION STACK EARTHQUAKE SHEARS VS. CAPACITY
FIGURE 3.8-35



QUAD CITIES STATION
UNITS 1 & 2

VENTILATION STACK EARTHQUAKE MOMENTS
VS. CAPACITY

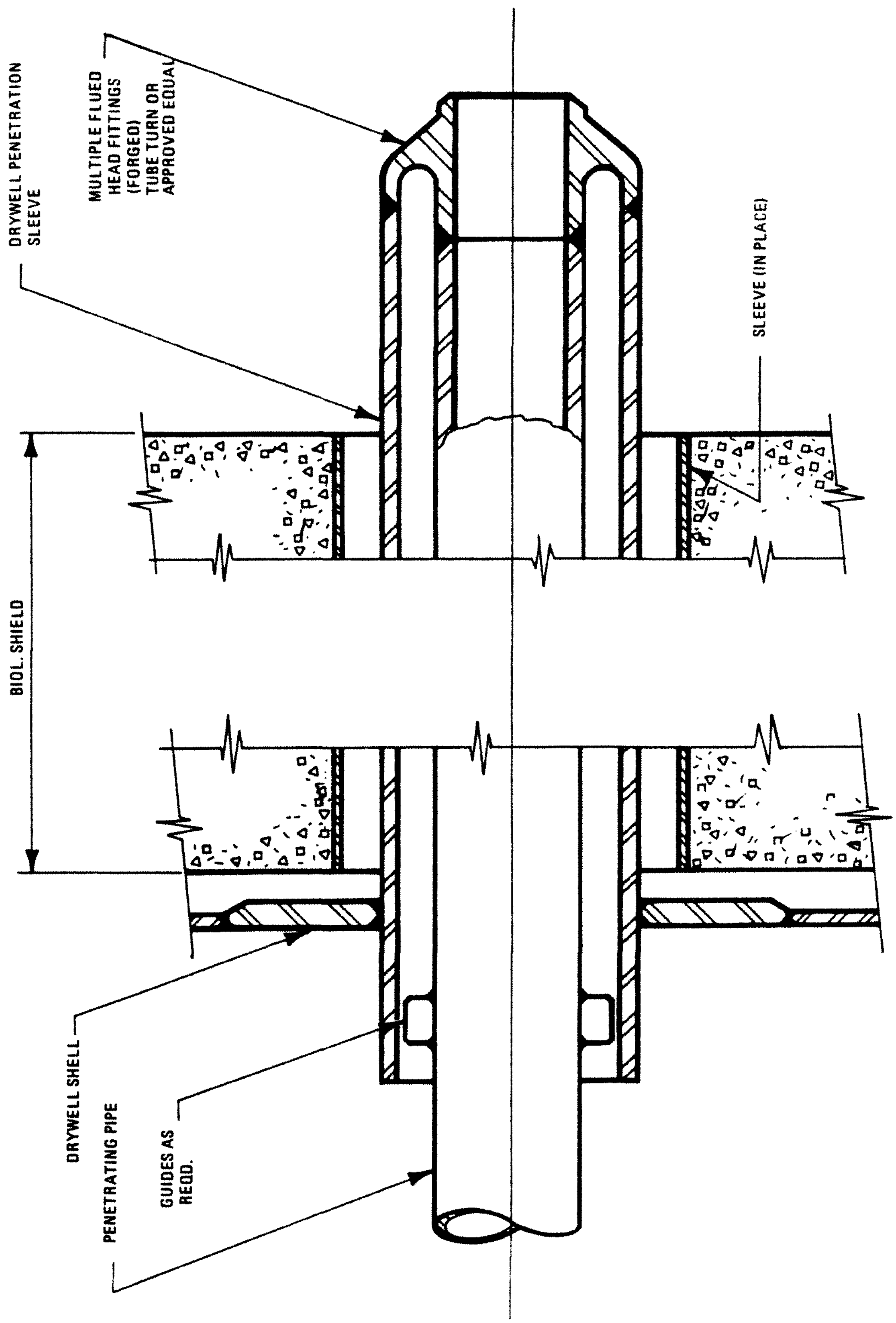
FIGURE 3.8-36



QUAD CITIES STATION
UNITS 1 & 2

HOT FLUID PIPING PENETRATION ASSEMBLY

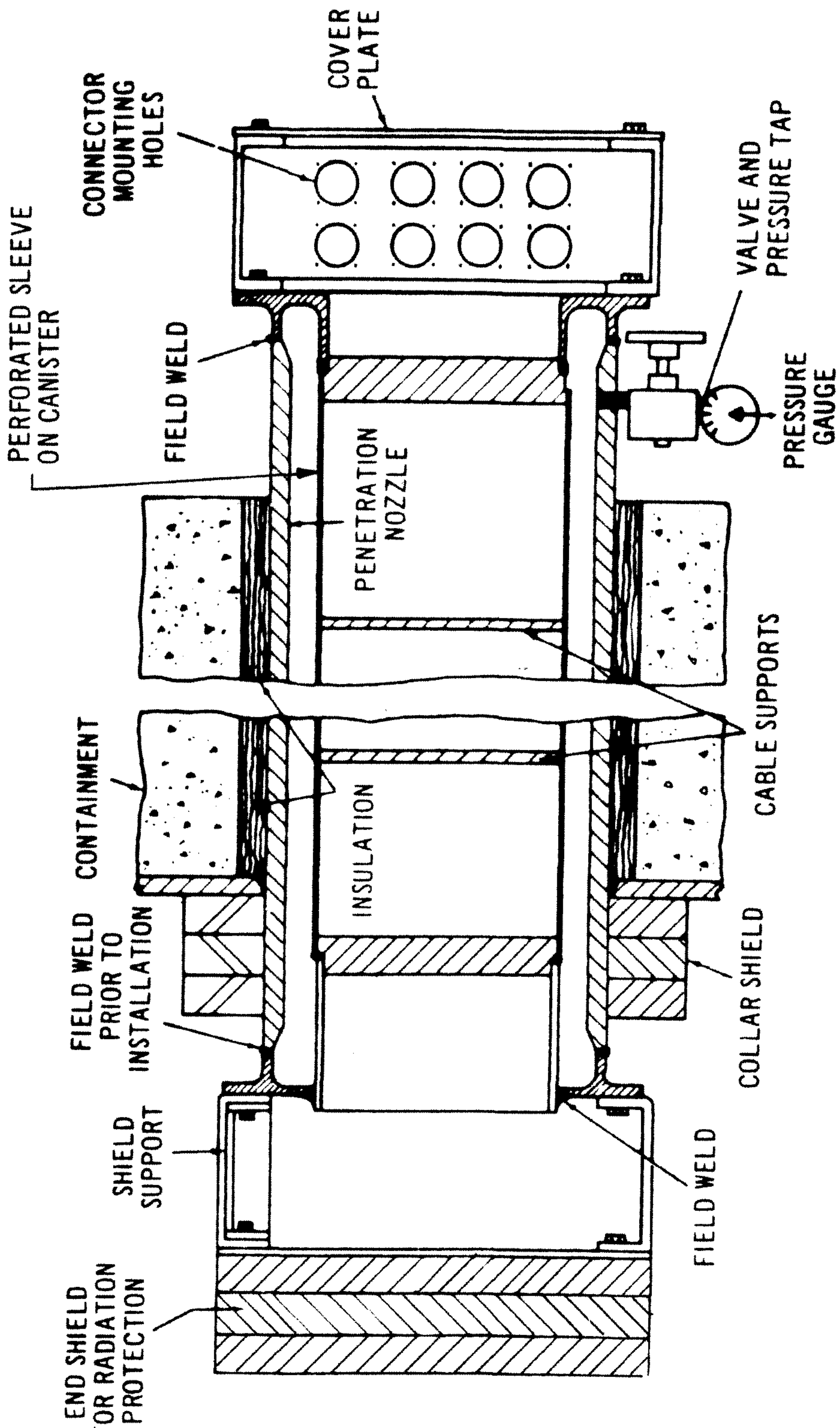
FIGURE 3.8-37



QUAD CITIES STATION
UNITS 1 & 2

COLD FLUID PIPING PENETRATION ASSEMBLY

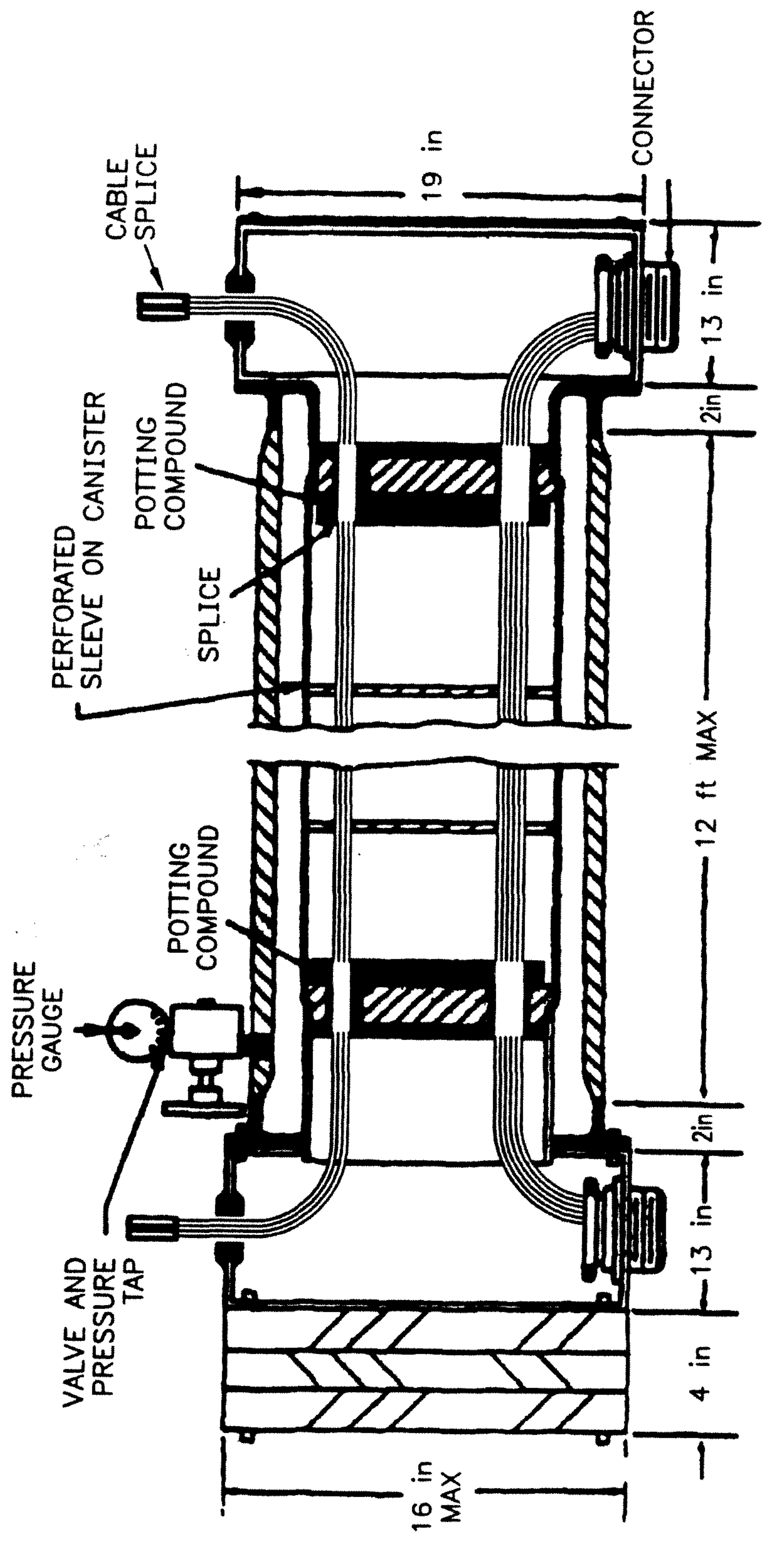
FIGURE 3.8-38



QUAD CITIES STATION
UNITS 1 & 2

TYPICAL ELECTRICAL PENETRATION ASSEMBLY
CANISTER

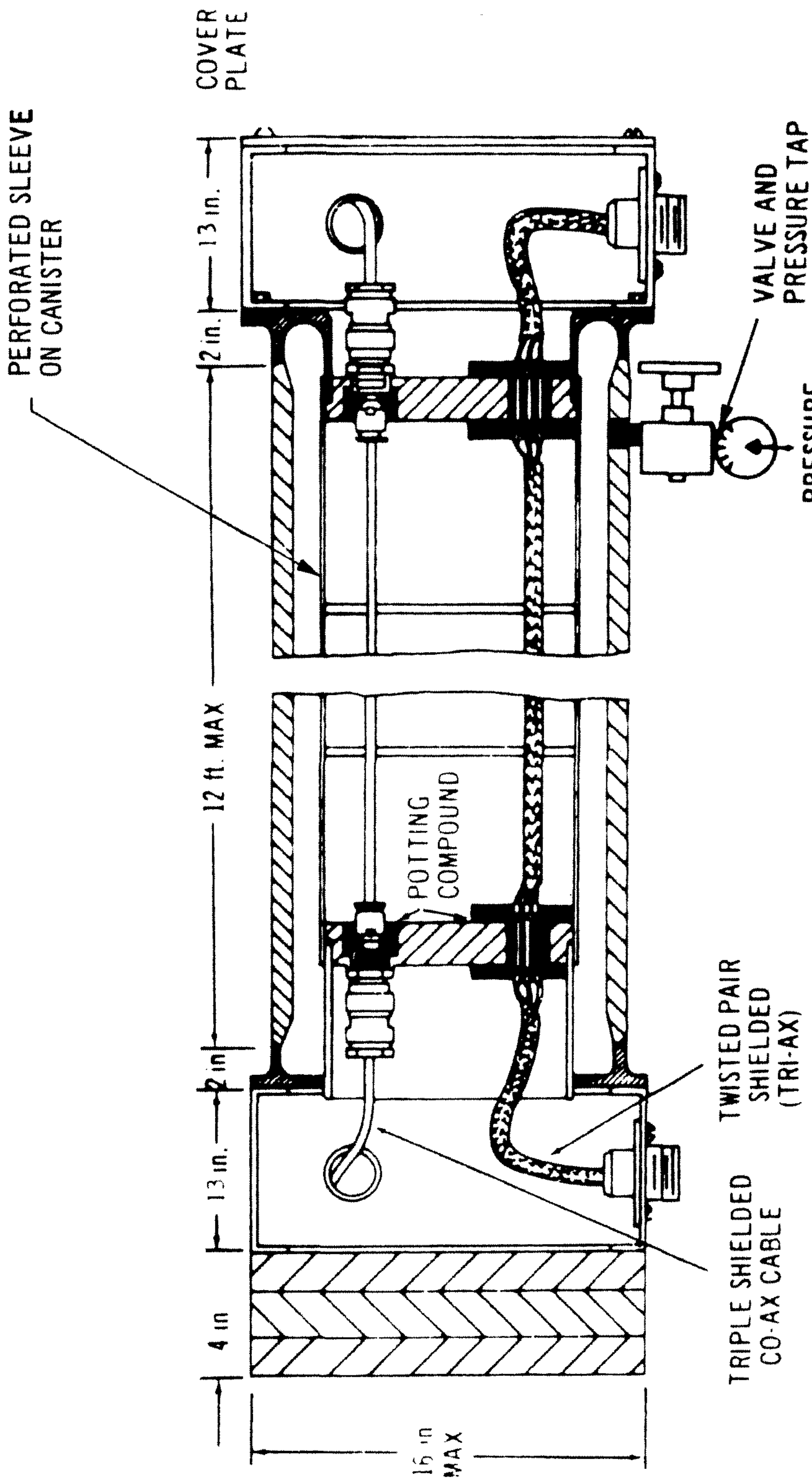
FIGURE 3.8-39



QUAD CITIES STATION
UNITS 1 & 2

GENERIC LOW VOLTAGE POWER AND CONTROL
ELECTRICAL PENETRATION ASSEMBLY

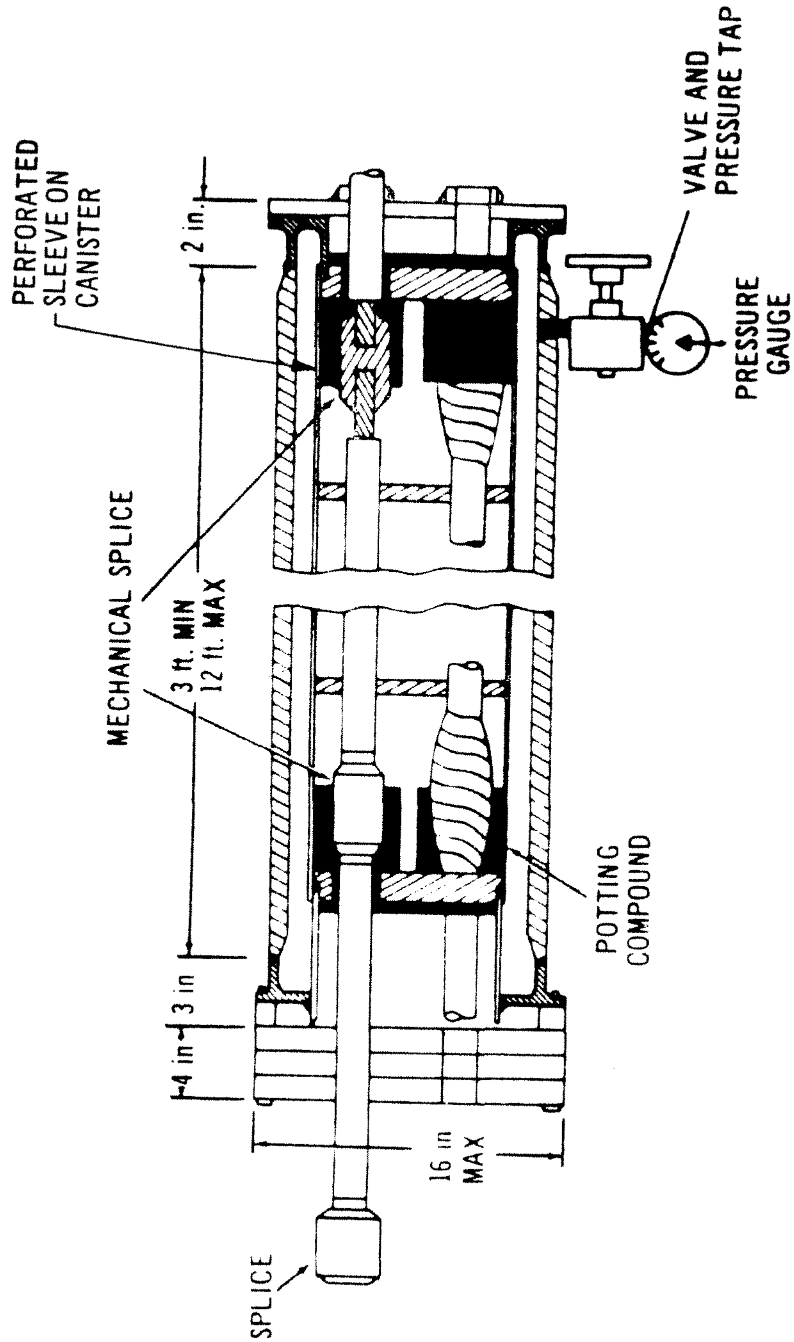
FIGURE 3.8-40
REVISION 5, JUNE 1999



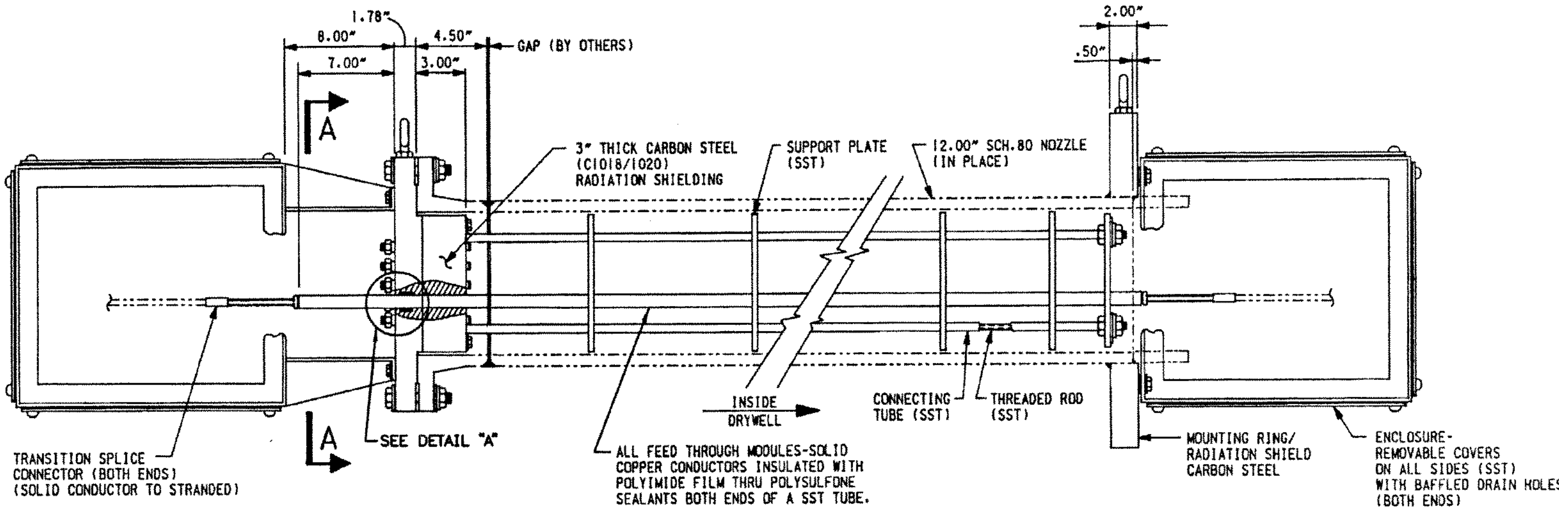
QUAD CITIES STATION
UNITS 1 & 2

SHIELDED SIGNAL CABLE PENETRATION ASSEMBLY

FIGURE 3.8-41

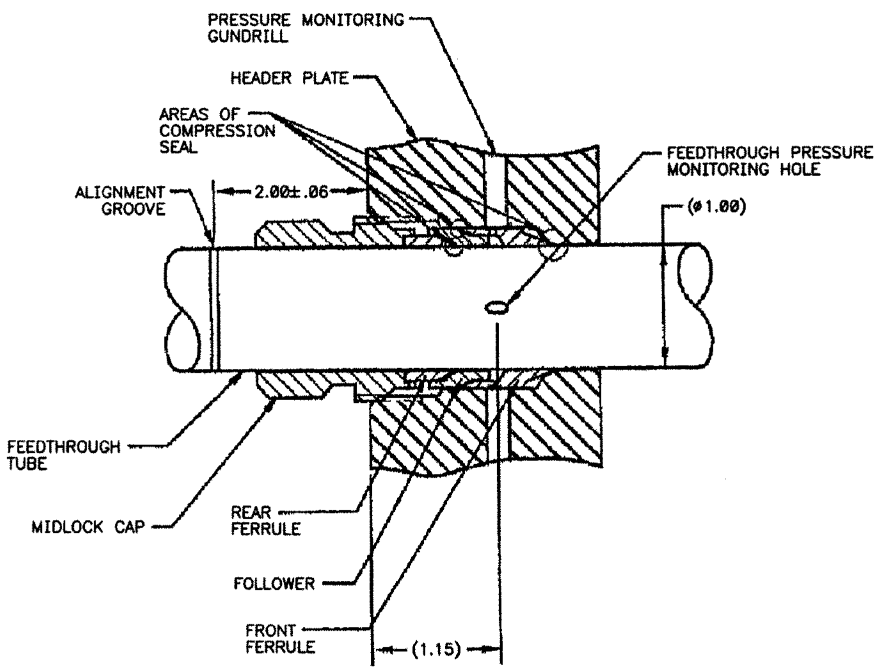


QUAD CITIES STATION UNITS 1 & 2
HIGH VOLTAGE POWER ELECTRICAL PENETRATION ASSEMBLY
FIGURE 3.8-42

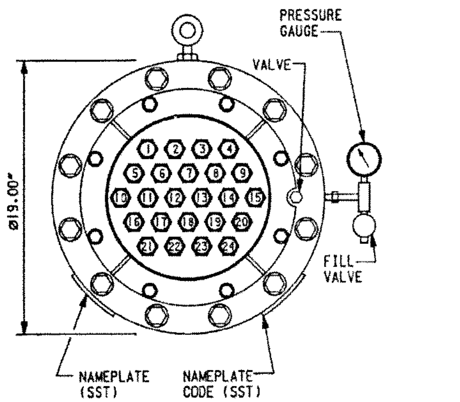


SECTIONAL DET.6

SEE CONAX CORP DWG.7BQ0-10001 FOR ADDITIONAL DETAILS.



DETAIL A

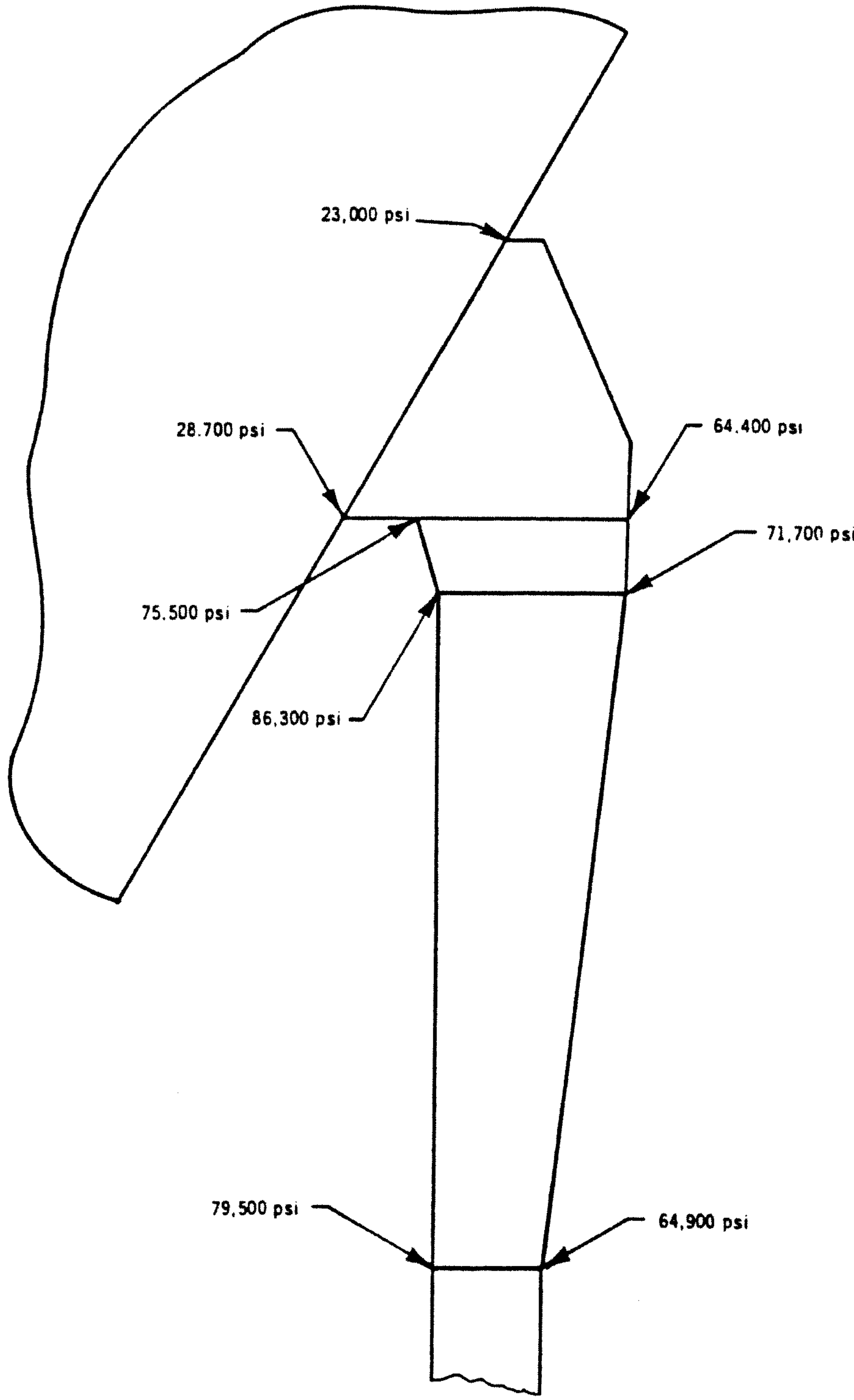


VIEW A-A

QUAD CITIES STATION
UNITS 1 & 2

GENERIC MODULAR STYLE LOW VOLTAGE POWER AND CONTROL ELECTRICAL PENETRATION ASSEMBLY

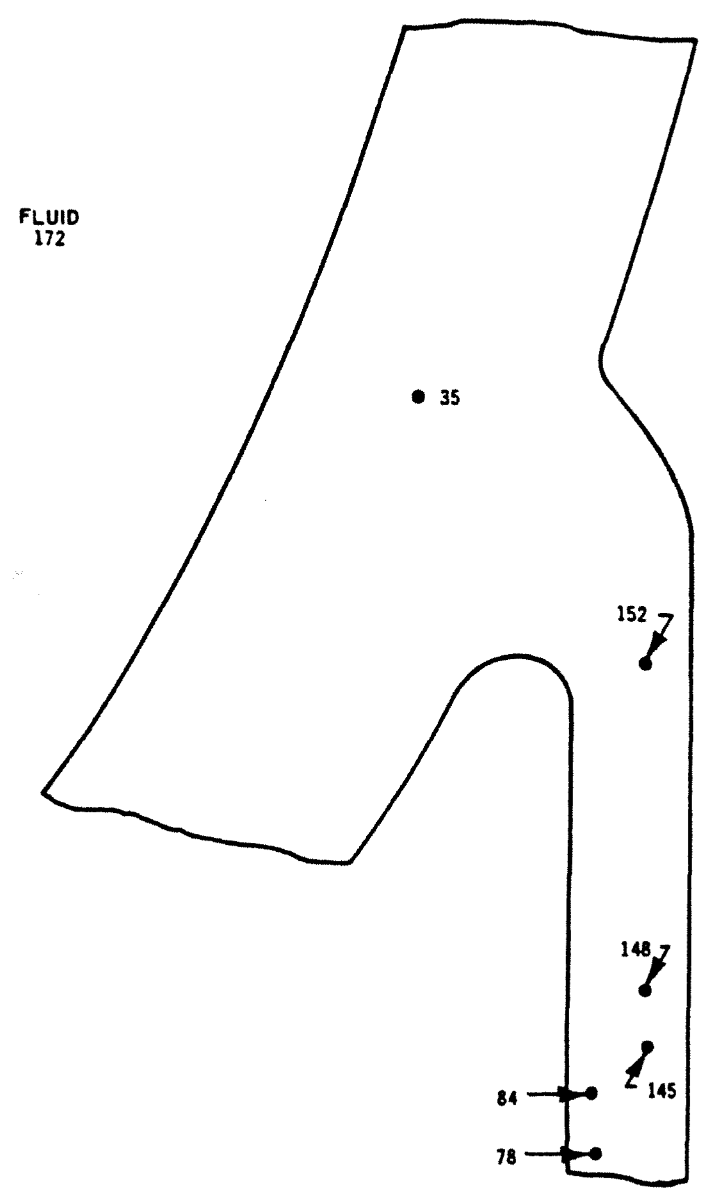
FIGURE 3.8-43
REVISION 8, OCTOBER 2005



QUAD CITIES STATION
UNITS 1 & 2

REACTOR VESSEL SUPPORT SKIRT PLUS SECONDARY STRESS INTENSITY RANGE

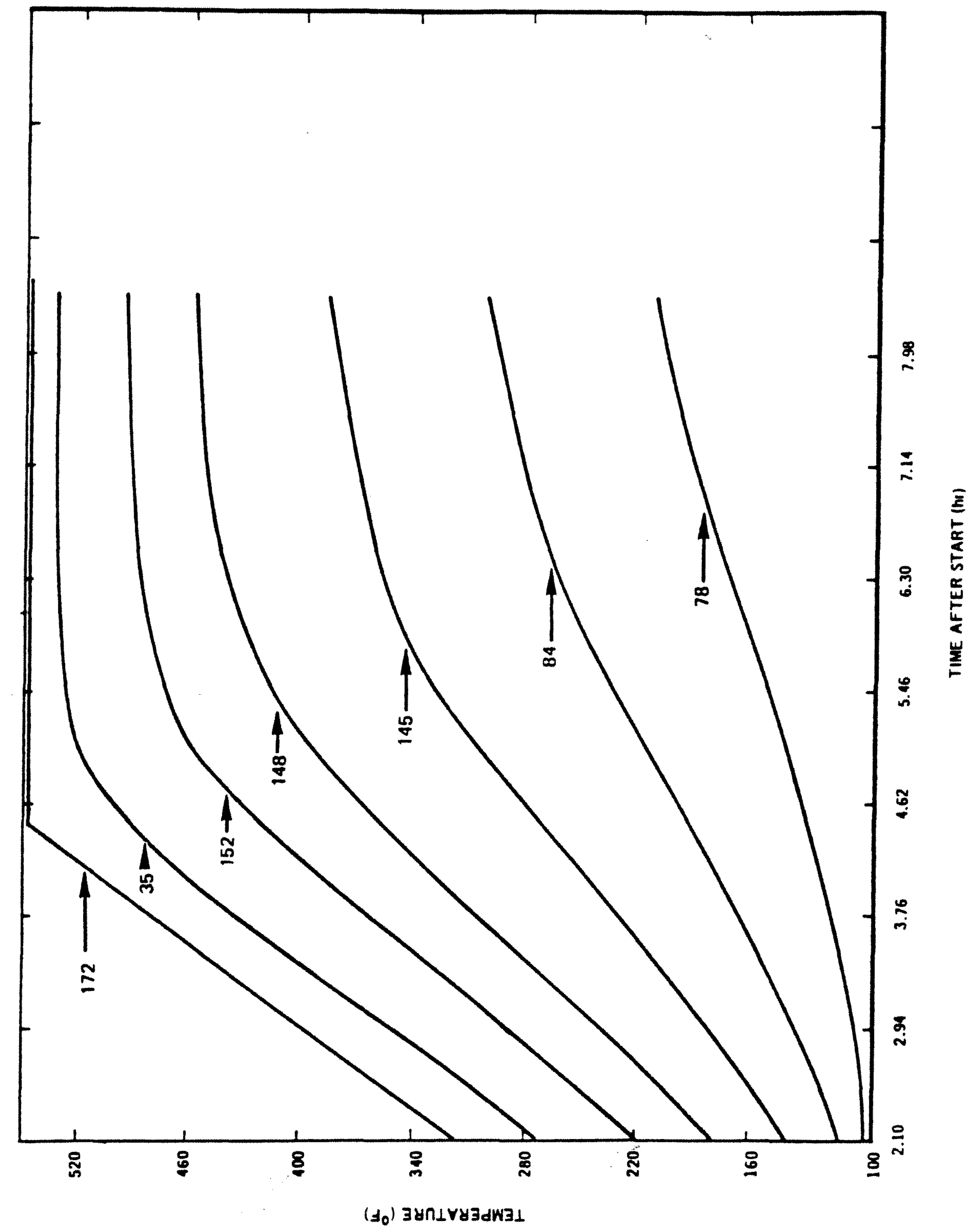
FIGURE 3.9-1



QUAD CITIES STATION
UNITS 1 & 2

REACTOR VESSEL SUPPORT SKIRT
TYPICAL POINTS FOR TEMPERATURE PLOTS

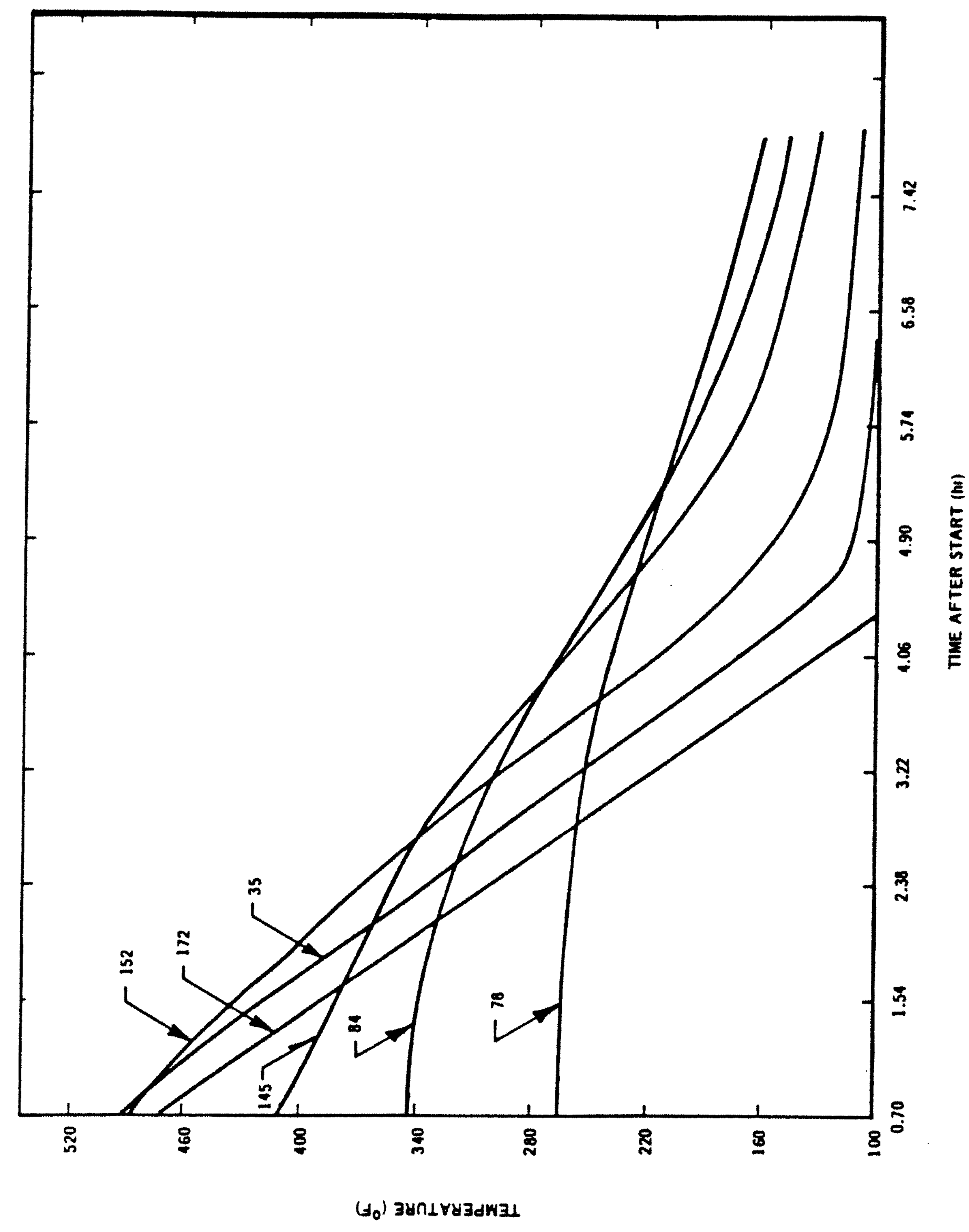
FIGURE 3.9-2



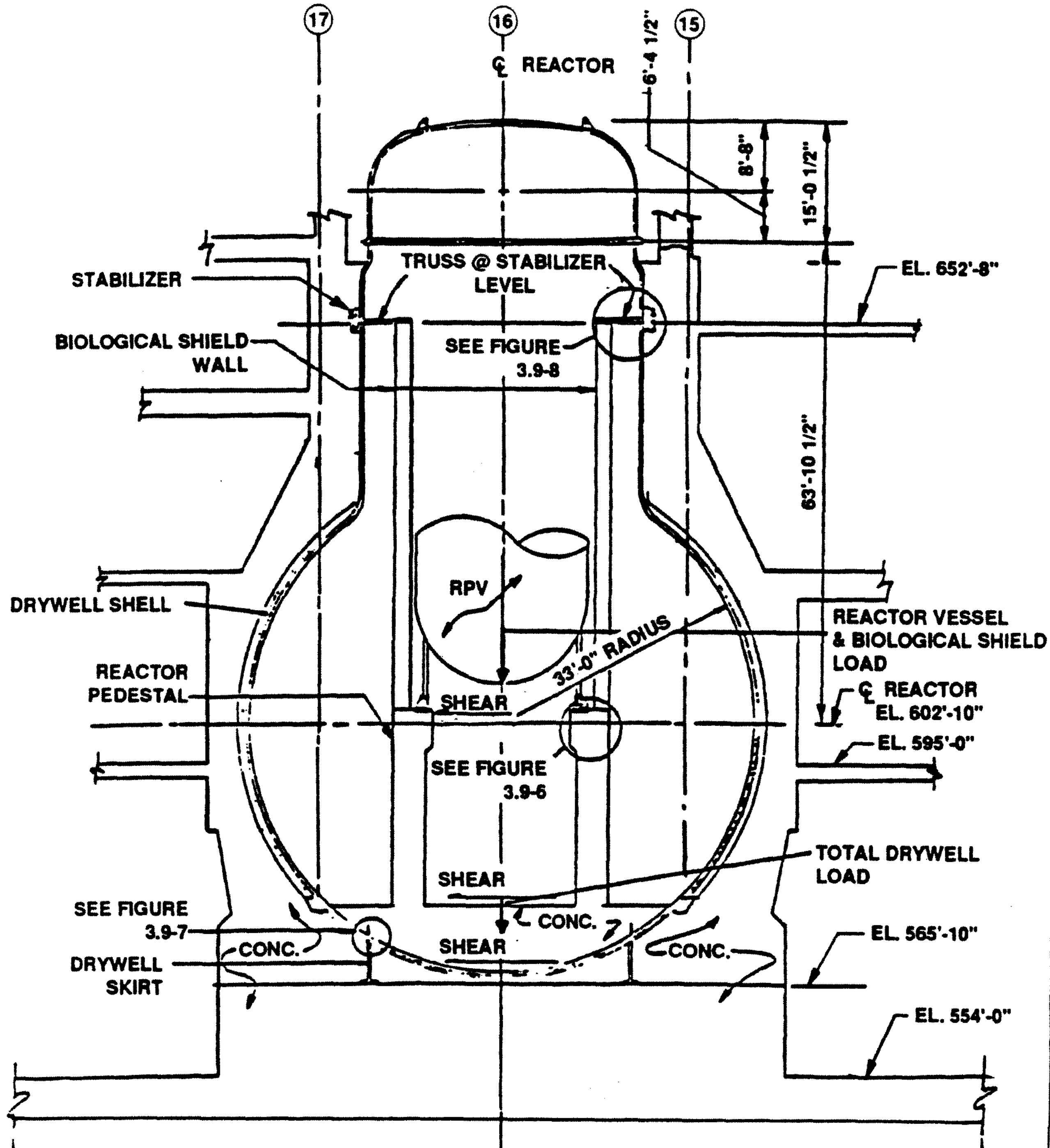
QUAD CITIES STATION
UNITS 1 & 2

REACTOR VESSEL SUPPORT SKIRT
TEMPERATURE DURING STARTUP

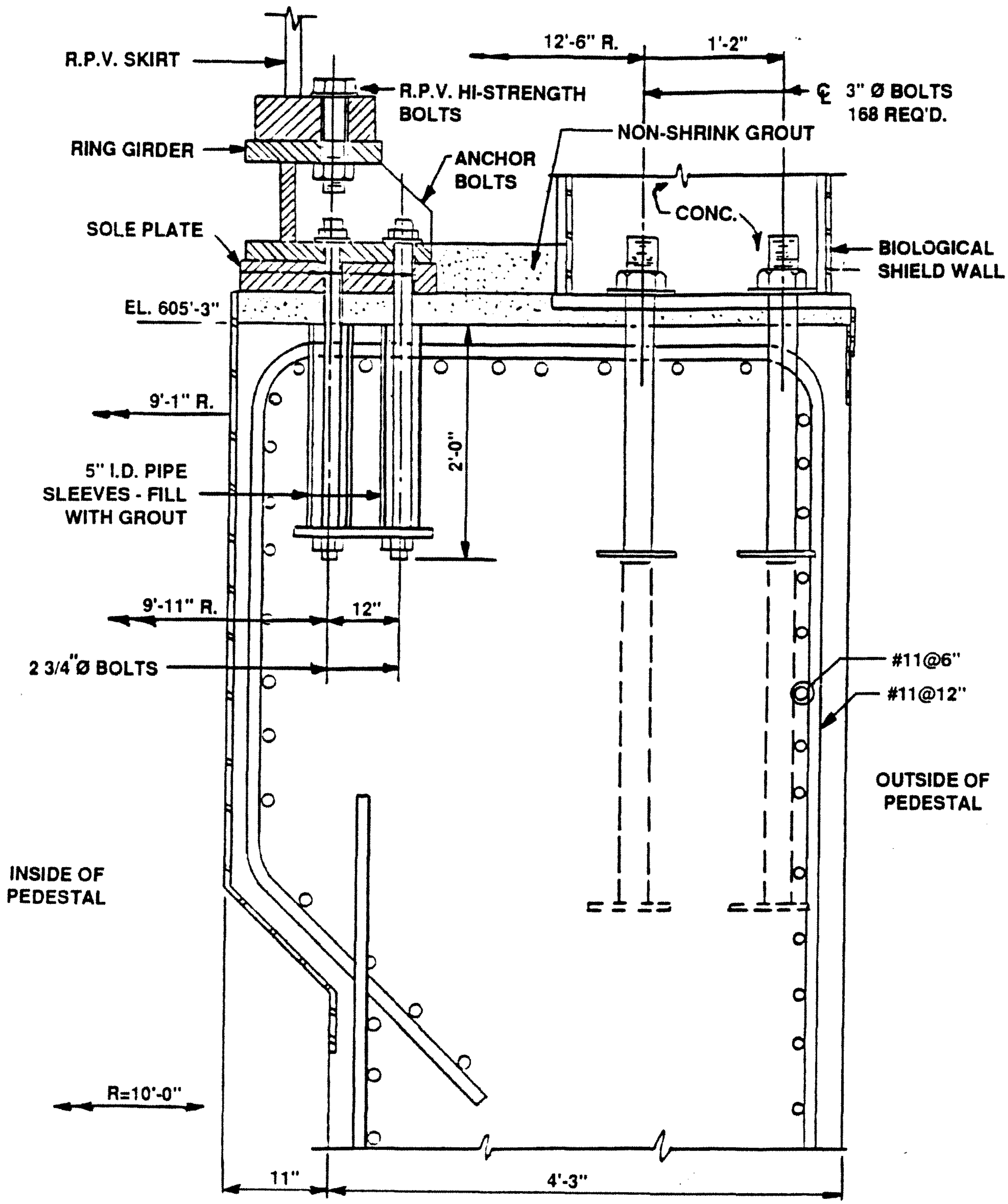
FIGURE 3.9-3



QUAD CITIES STATION UNITS 1 & 2
REACTOR VESSEL SUPPORT SKIRT TEMPERATURE DURING SHUTDOWN
FIGURE 3.9-4



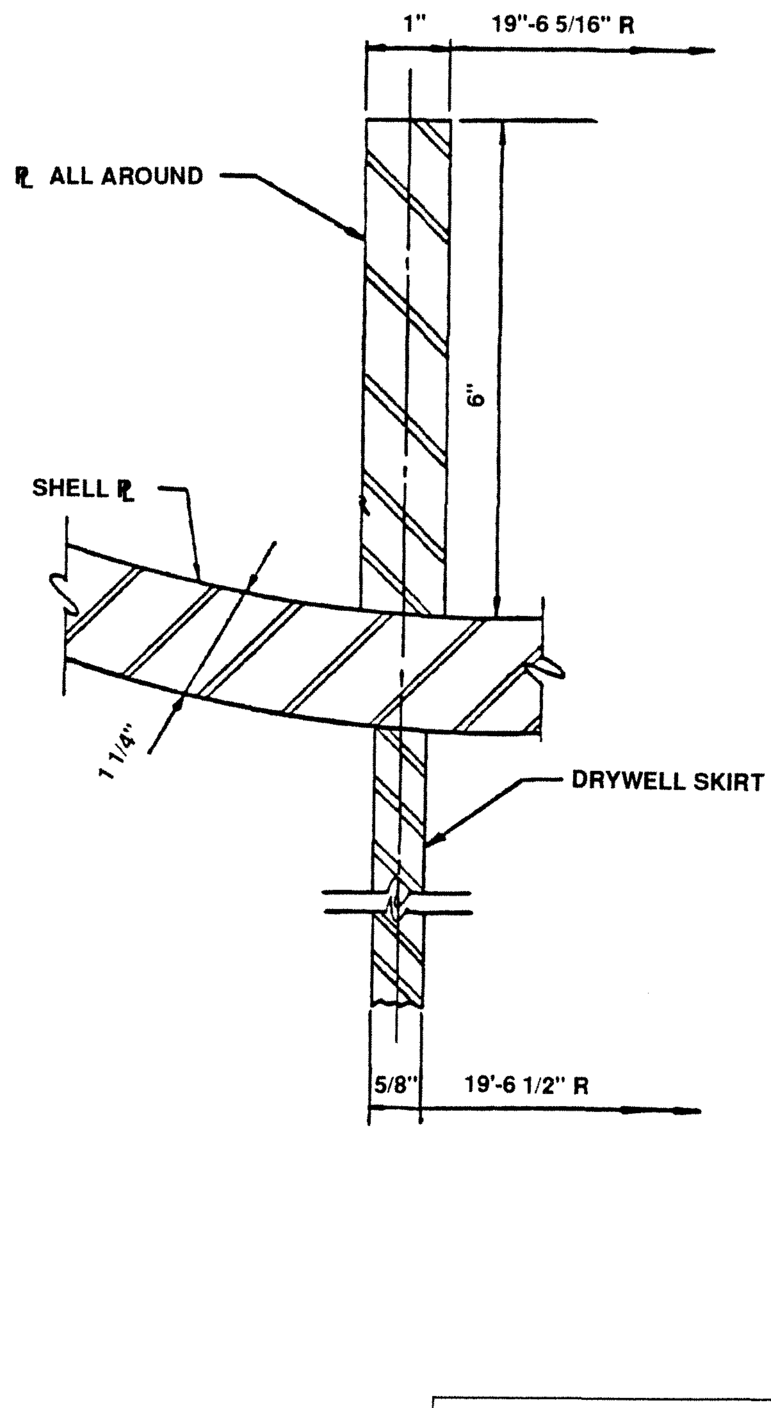
QUAD CITIES STATION UNITS 1 & 2
REACTOR & DRYWELL SUPPORTS
FIGURE 3.9-5



QUAD CITIES STATION
UNITS 1 & 2

REACTOR VESSEL SUPPORT SKIRT
ANCHORAGE DETAIL

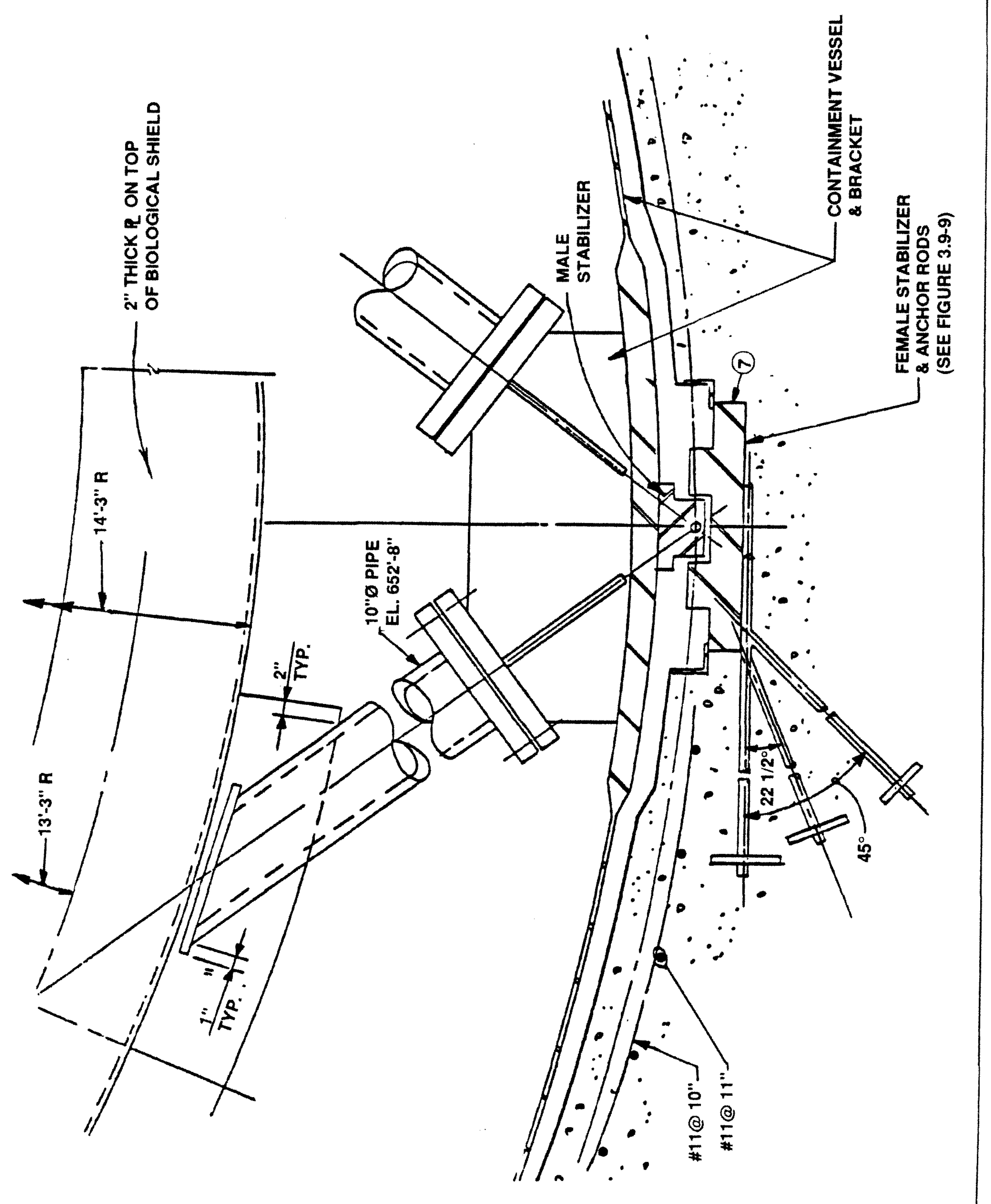
FIGURE 3.9-6



QUAD CITIES STATION
UNITS 1 & 2

DRYWELL SHELL - SKIRT DETAIL

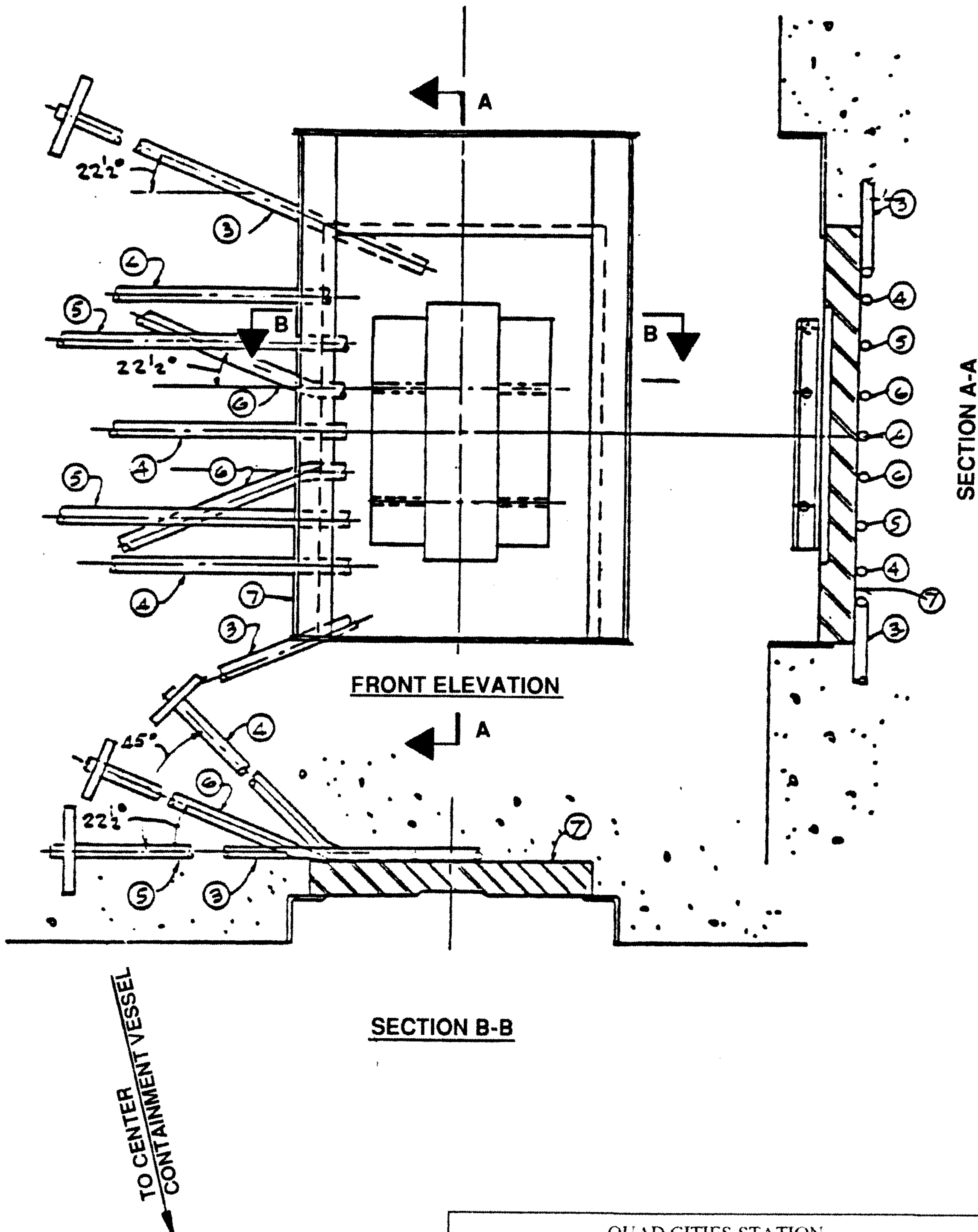
FIGURE 3.9-7



QUAD CITIES STATION
UNITS 1 & 2

RPV STABILIZER

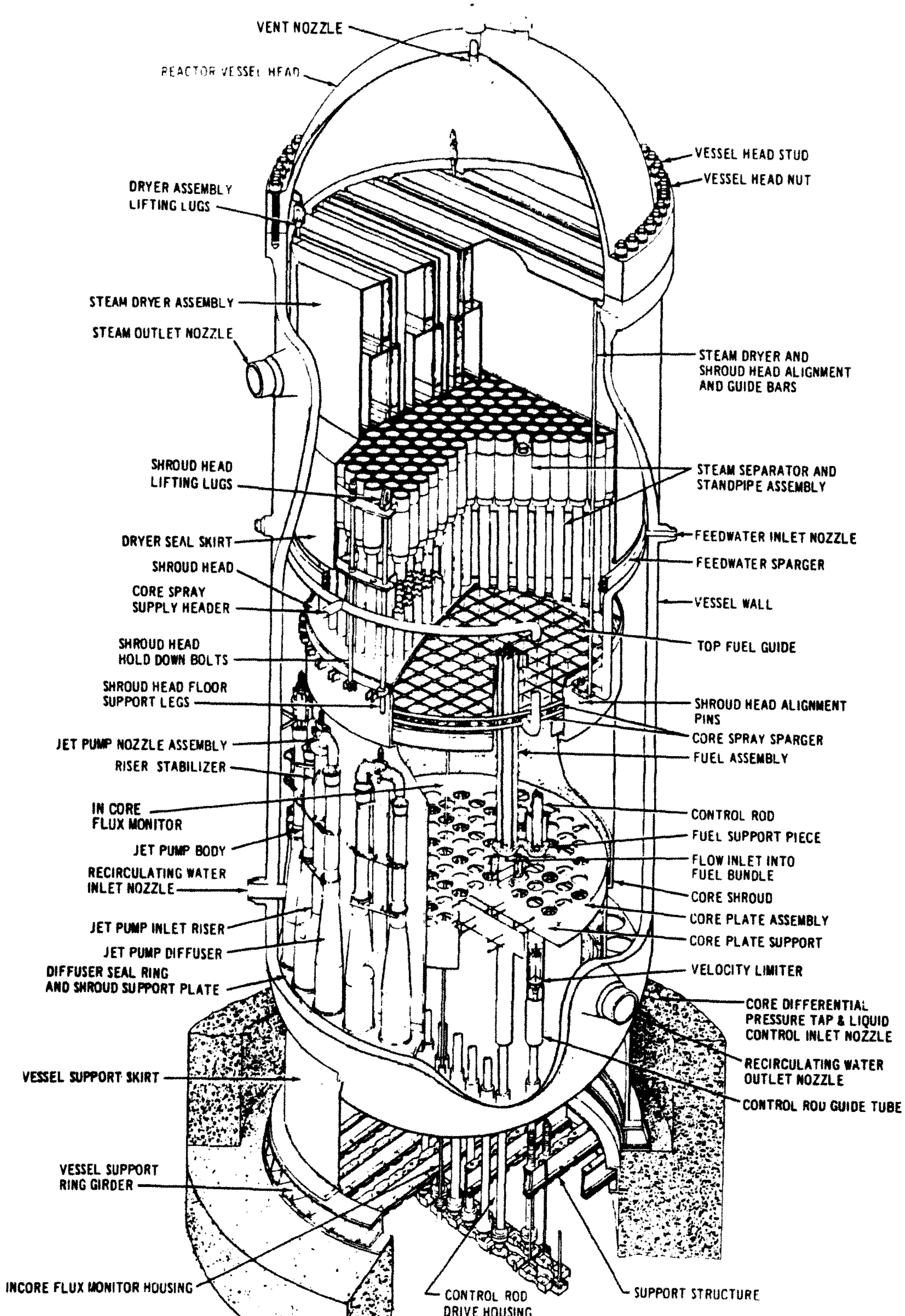
FIGURE 3.9-8



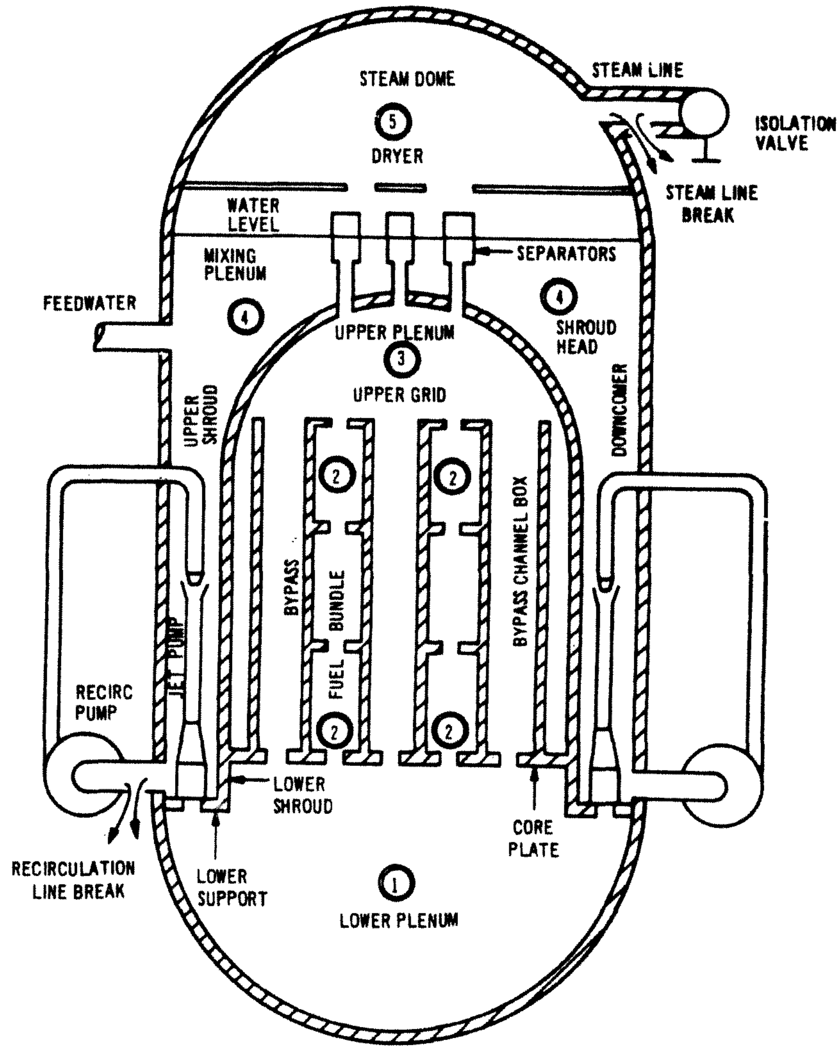
QUAD CITIES STATION
UNITS 1 & 2

RPV FEMALE STABILIZER & ANCHOR RODS

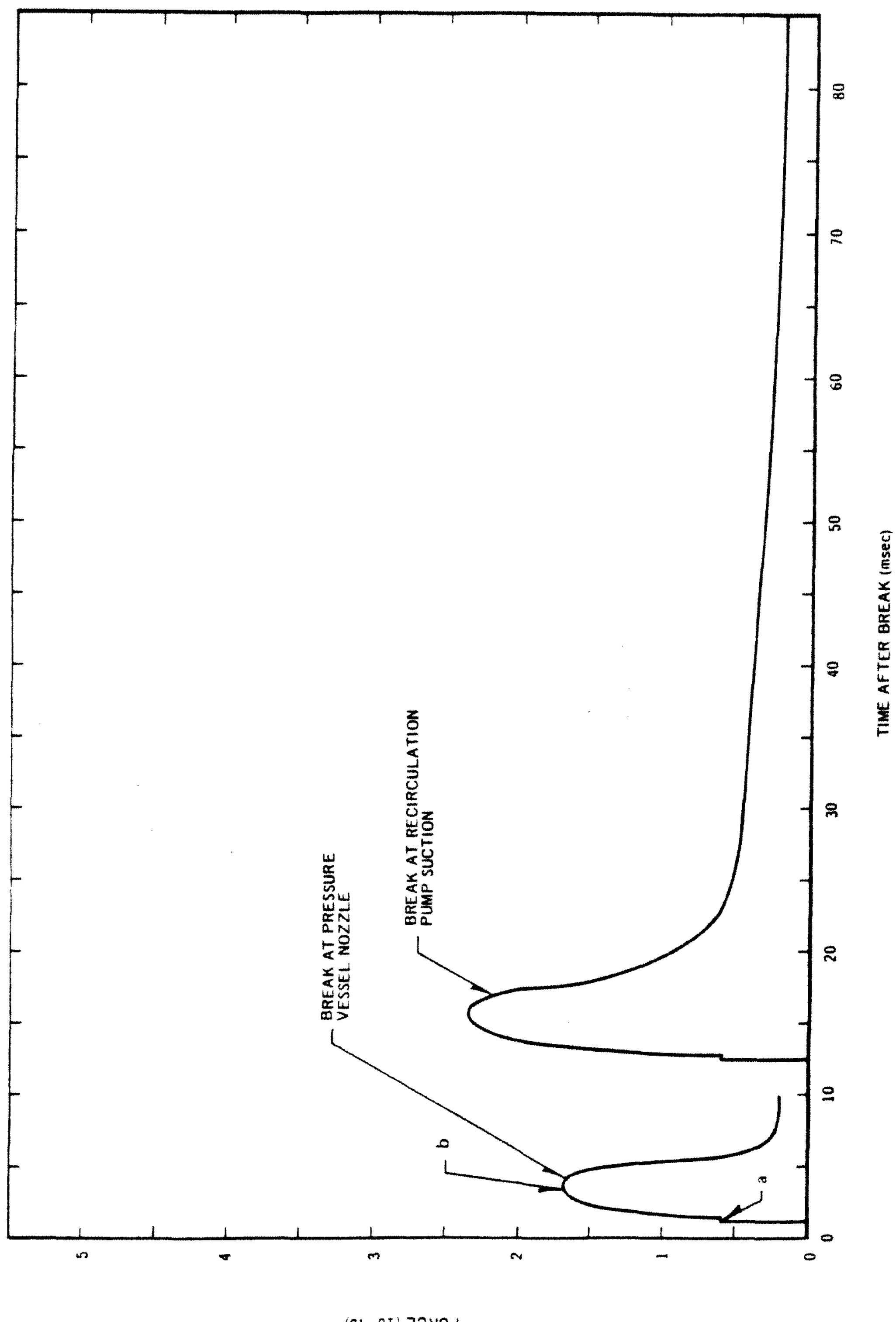
FIGURE 3.9-9



QUAD CITIES STATION UNITS 1 & 2
REACTOR VESSEL AND INTERNALS ISOMETRIC
FIGURE 3.9-10



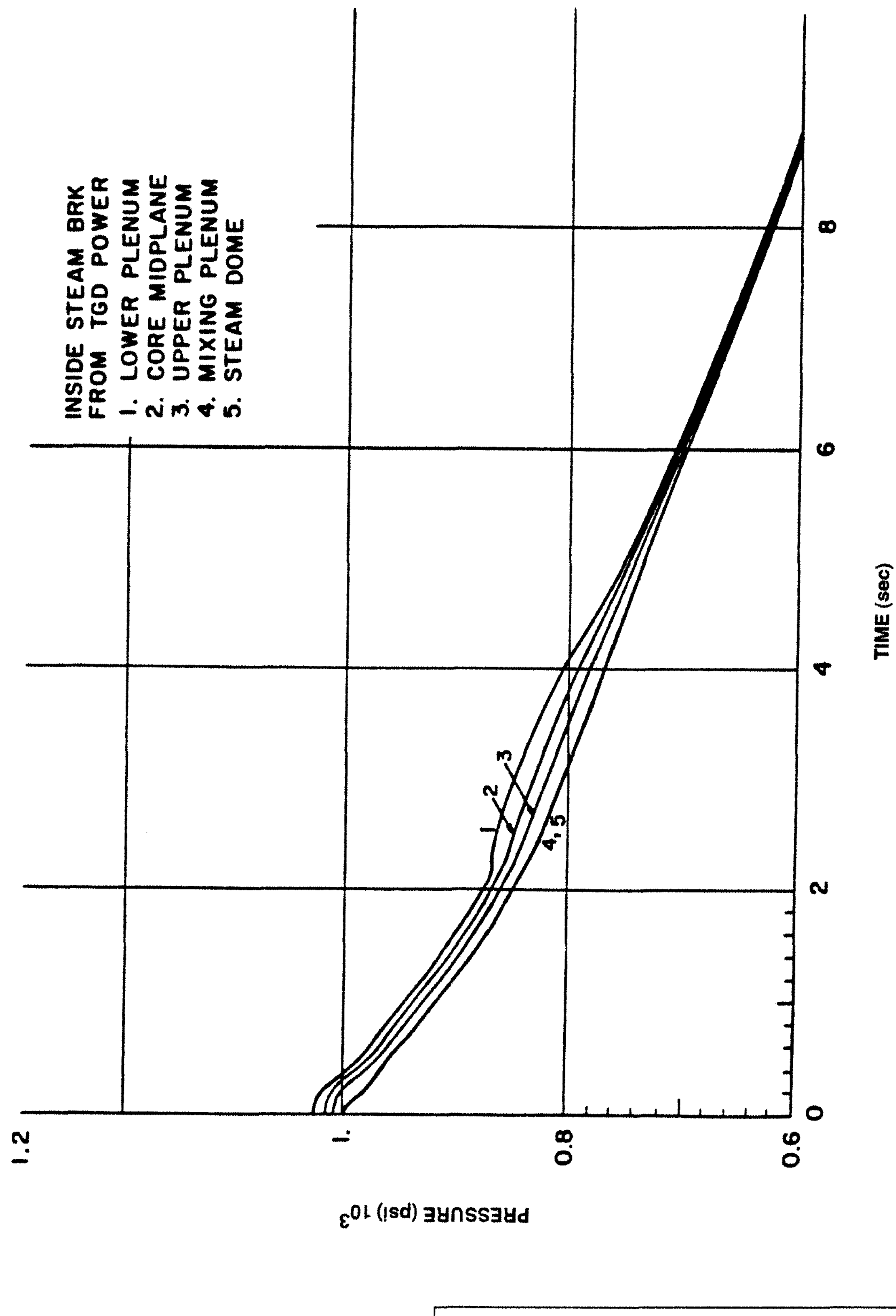
QUAD CITIES STATION UNITS 1 & 2
BWR REACTOR INTERNAL CONFIGURATION WITH JET PUMP
FIGURE 3.9-11



QUAD CITIES STATION
UNITS 1 & 2

REACTOR VESSEL SHROUD FORCE FOLLOWING RECIRCULATION LINE BREAK

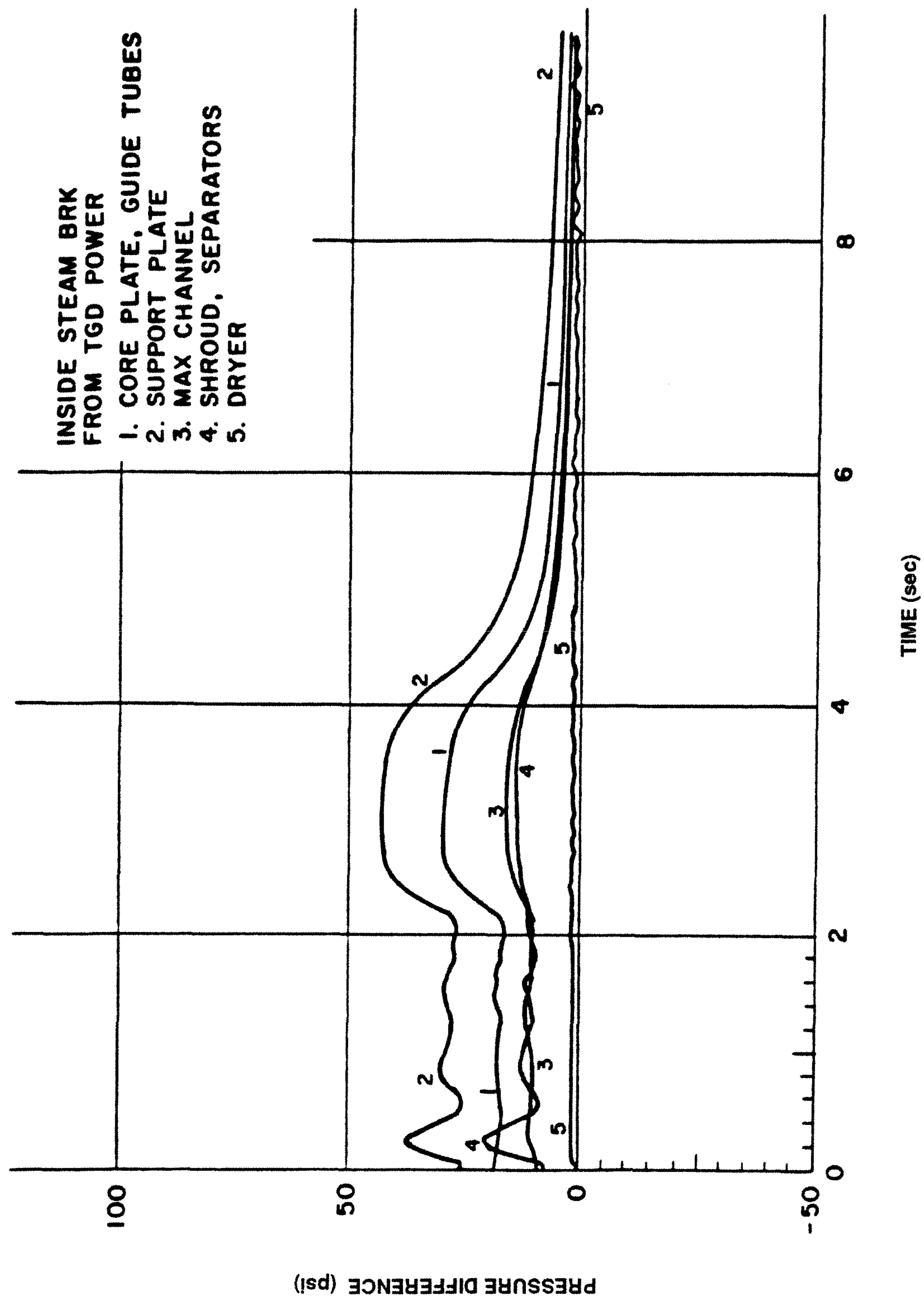
FIGURE 3.9-12



QUAD CITIES STATION
UNITS 1 & 2

BLOWDOWN TRANSIENT REACTOR PRESSURE
FROM 2511 MWT

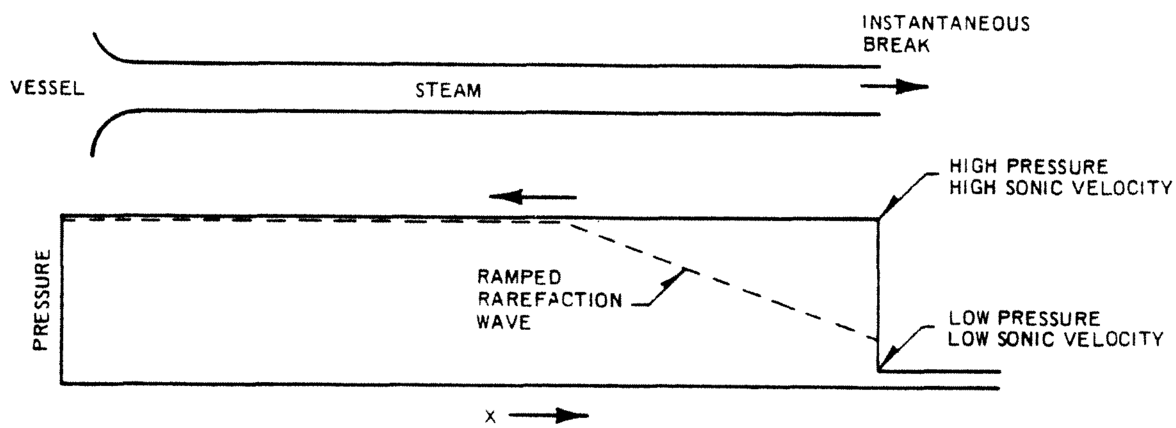
FIGURE 3.9-13



QUAD CITIES STATION
UNITS 1 & 2

BLOWDOWN TRANSIENT REACTOR DIFFERENTIAL PRESSURE FROM 2511 MWT

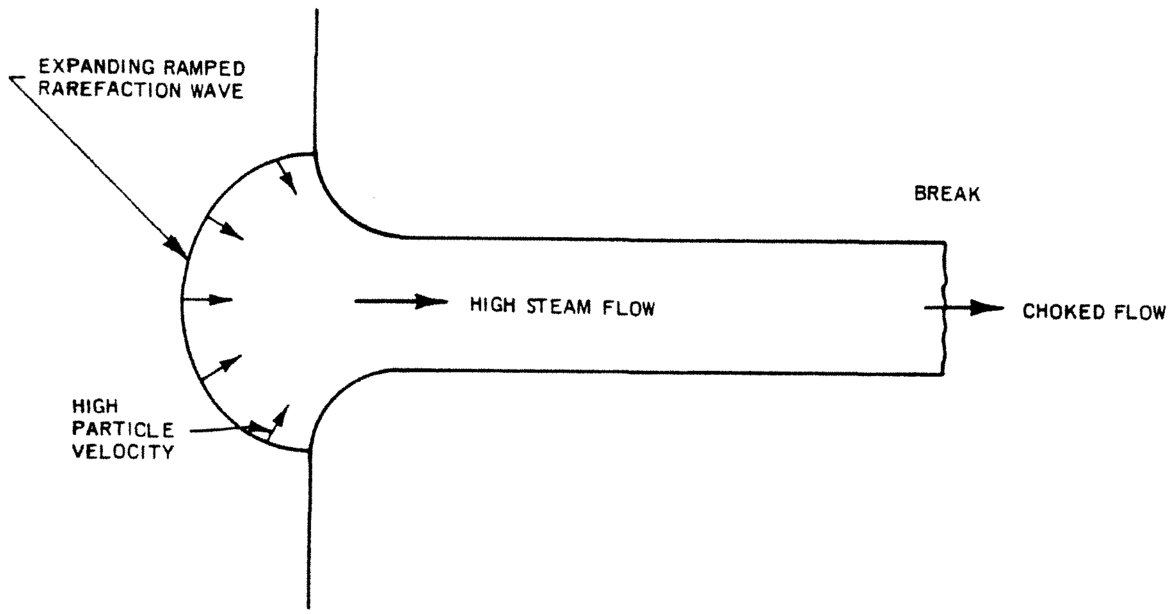
FIGURE 3.9-14



QUAD CITIES STATION
UNITS 1 & 2

COMPRESSIBLE FLUID WAVE PROPAGATION

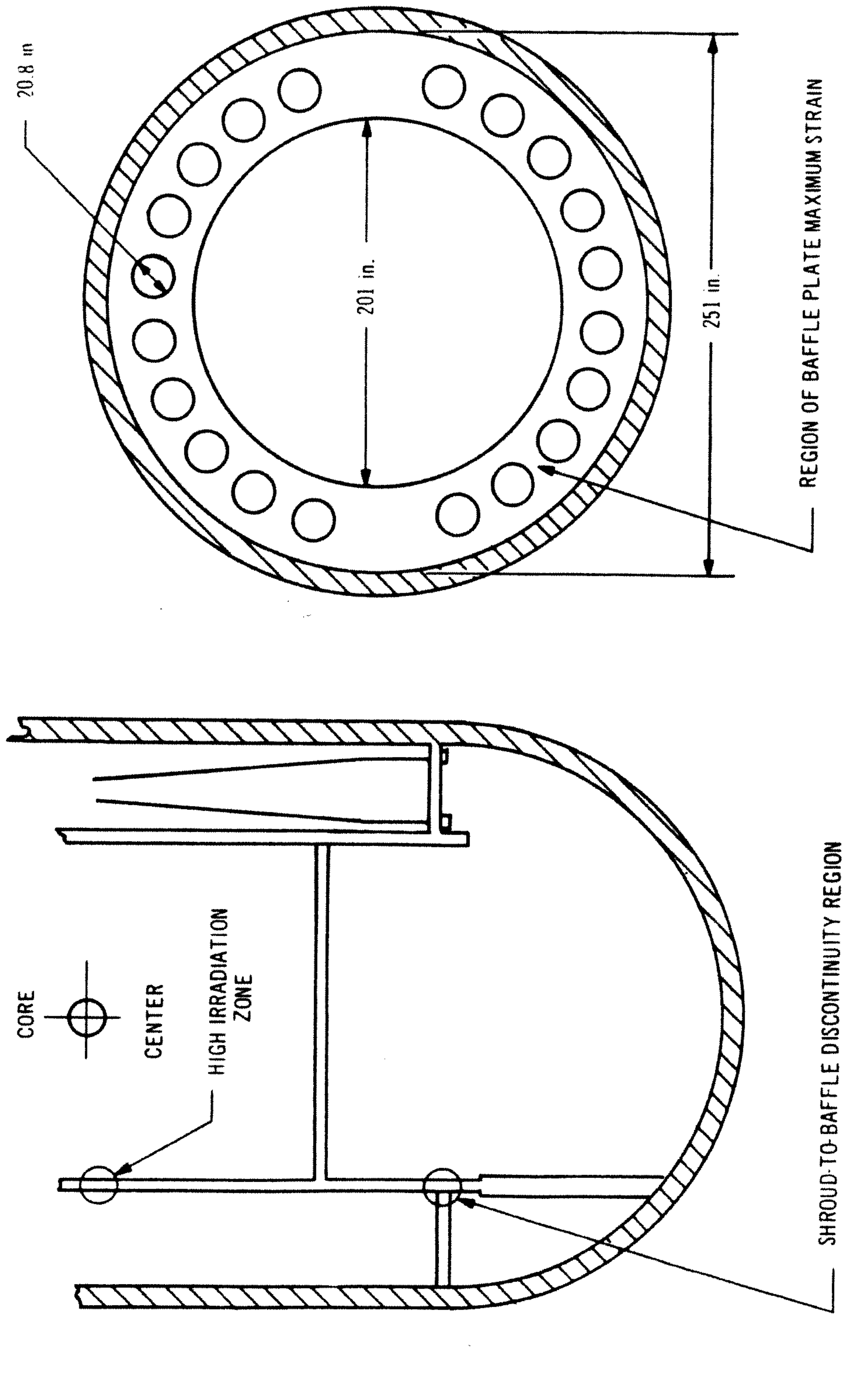
FIGURE 3.9-15



QUAD CITIES STATION
UNITS 1 & 2

COMPRESSIBLE FLUID WAVE EXPANSION

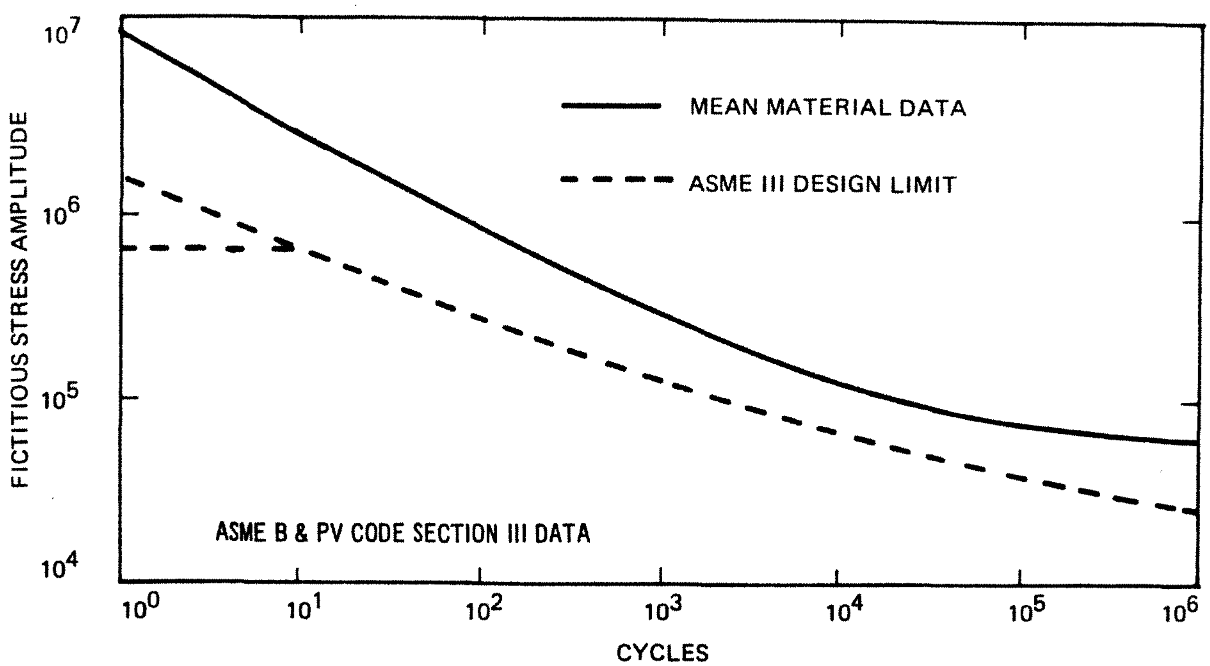
FIGURE 3.9-16



QUAD CITIES STATION
UNITS 1 & 2

REACTOR INTERNAL BAFFLE PLATE LIGAMENT
STRAIN STUDY

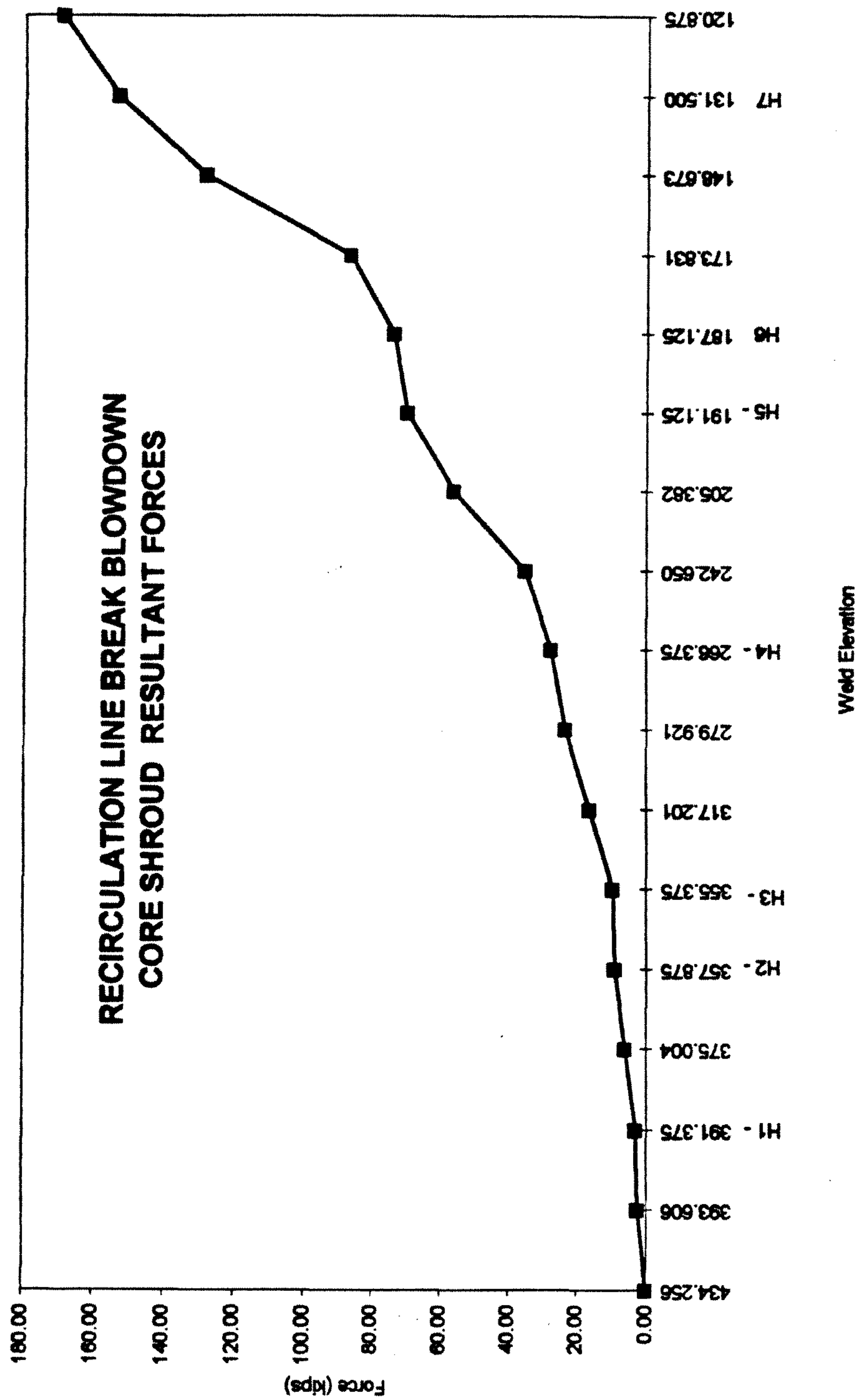
FIGURE 3.9-17



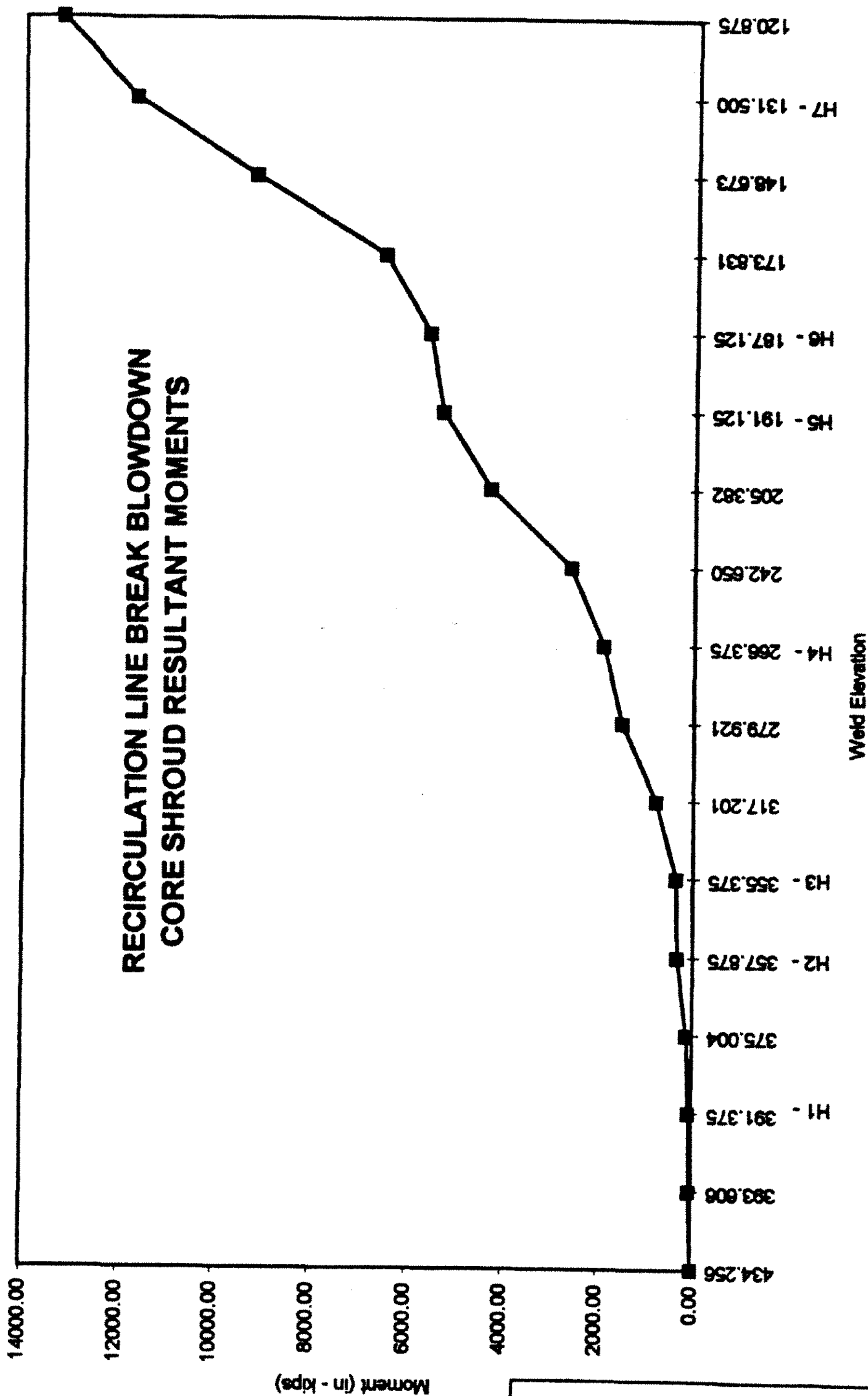
QUAD CITIES STATION
UNITS 1 & 2

FATIGUE CURVE FOR INCONEL AND STAINLESS STEEL

FIGURE 3.9-18



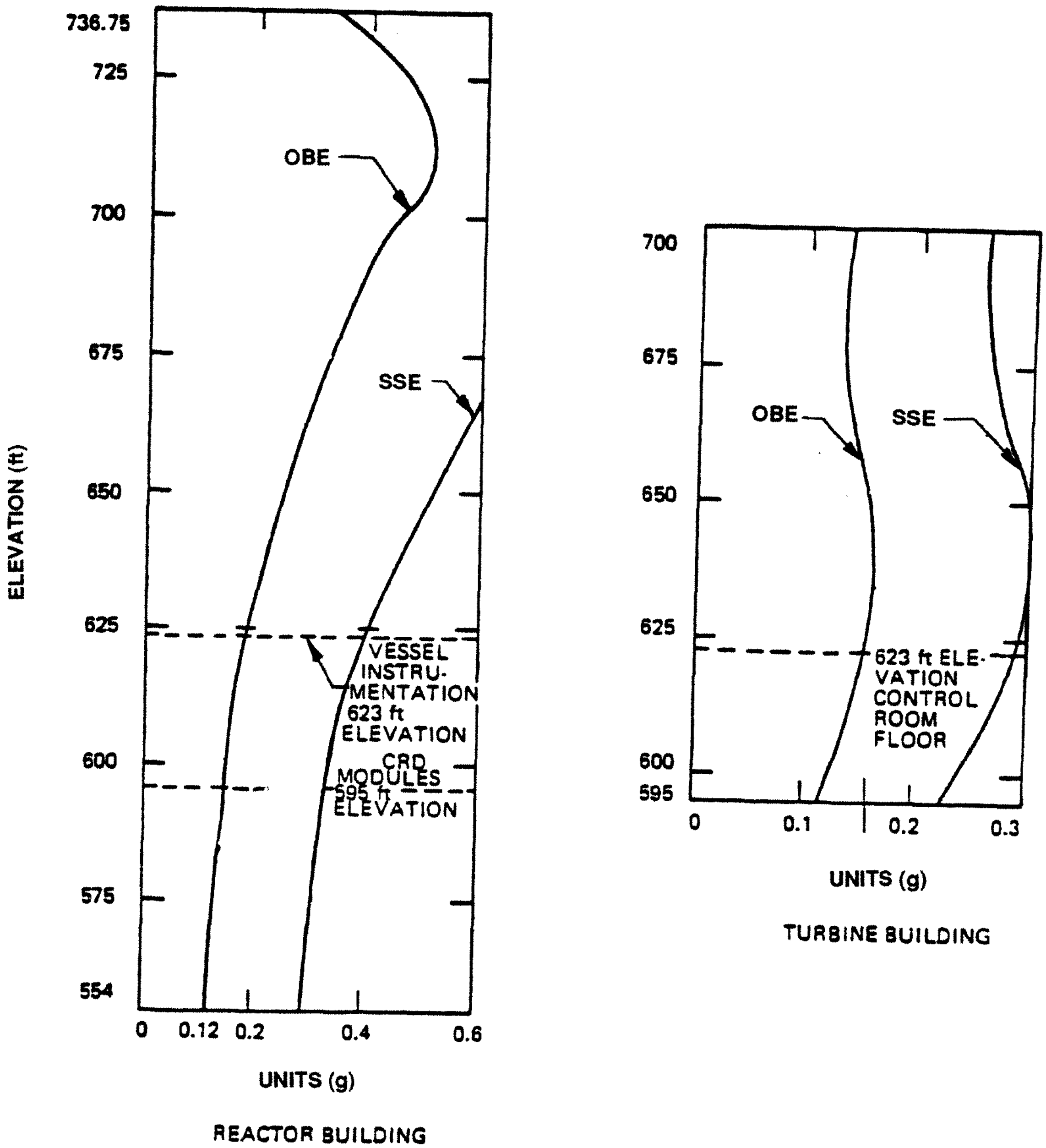
QUAD CITIES STATION UNITS 1 & 2
RECIRCULATION LINE BREAK BLOWDOWN CORE SHROUD RESULTANT FORCES
FIGURE 3.9-19 REVISION 3, DEC. 1995



QUAD CITIES STATION
UNITS 1 & 2

RECIRCULATION LINE BREAK BLOWDOWN
CORE SHROUD RESULTANT MOMENTS

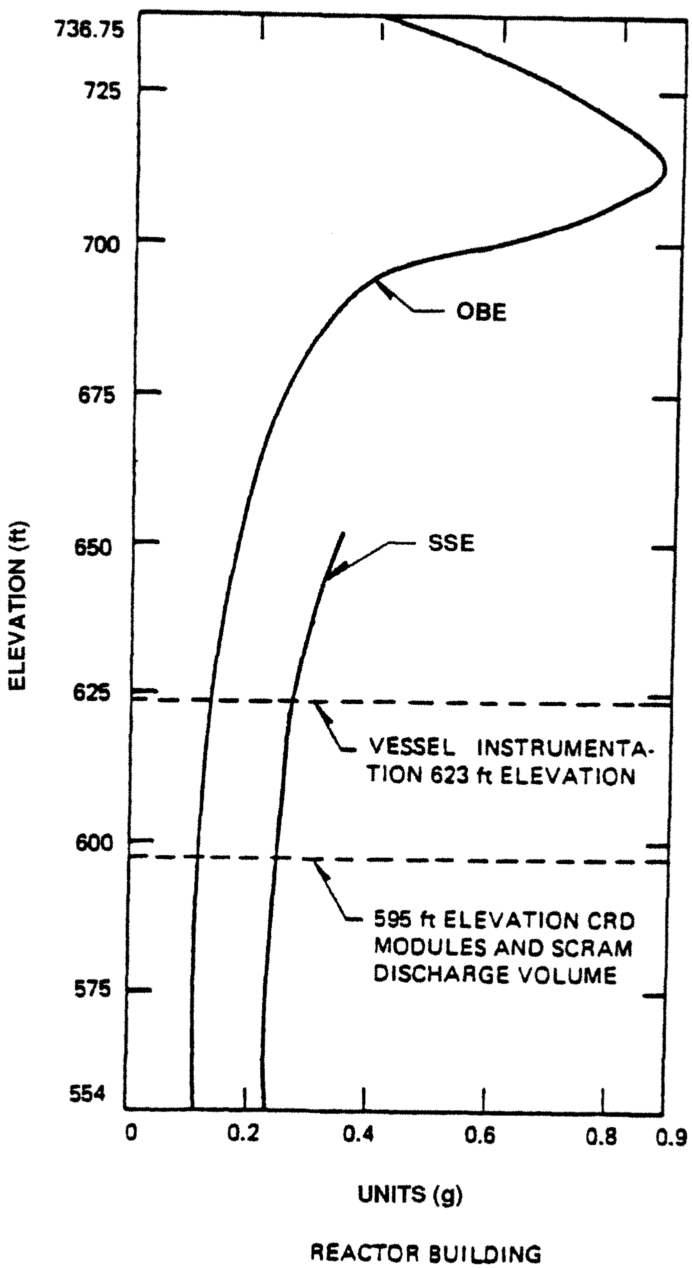
FIGURE 3.9-20
REVISION 3, DEC. 1995

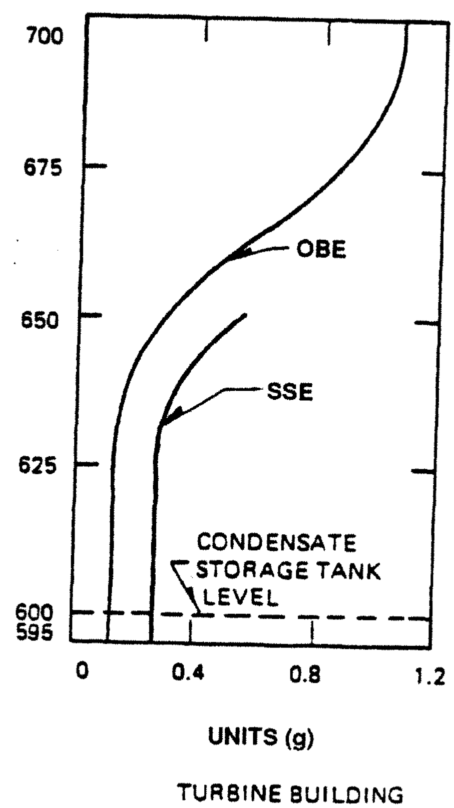


QUAD CITIES STATION
UNITS 1 & 2

REACTOR AND TURBINE BUILDINGS
ACCELERATION DIAGRAMS (N -S DIRECTION)

FIGURE 3.10-1





QUAD CITIES STATION UNITS 1 & 2
REACTOR AND TURBINE BUILDINGS ACCELERATION DIAGRAMS (E -W DIRECTION)
FIGURE 3.10-2

Information withheld in accordance with 10 CFR 2.390

QUAD CITIES STATION UNITS 1 & 2
ENVIRONMENTAL ZONE MAP (BASEMENT FLOOR PLAN) ELEVATION 554'-0"
FIGURE 3.11-1 SHEET 1 REVISION 7, JANUARY 2003

Figure 3.11-1, Sheet 1a

Zone	Normal (18)				HELB (1) (11) (23)			LOCA (5) (8) (9) (10) (12) (13)				
	Zone Temp (°F)	Press. (PSIA)	Humidity (%RH)	40-Yr Dose (Rads) (19)	Temp (°F)	Press (PSIA)	Humidity (%RH)	Temp (°F)	Press (PSIA)	Humidity (%RH)	1-Hr Dose (Rads) (2) (3) (16)	30-Day Dose (Rads) (2) (3) (16)
1 (15)	150	14.7	20-90	1.1E07	338 (MS) (14)	40	100 (C)	294	63	100 (C)	2.8E07	1.4E08
2 (20)	104	14.7	20-90	4.2E04	239 (HPCI)	15.2	100 (C)	173 (4)	14.7	100 (NC)	2.0E06	1.4E07
3	104 *	14.7	20-90	<1.0E04	110 (HPCI)	15.3	100 (C)	150 W/Room Cooler 185 W/O Room Cooler +	14.7	100 (NC)	1.4E05	2.8E06
4	104 *	14.7	20-90	<1.0E04	283 (RCIC)	15.3	100 (C)	150 W/Room Cooler 185 W/O Room Cooler +	14.7	100 (NC)	9.0E06	2.2E07
5	104 *	14.7	20-90	<1.0E04	223 (HPCI)	15	100 (C)	150 W/Room Cooler 176 W/O Room Cooler +	14.7	100 (NC)	2.8E05	5.6E06
6	104 *	14.7	20-90	<1.0E04	224 (HPCI)	15	100 (C)	150 W/Room Cooler 185 W/O Room Cooler +	14.7	100 (NC)	2.8E05	2.2E06
7	104 *	14.7	20-90	<1.0E04	230 (HPCI)	21	100 (C)	< 120 W/Room Cooler 185 W/O Room Cooler +	14.7	100 (NC)	9.0E06 (3)	2.2E07 (3)
8	125 (17)	14.7	20-90	2.2E06	120 (FW/MS)	14.7	100 (NC)	120	14.7	100 (NC)	<1.0E04	<1.0E04
8a	120	14.7	20-90	<1.0E04	200 (FW/MS)	17	100 (C)	120	14.7	100 (NC)	<1.0E04	<1.0E04
8b	120	14.7	20-90	1.2E07	120 (FW/MS)	14.7	100 (NC)	120	14.7	100 (NC)	<1.0E04	<1.0E04

(C) Condensing
(NC) Non-Condensing
(*) Temperature is 150°F (max) when the ECCS equipment in the room is operating.
(+) Temperature W/O room cooler is a pre-EPU value. Equipment is not qualified for this condition.

Figure 3.11-1, Sheet 1b

General Note:

Unless otherwise noted, refer to reference 3 for zone descriptions, parameter details, and associated references.

Notes:

1) For time/temperature profile during a HELB, refer to the following:

   A) HPCI line break outside drywell, see Reference 14.
   B) Main steam (MS) line break inside drywell, see Reference 5.
   C) Main steam (MS) line break outside drywell, see Reference 11.
   D) RWCU steam line break outside drywell, see Reference 15.
   E) RCIC line break in RCIC room/Torus room, see Reference 30.
   F) HELB evaluation for EPU conditions, see Reference 48.

2) For component unique radiation doses, refer to Reference 2 (pre-EPU); the factors given in References 35 and 45 must be utilized in conjunction with Reference 2 to obtain EPU values.
3) For components in the HPCI system, dose is <1.0E04 rads (pre-EPU). Reference 35 determines that the EPU value remains the same.
4) For pre-EPU time/temperature profile, in Torus area, see Reference 29. EPU time/temperature profile, see Reference 34. Reference 33 applies the profile.
5) For composite MSLB/LOCA temperature profile inside drywell, for pre-EPU, see Reference 1 Appendix C; EPU composite profile (Reference 32) was obtained by adding 4°F to the first 24 hours and 18°F to remaining profile from Reference 1.
6) Maximum normal temperature during a LOCA in the opposite unit is: See References 28 and 33.

   1) 116°F in Zones 11-17, 21, 24 & 27
   2) 110°F in Zones 29-34, 37, 38 & 40
   3) 150°F in Zone 22

7) For pre-EPU time/temperature profile in reactor building areas see Reference 38. EPU time/temperature profile for affected reactor building areas were conservatively obtained by adding EPU temperature increase in this area from Reference 34 to the profile from Reference 28. Reference 33 applies the profile.
8) One-year total integrated radiation dose inside drywell (Zones 1, 9, 20, 28, 36) is 1.76E08 rads (References 2 and 45). The one-year post-LOCA dose for outside the drywell area depends on the components distance from the pipe, pipe size, and circulated medium; the factors given in References 35 and 45 must be utilized in conjunction with Reference 2 to obtain EPU values.
9) Components inside the drywell (Zones 1, 9, 20, 28, 36) are exposed to demineralized water spray during LOCA/post-LOCA. See Reference 1.
10) Inside drywell flood elevation is 583'-0". See Reference 1.
11) For components located within compartments, the radiation is <5.0E04 rads during a postulated HELB. (Pre-EPU), See Reference 1; Reference 35 determines that the EPU value remains the same.
12) HPCI room temp is less than 120°F for a LOCA. See References 31 and 33.
13) Peak LOCA temperature and pressure is based on FSAR Figures 5.2-15 and 5.2-16. The peak LOCA temperature and pressure is bounded for EPU by this value, per Reference 32.
14) Main steam (MS) line break inside drywell (EPU), see Reference 32.
15) Equipment qualification dose to cables inside containment (EPU), see Reference 46.
16) For zone radiation doses (EPU), see References 35 and 45.
17) Normal temperature increase due to EPU, see Reference 34.
18) Normal temperatures were reviewed for EPU. See Reference 34.
19) 40 years normal radiation doses for EPU conditions, see References 35 and 45.
20) Post-LOCA environmental gamma dose to transmitter LT1-1641-5B in Zone 2, see Reference 47.
21) 150°F is the bulk average temperature of the drywell. The maximum measured local temperature around the ERVs is 184°F, see Reference 49.
22) Equipment located inside the Drywell enclosed by metal will experience a maximum post-LOCA dose of 36.495 MRADs 48 hours post-accident and 48.384 MRADs 30 days post-accident. These values were calculated in Reference 49.
23) MS-FW HELB pressure spike lasts less than 10 seconds (Reference 11).
24) Main Steam Tunnel communicates with the 1st floor of the Reactor Building. However, the resulting environmental conditions are bounded by the HPCI line break parameters. These conditions have been considered for the equipment required to mitigate a Main Steam Line Break.
25) Equipment specific calculation (Reference 51) used to qualify some equipment in zones 15, 21, and 24.

Figure 3.11-1, Sheet 1c

References:

1) Response to IE Bulletin 79-01B, Rev. 6, June 1986.
2) Bechtel Spec 13524-069-N201, Rev. 1, dated 4/10/85.
3) Bechtel Spec 13524-069-N202, Rev. 7, dated 11/14/95.
4) GE Report NEDO-24217.
5) GE Report NSEO-52-0682.
6) UFSAR Section 9.4, 6.2.1.3.2 and Figures 6.2-16 and 6.2-17.
7) NFS Calculation RSA-Q-90-02.
8) Bechtel Calculation QC-716-M-001
9) Bechtel Calculation QC-716-M-005
10) CECO Chron #204884
11) S&L Calculation 3C2-0181-001
12) Bechtel Letter BCAE-93-0095
13) Bechtel Calculation NUC-40
14) Bechtel Calculation QC-030-M-001
15) Bechtel Calculation QC-030-M-002
16) Bechtel Calculation QC-733-M-001
17) Bechtel Calculation NUC-1
18) Bechtel Calculation NUC-12
19) Bechtel Calculation NUC-13
20) Bechtel Calculation NUC-15
21) Bechtel Calculation NUC-16
22) Bechtel Calculation NUC-18
23) Bechtel Calculation NUC-19
24) Bechtel Calculation NUC-25
25) Bechtel Calculation NUC-28
26) Bechtel Calculation NUC-36
27) Bechtel Calculation NUC-50
28) Calculation QDC-0020-M-0551, Rev. 00
29) NDIT No. QDC-98-181
30) NFS Calculation BSA-Q-95-09
31) Calculation QDC-2300-M-0700
32) GE Task Report No. T0400 GE-NE-A22-00103-08-01, Revision 1, Class 3, December 2000, "Containment System Response"
33) Calculation No. QDC-0000-E-1005, Rev. 0, "Evaluation of Quad Cities Station EQ Binders for Extended Power Uprate Environmental Conditions"
34) GE Task Report No. T0610 (GE-NE-A22-00103-55-01, Revision 1, Class 3, August 2001, "Power-Dependant HVAC"
35) Calculation No. QDC-000-N-1023, Revision 0, "Impact of Power Uprate on Radiation Environment in the EQ Zones," dated 03/15/01
36) Report No. CE-12, "Iodine Activity on the SGTS Filter Following a Postulated Loss of Coolant Accident," Revision 1
37) Report No. CE-13, "SGTS Filter Loading, Post-LOCA," Revision 1
38) Report No. CE-14, "SGTS Dose Rate from Purging of Containment Due to Hydrogen Generation"
39) Report No. CE-15, "SGTS Dose Rate from Purging of Containment Due to Hydrogen Generation, Post-LOCA," Revision 1
40) Calculation No. NUC-17, "Radiation Dose Values for Local Distribution Panel," Revision 1
41) Calculation No. NUC-22, "Dose from Standby Gas Treatment System Filter Based Energy Inc.," Revision 1
42) Calculation No. NUC-51, "Beta Radiation Dose Reduction," Revision 1
43) Calculation No. SNUMB-7007S, "Radiation Source Term Information for NUREG-0578 Implementation"
44) Bechtel Specification No. 13524-069-N001, "Environmental Qualification Specification for Electrical Equipment in Response to IE Bulletin 79-01B/10CFR 50.49 for Quad Cities Nuclear Power Station Units 1 and 2," Revision 00
45) Calculation No. QDC-000-N-1022, Revision 0, "Dose and Dose Rate Scaling Factors to Evaluate Impact of Power Uprate on Radiological Equipment Qualification and Vital Access," dated 03/15/01
46) Calculation No. QDC-0000-N-1070, Revision 0, "Equipment Qualification Dose to Cable Inside Containment Following Extended Power Uprate"
47) Calculation No. QDC-1600-N-1116, Revision 0, "Post-LOCA Environmental Gamma Dose to Transmitter LT1-1641-5B Located in the Torus Compartment (Zone 2) Following Extended Power Uprate," dated 05/08/01
48) GE Task Report No. T 1009, GE-NE-A22-00103-73-01, Rev. 0, "HELB Sub-Compartment Evaluation," dated October 2000
49) Calculation QDC-1600-N-1483, Revision 0, Quad Cities Drywell EQ Dose Assessment
50) Engineering Change Request 384146, "Request for Review of PBI 0559 Mode Restriction Support Pre-Outage DEHC Work by Shaw"
51) Calculation No. FAI/11-465, Rev. 0, "Quad Cities Post-LOCA Room Temperature"

Figure 3.11-1, Sheet 1d

HELB Barrier Information

Block walls and floor plugs between the Turbine Cavity and the Turbine Building are assumed to be intact during a postulated Feed Water and Main Steam Line Break, (Reference 11).

The hinged checker plates between the Main Steam Tunnel and the D-Heater Bay are assumed to be open during a postulated Feed Water and Main Steam Line Break, (Reference 11).

The hinged checker plates between the Turbine Cavity and the Turbine Building are assumed to be open during a postulated Feed Water and Main Steam Line Break, (Reference 11).

A breach of the Reactor Building wall is not analyzed in this document or its references. A breach of this barrier is assumed to be governed by entry into a Technical Specification LCO, (Reference 50).

Plant Humidity Levels, (Reference 3)

OPEN PLANT AREAS:

Plant Area	**Normal**	**Spurious**	**(Max)**
1. No HELB No moisture source Normal ventilation With or without equipment heat source	20-90	5-95	95
2. No HELB Moisture source Normal ventilation Equipment heat source	20-90	5-95	95
3. No HELB Moisture source Normal ventilation No equipment heat source	20-90	5-100	100(NC)
4. HELB Moisture source Normal ventilation No equipment heat source	20-90	5-100	100(NC)
5. HELB Moisture source Normal ventilation Equipment heat source	20-90	5-95	95

CONFINED PLANT AREAS:

Plant Area	**Normal**	**Spurious**	**(Max)**
1. No HELB No moisture source Well ventilated (comfort controlled) Equipment heat source	40-70	20-90	90
2. No HELB No moisture source Normal ventilation With or without equipment heat source	20-90	5-95	95
3. No HELB Moisture source Normal ventilation Equipment heat source	20-90	5-95	95
4. No HELB Moisture source Normal or minimal ventilation No equipment heat source	20-90	5-100	100(NC)
5. HELB Moisture source Normal or minimal ventilation With or without equipment heat source	20-90	5-100	100(C)
Drywell and Steam Tunnel	20-90	2-100	100(C)

Information withheld in accordance with 10 CFR 2.390

QUAD CITIES STATION UNITS 1 & 2
ENVIROMENTAL ZONE MAP (BASEMENT FLOOR PLAN) ELEVATION 554'-Ø"
FIGURE 3.11-1 SHEET 2 REVISION 5, JUNE 1999

Information withheld in accordance with 10 CFR 2.390

QUAD CITIES STATION UNITS 1 & 2
ENVIRONMENTAL ZONE MAP (GROUND FLOOR PLAN) ELEVATION 595'-0"
FIGURE 3.11-1 SHEET 3 REVISION 7, JANUARY 2003

Figure 3.11-1, Sheet 3a

Zone	Normal (18) Zone Temp (°F)	Normal (18) Press. (PSIA)	Normal (18) Humidity (%RH)	Normal (18) 40-Yr Dose (Rads) (19)	HELB (1) (11) (23) Temp (°F)	HELB (1) (11) (23) Press (PSIA)	HELB (1) (11) (23) Humidity (%RH)	LOCA (5) (8) (9) (10) (12) (13) Temp (°F)	LOCA (5) (8) (9) (10) (12) (13) Press (PSIA)	LOCA (5) (8) (9) (10) (12) (13) Humidity (%RH)	LOCA (5) (8) (9) (10) (12) (13) 1-Hr Dose (Rads) (2) (3) (16)	LOCA (5) (8) (9) (10) (12) (13) 30-Day Dose (Rads) (2) (3) (16)
9 (15)	150 (21)	14.7	20-90	1.1E07	338 (MS) (14)	40	100 (C)	294	63	100 (C)	2.8E07	1.4E08
10	151 (17)	14.7	20-90	2.2E06	304 (MS)	27.5	100 (C)	150	14.7	100 (NC)	1.7E06	7.8E06
11	104 (6)	14.7	20-90	<1.0E04	194 (HPCI)	14.8	100 (C)	141 (7)	14.7	100 (NC)	1.5E05	2.8E06
12	104 (6)	14.7	20-90	<1.0E04	194 (HPCI)	14.8	100 (C)	141 (7)	14.7	100 (NC)	3.6E05	1.8E06
13	104 (6)	14.7	20-90	<1.0E04	194 (HPCI)	14.8	100 (C)	141 (7)	14.7	100 (NC)	1.5E05	2.8E06
14	104 (6)	14.7	20-90	<1.0E04	194 (HPCI)	14.8	100 (C)	141 (7)	14.7	100 (NC)	<1.0E04	2.8E05
15	104 (6)	14.7	20-90	<1.0E04	194 (HPCI)	14.8	100 (C)	141 (7)	14.7	100 (NC)	1.5E04	2.8E05
16	104 (6)	14.7	20-90	<1.0E04	194 (HPCI)	14.8	100 (C)	141 (7)	14.7	100 (NC)	<1.0E04	2.8E05
17	104 (6)	14.7	20-90	<1.0E04	194 (HPCI)	14.8	100 (C)	141 (7)	14.7	100 (NC)	<1.0E04	2.9E04
18	120	14.7	20-90	<1.0E04	120 (FW/MS)	14.7	100 (NC)	120	14.7	100 (NC)	<1.0E04	<1.0E04
18a	80	14.7	20-90	<1.0E04	80	14.7	90	80	14.7	90	<1.0E04	<1.0E04
19	104	14.7	20-90	<1.0E04	104	14.7	100 (NC)	104	14.7	100 (NC)	<1.0E04	<1.0E04
19a	124 (17)	14.7	20-90	2.2E06	200 (FW/MS)	17	100 (C)	120	14.7	100 (NC)	<1.0E04	<1.0E04

(C) Condensing
(NC) Non-Condensing

Information withheld in accordance with 10 CFR 2.390

QUAD CITIES STATION
UNITS 1 & 2

ENVIROMENTAL ZONE MAP
(GROUND FLOOR PLAN)
ELEVATION 595'-0"

FIGURE 3.11-1 SHEET 4
REVISION 5, JUNE 1999

Information withheld in accordance with 10 CFR 2.390

QUAD CITIES STATION UNITS 1 & 2
ENVIRONMENTAL ZONE MAP (MEZZANINE FLOOR PLAN) ELEVATION 623'-0"
FIGURE 3.11-1 SHEET 5 REVISION 7, JANUARY 2003

Figure 3.11-1, Sheet 5a

Zone	Normal (18)				HELB (1) (11)(23)			LOCA (5) (8) (9) (10) (13)				
	Zone Temp (°F)	Press. (PSIA)	Humidity (%RH)	40-Yr Dose (Rads) (19)	Temp (°F)	Press (PSIA)	Humidity (%RH)	Temp (°F)	Press (PSIA)	Humidity (%RH)	1-Hr Dose (Rads) (2) (3) (16)	30-Day Dose (Rads) (2) (3) (16)
20 (15)	150	14.7	20-90	2.2E07	338 (MS) (14)	40	100 (C)	294	63	100 (C)	2.8E07	1.4E08
21	104 (6)	14.7	20-90	<1.0E04	170 (HPCI) 201 (RWCU)	14.8	100 (C)	143 (7)	14.7	100 (NC)	2.8E05	3.5E06
22	104 (6)	14.7	20-90	6.5E06	214 (RWCU)	15.7	100 (C)	148 (7)	14.7	100 (NC)	<1.0E04	<1.0E04
23	150	14.7	20-90	2.2E06	304 (MS)	27.5	100 (C)	150	14.7	100 (NC)	<1.0E04	<1.0E04
24	104 (6)	14.7	20-90	<1.0E04	170 (HPCI) 201 (RWCU)	14.8	100 (C)	143 (7)	14.7	100 (NC)	5.2E04	4.6E05
25	124 (17)	14.7	20-90	2.2E06	200 (FW/MS)	17	100 (C)	120	14.7	100 (NC)	<1.0E04	<1.0E04
26	120	14.7	20-90	<1.0E04	120 (FW/MS)	14.7	100 (NC)	120	14.7	100 (NC)	4.3E05	2.1E06
26a	80	14.7	20-90	<1.0E04	80	14.7	90	80	14.7	90	<1.0E04	<1.0E04
27	104 (6)	14.7	20-90	<1.0E04	170 (HPCI) 201 (RWCU)	14.8	100 (C)	143 (7)	14.7	100 (NC)	4.3E05	2.8E06

(C) Condensing
(NC) Non-Condensing

Information withheld in accordance with 10 CFR 2.390

QUAD CITIES STATION UNITS 1 & 2
ENVIROMENTAL ZONE MAP (MEZZANINE FLOOR PLAN) ELEVATION 623'-0"
FIGURE 3.11-1 SHEET 6 REVISION 5, JUNE 1999

Information withheld in accordance with 10 CFR 2.390

QUAD CITIES STATION UNITS 1 & 2
ENVIRONMENTAL ZONE MAP (MAIN FLOOR PLAN) ELEVATION 647'-6"
FIGURE 3.11-1 SHEET 7 REVISION 7, JANUARY 2003

Figure 3.11-1, Sheet 7a

Zone	Normal (18)				HELB (1) (11) (23)			LOCA (5) (8) (9) (10) (13)				
	Zone Temp (°F)	Press. (PSIA)	Humidity (%RH)	40-Yr Dose (Rads) (19)	Temp (°F)	Press (PSIA)	Humidity (%RH)	Temp (°F)	Press (PSIA)	Humidity (%RH)	1-Hr Dose (Rads) (2) (3) (16)	30-Day Dose (Rads) (2) (3) (16)
28 (15)	150	14.7	20-90	1.1E07	338 (MS) (14)	40	100 (C)	294	63	100 (C)	2.8E07	1.4E08
29	104 (6)	14.7	20-90	<1.0E04	186 (RWCU) 153 (HPCI)	14.8	100 (C)	124 (7)	14.7	100 (NC)	4.3E05	2.2E06
30	104 (6)	14.7	20-90	<1.0E04	186 (RWCU) 153 (HPCI)	14.8	100 (C)	124 (7)	14.7	100 (NC)	4.6E05	2.1E06
31	104 (6)	14.7	20-90	<1.0E04	186 (RWCU) 153 (HPCI)	14.8	100 (C)	124 (7)	14.7	100 (NC)	4.3E05	2.2E06
32	104 (6)	14.7	20-90	<1.0E04	186 (RWCU) 153 (HPCI)	14.8	100 (C)	124 (7)	14.7	100 (NC)	2.1E05	7.8E05
33	104 (6)	14.7	20-90	4.3E06	104	14.7	100 (NC)	124 (7)	14.7	100 (NC)	<1.0E04	<1.0E04
34	104 (6)	14.7	20-90	<1.0E04	186 (RWCU) 153 (HPCI)	14.8	100 (C)	124 (7)	14.7	100 (NC)	3.6E05	1.8E06
35	120	14.7	20-90	<1.0E04	120 (FW/MS)	14.7	100 (NC)	120	14.7	100 (NC)	<1.0E04	<1.0E04
35a	104	14.7	20-90	<1.0E04	104	14.7	100 (NC)	104	14.7	100 (NC)	<1.0E04	<1.0E04

(C) Condensing
(NC) Non-Condensing

Information withheld in accordance with 10 CFR 2.390

QUAD CITIES STATION UNITS 1 & 2
ENVIROMENTAL ZONE MAP (MAIN FLOOR PLAN) ELEVATION 647'-6"
FIGURE 3.11-1 SHEET 8 REVISION 5, JUNE 1999

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QUAD CITIES STATION UNITS 1 & 2
ENVIRONMENTAL ZONE MAP (REACTOR FLOOR PLAN) ELEVATION 666'-6"
FIGURE 3.11-1 SHEET 9 REVISION 7, JANUARY 2003

Figure 3.11-1, Sheet 9a

Zone	Normal (18)				HELB (1) (11)			LOCA (5) (8) (9) (10) (13)				
	Zone Temp (°F)	Press. (PSIA)	Humidity (%RH)	40-Yr Dose (Rads) (19)	Temp (°F)	Press (PSIA)	Humidity (%RH)	Temp (°F)	Press (PSIA)	Humidity (%RH)	1-Hr Dose (Rads) (2) (3) (16)	30-Day Dose (Rads) (2) (3) (16)
36 (15)	150	14.7	20-90	1.1E07	338 (MS) (14)	40	100 (C)	294	63	100 (C)	2.8E07	1.4E08
37	104 (6)	14.7	20-90	<1.0E04	104	14.7	100 (NC)	122 (7)	14.7	100 (NC)	8.7E05	4.8E06
38	104 (6)	14.7	20-90	<1.0E04	104	14.7	100 (NC)	111 (7)	14.7	100 (NC)	<1.0E04	<1.0E04
39	120	14.7	20-90	<1.0E04	120 (FW/MS)	14.7	100 (NC)	120	14.7	100 (NC)	<1.0E04	<1.0E04
40	104 (6)	14.7	20-90	<1.0E04	104	14.7	100 (NC)	109 (7)	14.7	100 (NC)	<1.0E04	<1.0E04

(C) Condensing
(NC) Non-Condensing

Information withheld in accordance with 10 CFR 2.390

QUAD CITIES STATION UNITS 1 & 2
ENVIROMENTAL ZONE MAP (REACTOR FLOOR PLAN) ELEVATION 666'-6"
FIGURE 3.11-1 SHEET 10 REVISION 5, JUNE 1999