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*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.

4.0 REACTOR

4.1 SUMMARY DESCRIPTION

This chapter includes the following sections: [4.1-1]

Section	Description
4.2	Reactor fuel system design
4.3	Nuclear design and stability
4.4	Thermal and hydraulic design
4.5	Reactor materials
4.6	Functional design of reactivity control systems

The equipment and evaluations presented in this chapter are applicable to either Unit 1 or Unit 2.

Section 4.2 describes the design of fuel assemblies used in Quad Cities Units 1 and 2, SVEA-96 Optima2 fuel types supplied specifically by Westinghouse Electric Company (WEC) and the ATRIUM 10XM fuel provided by AREVA. Section 4.2 also describes the use of gadolinia (Gd_2O_3) as a burnable neutron absorber in UO_2 fuel. A detailed description of WEC fuel designs is contained in (Westinghouse Document) WCAP - 15942[9] and reload specific Fuel Design Reports[10]. [4.1-2] A detailed description of the AREVA ATRIUM 10XM fuel design is contained in ANP-3305P Revision 2 (Reference 11).

Section 4.3 presents a discussion of nuclear design and reactor stability, including discussions of reactivity coefficients and their contributions to stability.

Section 4.4 presents a summary of the thermal and hydraulic design of the reactor core and the reactor coolant system (including a description of the power-flow map). The presentation stresses the safety limit minimum critical power ratio (MCPR), the operating limit MCPR, the maximum linear heat generation rate (MLHGR), and the maximum average planar linear heat generation rate (MAPLHGR). The MCPR calculation methodology, the critical power correlations, and the associated uncertainties are also discussed. In order to provide reference data for the Technical Specifications, the core power safety limit and limiting safety system settings are discussed. Transient analyses have shown the effectiveness of the protection system in preventing the unit from approaching conditions of safety concern. [4.1-3]

A typical power-flow map is shown in Figure 4.4-1. The extensions to the operating region above the 100% Flow Control Line (FCL) and 100% Core Flow include the Maximum Extended Load Line Limit Analysis (MELLLA) region above 100% FCL and the Increased Core Flow (ICF) region beyond 100% Core Flow. MELLLA is discussed in Section 4.4.3.1.8, and ICF is discussed in Section 4.4.3.1.9. Additionally, analyses of EPU evaluated several Equipment out of Service (EOOS) operating flexibility options (Reference 8). For a discussion of these EOOS options, see the cycle specific reload documentation.

Steady operation at any power level is warranted as long as the following requirements are met: [4.1-5]

A. The MCPR, MLHGR, and MAPLHGR limits are satisfied, and

B. Transient and accident analysis results have been shown to be valid for power levels up to the level in question.

By establishing approved thermal limits for the fuel, it is possible to permit power outputs up to the maximum licensed power level whenever the power distribution is favorable. [4.1-6]

Principal core and fuel design data are summarized in Table 4.1-1. Principal nuclear design limits, targets, and typical values are summarized in Table 4.1-2. Principal thermal and hydraulic design values are summarized in Table 4.1-3. [4.1-7]

Reactor vessel internals are described in Section 3.9.5. Reactor internals materials are described in Section 4.5.2. Quality control methods are used during the fabrication and assembly of reactor internals to assure that design specifications are met. [4.1-8]

Section 4.6 addresses the design of reactivity control systems including the control rod drive system, and the design of the control rods. Control rod drive system structural materials are described in Section 4.6.2.1. [4.1-9]

4.1.1 References

1. Deleted.

2. Deleted.

3. Deleted.

4. Deleted.

5. Deleted.

6. Deleted.

7. Deleted.

8. “Dresden 2 and 3, Quad Cities 1 and 2 Equipment Out-of-Service and Legacy Fuel Transient Analysis,” General Electric Company, GE-NE-J11-03912-00-01-R3, September 2005.

9. WCAP-15942-P-A, "Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors – Supplement 1 to CENP-287," March 2006.

10. "Generic SVEA-96/Optima2 Fuel Assembly Design Package" Westinghouse Letter PD1-05-100, November 1, 2005 (included in OPTIMA2-TR032QC-FUEL_ASSY, Optima2 Fuel Assembly Design Package – Quad Cities).

11. ANP-3305P, Revision 2, “Mechanical Design Report for Quad Cities and Dresden ATRIUM 10XM Fuel Assemblies,” AREVA Inc., August 2016.

Table 4.1-1

CORE AND FUEL DESIGN

Core		
Equivalent core diameter		182.2 in
Circumscribed core diameter		189.7 in
Core lattice pitch (control cell)		12 in
Number of fuel assemblies in the core		724
Fuel Assembly		
	ATRIUM 10XM	Optima2
Fuel rod array	10 x 10	10 x 10
Fuel rod pitch	Note 4 Zircaloy	Note 2
Channel material	Note 4	Zircaloy-2
Fuel assembly weight Note 1	91	624 lb
Number of fuel rods	Note 5	96
Number of water rods	---	Note 3
Heat transfer area		Note 2
Fuel Rod, Cold		
	ATRIUM 10XM	Optima2
Fuel pellet diameter	Note 4	Note 2
Cladding thickness	Note 4	Note 2
Cladding outside diameter	Note 4	Note 2
Active fuel length	Note 4	145.28 in
Fuel material	Note 4	UO_2
Cladding material	Note 4	Zircaloy-2

Note 1: Approximate weight, including channel. Actual weight varies with fuel loading.

Note 2: Material is Westinghouse proprietary. See WCAP-15942-P-A, "Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors – Supplement 1 to CENP-287."

Note 3: SVEA channel contains a watercross (See Figure 4.2-6b). There is no need for water rod(s).

Note 4: This information is proprietary to AREVA and is available in the Reference 11 ANP-3305P, Rev. 2 Mechanical Design Report (subject to the proprietary marking) or the cycle-specific reload analysis report.

Note 5: AREVA ATRIUM 10XM fuel has a square internal water channel that displaces a 3 x 3 array of rods.

Table 4.1-1 (Cont'd)

CORE AND FUEL DESIGN

Movable Control Rods	
Number of control rods in the core	177
Shape	Cruciform
Pitch	12.0 in
Stroke	144 in
Width	9.75 in
Control length	143 in
Control material	B_4C and Hafnium
Burnable Neutron Absorber	
Control material	Gd_2O_3
Location	Mixed with UO_2 within selected portions of several fuel rods per fuel assembly
Concentration	Location dependent

Table 4.1-2

NUCLEAR DESIGN LIMITS, TARGETS, AND TYPICAL VALUES

Reactivity Control		
Cold shutdown k_{eff},rod of maximum worth stuck out fully — operating limit	Highest worth control rod analytically determined	0.9962
	Highest worth control rod determined by test.	0.9972
Cold shutdown k_{eff}, rod of maximum worth stuck out fully— design goal	0.99 (0.99 k_{eff} core design is based on projected conditions for the previous cycle)	
Reactivity insertion rate of control rods	Movement of one notch of one rod at 5.14 in/s maximum	
Standby liquid control shutdown k_{eff}, design limit	0.99	
Reactivity Coefficients		
Moderator temperature coefficient	See Cycle Startup Report	
Moderator void coefficient	See Cycle Specific Reload Document	
Fuel temperature (Doppler) coefficient	See Cycle Specific Reload Document	
Power coefficient for xenon stability	More Negative than -0.01 (Δ-k/k)/(Δ -P/P)	
Typical Core Average Neutron Flux		
Thermal Neutrons	3.5×10^{13} n/cm^2-sec[Note 1]	
Neutrons > 1 MeV	3.7×10^{13} n/cm^2-sec	
Typical Excursion Parameter Values		
Prompt neutron lifetime *(l)*	40 microseconds	
Effective delayed neutron fraction (ß) — at 0 MWd/t — at 10,000 MWd/t	 0.0072 0.0056	

Note 1: The "thermal" flux values from the plant computer may be significantly different from this value if the energy grouping in the computer calculation does not represent a truly thermal spectrum.

Table 4.1-3

THERMAL AND HYDRAULIC DESIGN

Design thermal output		2957 MWt
Reactor pressure at steam dome (nominal operating point)		1020 psia
Steam flow rate		11.713 x 10^6 lb/hr
Rated recirculation flow rate*		98 x 10^6 lb/hr
Core subcooling (relative to mid-plane pressure)		24.1
Maximum LHGR	SVEA–96 Optima2 ATRIUM 10XM	13.72 kW/ft See Reload Licensing Report
Safety limit MCPR		See Reload Licensing Report

* Analyzed for Increased Core Flow to 108% of rated.

4.2 FUEL SYSTEM DESIGN

4.2.1 Design Bases

Both units contain nuclear fuel assemblies designed and fabricated by Westinghouse Electric Company (WEC) and AREVA. This section provides description of the fuel and methodology utilized. [4.2.1]

4.2.1.1 Fuel Assemblies

The fuel rods and assemblies are designed to assure, in conjunction with the core, nuclear, thermal, hydraulic, and unit equipment characteristics, the nuclear instrumentation, and the reactor protection system, that fuel damage will not occur during normal operation or transients caused by any single equipment malfunction or any single operator error. Fuel damage is defined as perforation of the fuel cladding which would permit the release of fission products to the reactor coolant. [4.2.2]

The mechanisms which can cause fuel damage in reactor transients are: [4.2.3]

A. Severe overheating of the fuel cladding caused by inadequate cooling. For design purposes the critical heat flux, or equivalently, the critical power (the onset of the transition from nucleate boiling to film boiling) is conservatively defined as the limit. However, experimental data indicates that fuel damage will not occur until well into the film boiling regime.

B. Fracture of the fuel cladding due to stress caused by the relative expansion of the UO_2 pellet. For design purposes, a value of 1% strain of the Zircaloy cladding is used as the limit. Below this value, fuel damage due to overstraining of the fuel cladding is not expected to occur.

Fuel damage from cladding overheat is prevented by avoiding boiling transition in the core. Boiling transition is prevented by observance of the MCPR operating and safety limits. The MCPR Fuel Cladding Integrity Safety Limit is defined such that during sustained, steady-state operation at the MCPR Safety Limit, at least 99.9% of the fuel rods in the core are expected to avoid boiling transition with a 95% confidence level[22]. The MCPR Operating Limit is defined such that the occurrence of the most severe moderately frequent anticipated event will not result in violation of the MCPR Safety Limit. These limits together assure that at least 99.9% of the fuel rods in the reactor core will be expected to avoid boiling transition during normal operation and anticipated operational occurrences. [4.2.4] Details of this reload licensing analysis process and setting of MCPR safety limits and MCPR operating limits is provided in Section 4.4.4.1.

Applicable Westinghouse CPR correlations that are NRC-approved are applied to determine the MCPR Safety Limit (SLMCPR) for the Westinghouse fuel. In addition, Westinghouse applies the approved methodology in Reference 22 to establish CPR correlations and conservative factors, which are used to determine the Operating Limit MCPR (OLMCPR) for any co-resident non-Westinghouse fuel that may be present in the core during the cycle operations.

For Quad Cities AREVA reload cores (starting with Quad Cities Unit 1 Cycle 25), the following NRC approved correlations and application methodology are used to calculate CPR and the SLMCPR: The Reference 28 ACE correlation is used for CPR calculations for ATRIUM 10XM fuel, the Reference 29 SPCB correlation is used for CPR calculations of the OPTIMA2 fuel applying the Reference 30 methodology for use of approved AREVA critical power correlations for non-AREVA co-resident fuel, and the Reference 31 process is used to calculate SLMCPR with the SAFLIM3D code.

To prevent fuel damage by thermal stress, the linear heat generation rate (LHGR) is used as a fuel design parameter. The purpose of the LHGR limits is to prevent strain of the cladding from exceeding 1% and to prevent fuel melting. The LHGR limits are a function of fuel type, exposure, and gadolinia content of the fuel. A further discussion of LHGR is contained in Section 4.4. [4.2.5]

The WEC Methodology uses an advanced computer code (e.g., STAV (WEC)) for fuel rod thermal-mechanical calculations. Some details of this WEC methodology are included in the following sub-paragraphs:

This model calculates LHGRs in compliance with the fuel design basis criteria of:

A. No fuel melting during normal steady-state operation and whole core anticipated operational occurrences; and

B. A small amount of fuel melting not exceeding 1% cladding strain for local anticipated operational occurrences. [4.2.6]

The effects of fuel densification have been considered in the fuel performance analysis. A decrease in the length of pellets could result in the formation of axial gaps in the column of fuel pellets within a fuel rod. A decrease in pellet radius could result in the increase in the radial clearance between the fuel pellet and the fuel rod cladding. The following four principal effects are associated with the dimensional changes resulting from densification. [4.2.7]

1. Axial gaps produce a local increase in the neutron flux and generate a local power spike.

2. A decrease in pellet length directly results in a proportional increase in the LHGR.

3. If relatively large axial gaps form, creepdown of the cladding later in life may lead to the collapsing of the cladding into the gaps.

4. Decreased pellet radius results in decreased pellet-clad thermal conductance (gap conductance) which increases the fuel pellet temperature and stored energy and decreases the heat transfer capability of the fuel rod.

To account for these four effects, the WEC models for fuel densification consist of four parts: power spike model, linear heat generation model, cladding creep collapse model, and stored energy model. These models can be used to conservatively evaluate the effects of WEC fuel densification.

The net effect of gadolinia in the fuel on densification has been found to be beneficial, i.e., it tends to alleviate the power spiking due to fuel densification. [4.2.8]

The AREVA methodology uses the approved RODEX4 code and licensing analysis bases described in BAW-10247PA (Reference 23) to establish ATRIUM 10XM fuel LHGR limits that meet required steady-state and AOO transient criteria.

The potential problems of channel bow in BWRs that could impact local peaking and reduce available MCPR margin, as addressed in IE Bulletin 90-02, have been considered for Quad Cities. Since the issuance of IE Bulletin 90-02, the Westinghouse analysis for reload fuel includes the effects of channel bow; this is accomplished by adjusting the bundle R-factors to accommodate channel bow. [4.2.9] For Quad Cities AREVA reload cores (starting with Quad Cities Unit 1 Cycle 25), AREVA includes the impact of fluence gradient induced channel bow using an NRC approved model in the approved methodologies for establishing the Safety Limit MCPR (Reference 31) and the fuel thermal-mechanical design limits (Reference 23).

The fuel vendors have NRC approved emergency core cooling system (ECCS) analytic codes for evaluating the effects of loss-of-coolant accidents (LOCA) in accordance with 10 CFR 50, Appendix K, to assure that the peak cladding temperature (PCT) of the hottest fuel rod in the core will not exceed the NRC imposed PCT limit of 2200°F under postulated accident conditions. The PCTs calculated are less than the regulatory limit per 10 CFR 50.46. See Section 6.3 for ECCS discussions and Section 15.6.5 for LOCA analysis. [4.2.11]

Nuclear Fuel is designed to retain cladding integrity through normal operation and anticipated operational occurrences. The fuel is designed to an LHGR limit, which is the upper bound of the multiple design histories used in the statistical evaluation of fuel rod performance. Because of fissile inventory and operational power-flattening requirements, a fuel rod is physically incapable of operating at the LHGR limit for its entire lifetime. Any fuel rod in the core may, however, operate for an extended period at the LHGR limit without jeopardizing fuel cladding integrity at any point in its design lifetime.

A detailed description of the design criteria and evaluation methodology for Westinghouse Optima2 fuel is contained in WCAP-15942-P-A[19] and WCAP-15836-P-A[20].

A detailed description of the thermal-mechanical design criteria and evaluation methodology for AREVA fuel is contained in BAW-10247PA (Reference 23).

4.2.1.2 Control Rods

Design information for the control rods is contained in Section 4.6.

4.2.1.3 Burnable Neutron Absorber

The loading of gadolinia (Gd_2O_3) as burnable neutron absorber in a fuel bundle is determined by the need to: [4.2.12]

A. Meet shutdown margin requirements throughout the cycle (see Section 4.2.3.4.2);

B. Meet thermal margin requirements by adjustment of the axial power shape, particularly in the lower portion of the core; and

C. Maintain hot excess reactivity at a level which can be readily controlled at all exposure points during the cycle.

These needs must be balanced against the desire to minimize undepleted gadolinia at the end of cycle. Undepleted gadolinia can lead to a loss in attainable cycle exposure.

4.2.2 Description and Design Drawings

This section provides descriptions of the fuel designs provided by each current fuel vendor. [4.2.13]

4.2.2.1 Fuel Assemblies

The Westinghouse fuel assemblies used at Quad Cities are SVEA-96 Optima2 fuel assemblies. Descriptions of the fuel assemblies are contained in WCAP-15942-P-A, "Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors – Supplement 1 to CENP-287" [19] and reload specific reports [20] for Westinghouse. [4.2.14]

Beginning with Unit 2 Cycle 22 and continuing in subsequent cycles of Unit 2, Westinghouse Lead Test Assemblies (LTAs) are inserted into non-limiting core locations for demonstration purposes. The LTA program consists of eight Lead Test Assemblies of OPTIMA2 fuel with Low Tin ZIRLO channels.

The top view of a typical fuel cell is shown in Figure 4.2-4. For Westinghouse fuel assemblies, the lifting handle is directly attached to the fuel channel. The identifying assembly serial number is engraved on the top of the handle. No two assemblies bear the same serial number. A boss may project from one side of the handle to aid in ensuring proper fuel assembly orientation. For Westinghouse fuel, bypass flow is controlled by a bypass flow hole in the SVEA channel. [4.2.16]

The fuel rods contain UO_2 pellets manufactured by compacting and sintering UO_2 powder into cylindrical pellets. The pellets are enclosed in Zircaloy-2 tubes (cladding), which are evacuated, back-filled and pressurized with helium, and sealed by welding Zircaloy plugs in each end. The fuel rod cladding thickness is adequate to satisfy the requirement that the cladding be "free standing." That is, the cladding has adequate margin against collapse under normal operating pressures and overpressures, including hydro-pressure test conditions of 200°F and over 1900 psia. Gaseous products released from the pellet accumulate in a plenum at the top of the rod. Sufficient plenum volume is provided to prevent excessive internal pressure from gases accumulated over the design life of the fuel. [4.2.17]

A channel encloses the fuel bundle. Channels are made of Zircaloy-2. The channel performs the following functions: [4.2.18]

A. Forms the fuel assembly coolant flow path outer periphery;

B. Provides surfaces for control rod guidance in the reactor core;

C. Provides structural stiffness to the fuel bundle during lateral loadings applied from fuel rods through the fuel spacers;

D. Transmits fuel assembly seismic loadings to the top guide and fuel support of the core internal structures;

E. For SVEA-96 Optima2 fuel, carries the load of the bundle during fuel handling;

F. Provides a heat sink during LOCA; and

G. Provides a stagnation envelope for incore fuel sipping.

For SVEA-96 Optima2 fuel, the cruciform shaped channel (cross-section shown on Figure 4.2-6b) is attached to a bottom inlet piece by means of four screws. A handle is attached to the top of the channel, which carries the load when the fuel assembly is moved.

Proper orientation of fuel assemblies in the reactor core is readily verified by visual observation and is assured by procedural requirements during core loading verification. There are several ways of confirming proper fuel assembly orientation (see Figure 4.2-4): [4.2.19]

A. All assembly serial numbers point toward the center of the cell;

B. The channel fastener spring-clip assemblies are located adjacent to the center of the control rod;

C. If present, the protrusions (lugs) on the assembly handles all point toward the adjacent control rod;

D. There is cell-to-cell symmetry; and

E. All channel spacing buttons face one another.

The upper tieplate is a cast and machined grid plate with attached bail handle to provide for fuel assembly handling and orientation. A unique serial identification number is engraved on the bail handle of each tieplate. This number can be read under water to allow identification of the assemblies in the core. [4.2.23]

Westinghouse Optima2 fuel was first introduced into the Quad Cities Unit 2 reactor core beginning with Unit 2 Cycle 19.

The SVEA-96 Optima2 fuel assembly (Figure 4.2-5) consists of three basic components:

- the fuel bundle,
- the fuel channel, and
- the handle.

The fuel bundle consists of 96 fuel rods, arranged in four 5x5-1 sub-bundles. The sub-bundles are separated by a cruciform internal structure (watercross) in the channel. The sub-bundles are inserted into the channel from the top. This design principle has always been used in Westinghouse BWR fuel and eliminates any leakage flow ambiguities at the bottom end of the channel. This feature also avoids stresses in the tie-rods during normal fuel handling since the weight of the fuel bears on the channel and bottom support. The bottom support with the integrated debris filter is designed to prevent potentially damaging debris from entering the fuel bundle. The sub-bundles are freestanding inside the channel. There is sufficient space for sub-bundle growth at the top of the assembly to avoid restriction due to differential growth between the fuel bundles and the channel. The bottom of the transition piece, or "nose piece," seats in the fuel support piece.

The fuel channel for the SVEA-96 Optima2 consists of an inlet piece and a Zircaloy channel. The inlet piece, which is composed of a transition piece and a bottom support, is equipped with an integrated debris filter and is fastened to the channel by four screws. The Zircaloy channel consists of an outer channel with a square cross section and an internal double-walled, cruciform structure, or "watercross," which forms channels for non-boiling water as shown in Figures 4.2-6a and 4.2-6b. The watercross structure is composed of a square central water channel and smaller water channels in each of the four wings. The watercross structure, along with the outer channel walls, form four sub-channels in which the sub-bundles are positioned. A screw is welded to the top end of the cross which attaches the handle with the leaf spring to the fuel channel.

The handle and leaf spring design for the Optima2 assembly is shown in Figure 4.2-7, and is fitted to the top end of the channel. The handle is designed for lifting with the standard handling equipment at the plant. An individual identification number for the fuel assembly is engraved in the handle. The handle is equipped with a double leaf spring which maintains contact with the corresponding springs on adjacent assemblies and firmly presses the fuel assembly into the corner of the upper core grid.

A complete description of the SVEA-96 Optima2 fuel assembly is given in WCAP-15942-P-A [19].

AREVA ATRIUM 10XM reload fuel assemblies are used for Quad Cities Unit 1 starting with Unit 1 Cycle 25. The ATRIUM 10XM fuel bundle geometry consists of a 10x10 fuel lattice with a square internal water channel that displaces a 3x3 array of rods. The fuel assembly consists of a lower tie plate, an upper tie plate, full-length and part-length rods. The structural members of the fuel assembly include the tie plates, spacer grids, central water channel and miscellaneous assembly hardware. Figure 4.2-3 provides a diagram of the ATRIUM 10XM fuel assembly. The fuel assembly is accompanied by an advanced fuel channel and a winged fuel channel fastener. The ATRIUM 10XM fuel is designed for mechanical, nuclear, and thermal-hydraulic compatibility with other fuel designs. ATRIUM 10XM incorporates a debris filter in the lower tie plate. A detailed description of the ATRIUM 10XM design is provided in Reference 27.

The Atrium 10XM assemblies are designed in compliance with U. S. NRC approved fuel licensing criteria defined in Reference 24, which provides a set of generic acceptance criteria to be satisfied by AREVA for new BWR fuel designs. In accordance with the process described in Reference 24, demonstration that the new fuel designs meet the generic acceptance criteria is required. In addition, Reference 24 requires that AREVA provide the NRC with a summary of the evaluation of the design relative to the acceptance criteria. Reference 25 meets that requirement and contains a detailed description of the ATRIUM 10XM design, the ATRIUM 10XM evaluation results, a summary of the operating experience and the post-irradiation examination (PIE) results supporting the ATRIUM 10XM design features. Documentation of AREVA analyses performed to demonstrate compliance with any design and licensing criteria for a specific plant and/or cycle is provided as part of the normal reload licensing document package.

4.2.2.1.1 Special Test Fuel Types

The Quad Cities units have, over the years, operated with a number of special test fuel types. These fuel types have been analyzed and safety evaluations have been performed to assure that the units can operate safely with these fuel types. The following sections describe the fuel types that have significant impacts on current fuel design. [4.2.24]

4.2.2.1.1.1 Special Barrier Fuel Bundles

Experience in the nuclear industry with fuel rods of Zircaloy-clad uranium dioxide (UO_2) has revealed several causes of fuel rod failure. Most of these causes have been corrected by innovative design modifications and by improvements in manufacturing processes. One type of failure, pellet-cladding interaction (PCI), is closely linked to the power history of the fuel rod and to the magnitude, rate, and duration of power changes. Pellet-cladding interaction fuel failures have occurred in all types of water-cooled reactors that are fueled with UO_2 sheathed in Zircaloy: boiling water reactors (BWRs), pressurized water reactors (PWRs), Canadian deuterium-moderated reactors (CANDUs), and steam generating heavy water reactors (SGHWRs). [4.2.25]

Pellet-cladding interaction was first identified by GE in a BWR plant in 1971. Since then, GE embarked upon a program to understand the PCI phenomenon and to develop potential remedies. This effort resulted in a series of design changes to minimize the frequency of these types of failures. In early 1972, the 7x7 fuel rod design was improved by reducing the ratio of pellet length to diameter, chamfering the pellet ends, eliminating pellet dishing to reduce localized strain, and increasing the cladding heat treatment temperature to reduce variability in cladding mechanical properties. Pellet-cladding interaction failures are dependent on fuel rod power histories. Rod powers were reduced with the 8x8 fuel design introduced in the spring of 1974. The 8x8R fuel design further reduced fuel rod local peaking. In later design changes, the 8x8R fuel rods were prepressurized to improve gap conductance and thus lower fuel temperatures. This reduction in fuel temperature is equivalent to a reduction in linear power and acts to increase resistance to PCI.

A further innovation was the development of "barrier fuels" which have a special fuel cladding designed to protect the Zircaloy sheath from the harmful effects of localized stress and reactive fission products during reactor service. Considerable testing has been performed on barrier fuels, as exemplified by the testing documented in NEDM-31383P-6[3].

The barrier fuel design selected for testing had a fuel sheath, or cladding, constructed of Zircaloy-2, and a liner of high-purity zirconium metallurgically bonded to the Zircaloy. The zirconium liner was 0.003-inch thick, i.e., approximately 10% of the total cladding wall thickness. The dimensions of the fuel rods and the mechanical design of the fuel bundle were the same as the prepressurized 8x8 D-lattice retrofit bundle (P8x8R).

With the successful completion of the barrier fuel demonstration program and the implementation of PCI-resistant zirconium barrier fuel, there has been an overall increase in fuel performance for plants using the barrier fuel.

4.2.2.1.1.2 Unit 1 High Burnup Barrier Lead Test Assembly (BLTA) Program

A special test program was started in Unit 1 at BOC 10 to reirradiate six fuel rods from a BLTA that had previously been irradiated in Unit 2 from Cycles 5 through 9. The goal of the program is to gain operational experience with barrier fuel at exposures up to, and possibly in excess of, 60 GWd/MT (where MT denotes metric ton) on a rod-average basis. The six donor rods were selected from BLTA LJB587 and extensively precharacterized and examined for suitability for continued irradiation. GE performed an evaluation that demonstrated that for the projected peak pellet exposures — which may exceed the design limit on such exposure — all design and licensing criteria were satisfied. [4.2.26]

A host bundle, LYD396, of BP8x8R design that had been irradiated during Cycles 9 and 10 was reconstituted with the six donor rods at BOC 11. A new bundle skeleton (upper and lower tieplate, spacers, spacer capture water rod, locking tabs and nuts) was used to accommodate the different end plug diameters that exist between the original BLTA rods and the host bundle rods. The program was terminated at EOC 12. Detailed examination of the six test rods revealed a small number of areas on four of the test rods which exhibited localized spot spallation or loss of clad material from the outer diameter of the rods. A hot cell examination of the six BLTA rods is planned upon shipment of the BLTA skeleton LYJ449X containing the six test rods to GE's Vallecitos Laboratory. [4.2-27]

The Unit 1 High Burnup Barrier Lead Test Assembly Program was completed and is documented in Reference 18. Fuel performance was satisfactory.

4.2.2.2 Control Rods

Description and drawings of the control rods are contained in Section 4.6.

4.2.2.3 Burnable Neutron Absorber

4.2.2.3.1 Description of the Design

A characteristic of the fuel designs is axial zoning of the fuel bundle. In the fuel bundle, the number of gadolinia rods may vary, and the gadolinia concentration may vary with axial position. [4.2.28]

Enrichment and gadolinia distribution for AREVA and Westinghouse fuel in Quad Cities is provided in the cycle specific Fuel Design Report.

Fuel nuclear methods are discussed in EMF-2158 (Reference 26) for AREVA fuel and CENPD-300-P-A[22] for Westinghouse fuel.

Reactivity control and shutdown margin are discussed in Sections 4.2.3.4.2 and 4.3.2.1.3.

4.2.2.3.2 Material Properties

The Gd_2O_3 is uniformly distributed in the UO_2. The addition of small amounts of Gd_2O_3 to UO_2 has effects on both the conductivity and melting temperature of the solid pellet. (See Section 4.3.2.) These effects are accounted for in the design and safety evaluations performed for the fuel rod as described in Section 4.2.3. [4.2.29]

4.2.2.3.3 Operating Experience with Gadolinia-Containing Fuel

AREVA and Westinghouse have designed and fabricated nuclear fuel containing gadolinia burnable absorber for PWRs and BWRs for over thirty-five years. Westinghouse BWR experience is summarized in WCAP-15942-P-A[19]. [4.2.30] [4.2.31]

4.2.3 Design Evaluation

4.2.3.1 Fuel System Damage

This section applies to normal operation and anticipated operational occurrences except for Sections 4.2.3.1.3, 4.2.3.1.7, and 4.2.3.1.8 which apply to normal operation only. [4.2-32]

4.2.3.1.1 Stress/Strain

4.2.3.1.1.1 Bases

The fuel assembly components are evaluated to ensure that the fuel will not fail due to stresses or strains exceeding the fuel assembly component mechanical capability. [4.2-33]

4.2.3.1.1.2 Limits

For Westinghouse fuel, stress and strain limits are defined in Section 3.2 of WCAP-15942-P-A[19]. For AREVA Quad Cities Unit 1 ATRIUM 10XM fuel, see ANP-3305P Rev. 2 (Reference 27), Section 3.3.1 for the applicable design limits.

4.2.3.1.1.3 Evaluations

Analysis of the SVEA-96 Optima2 fuel shows stress and strain limits are not expected to be violated during normal operation or during AOOs[19]. For AREVA Quad Cities Unit 1 ATRIUM 10XM fuel, see ANP-3305P Rev. 2 (Reference 27), Table 3-1. The applicable design criteria are met.

4.2.3.1.2 Fatigue

4.2.3.1.2.1 Bases

The fuel assembly and the fuel rod cladding are evaluated to ensure that strains due to cyclic loadings will not exceed the fatigue capability. [4.2-34]

4.2.3.1.2.2 Limits

For Westinghouse fuel, limits for strains due to cyclic loading are defined in Section 3.2 and 3.3 of WCAP-15942-P-A[19]. For AREVA Quad Cities Unit 1 ATRIUM 10XM fuel, see ANP-3305P Rev. 2 (Reference 27), Section 3.3.2 for the applicable design limits.

4.2.3.1.2.3 Evaluations

Analysis of the AREVA ATRIUM 10XM and SVEA-96 Optima2 fuel shows that cyclic fatigue failure is not expected to occur during normal operation or during AOOs[19]. For AREVA Quad Cities Unit 1 ATRIUM 10XM fuel, see ANP-3305P Rev. 2 (Reference 27), Table 3-1. The applicable design criteria are met.

4.2.3.1.3 Fretting Wear

4.2.3.1.3.1 Bases

The fuel assembly is evaluated to ensure that fuel will not fail due to fretting wear of the assembly components. [4.2-35]

4.2.3.1.3.2 Limits

For Westinghouse fuel, fretting wear limits are set in Section 3.2 of WCAP-15942-P-A[19]. For AREVA Quad Cities Unit 1 ATRIUM 10XM fuel, see ANP-3305P Rev. 2 (Reference 27), Section 3.3.3 for the applicable design limits.

4.2.3.1.3.3 Evaluations

For SVEA-96 Optima2 fuel, examinations of a large number of irradiated fuel assemblies shows that the wear limits set in WCAP-15942-P-A[19], and substantiated in endurance tests, are met during normal operation and AOOs. For AREVA Quad Cities Unit 1 ATRIUM 10XM fuel, see ANP-3305P Rev. 2 (Reference 27), Table 3-1. The applicable design criteria are met.

4.2.3.1.4 Oxidation and Corrosion Products

4.2.3.1.4.1 Bases

The fuel rod is evaluated to ensure that the cladding temperature increase and cladding material thinning due to cladding oxidation and the cladding temperature increase due to the buildup of corrosion products do not result in fuel rod failure due to reduced cladding strength. [4.2-36] The bases for limitations on oxidation and corrosion products for AREVA fuel are shown in Section 3.3.4 of ANF-89-98(NP)(A) (Reference 24).

4.2.3.1.4.2 Limits

For Westinghouse fuel, oxidation and corrosion limits are defined in Section 3.3 of WCAP-15942-P-A [19]. The corrosion thickness for the AREVA methodology is provided in the NRC SER for the LAR that introduced AREVA fuel into the Quad Cities and Dresden cores and is a proprietary value. [32]

4.2.3.1.4.3 Evaluations

Analyses of the AREVA ATRIUM 10XM and SVEA-96 Optima2 fuel include an allowance for corrosion and oxidation, where appropriate, in the evaluation of performance against other limits. Extensive post-irradiation examinations have confirmed that those allowances are conservative.

In the event abnormal crud is observed for a plant, a specific analysis is required to address the higher crud level. An abnormal level of crud is defined by a formation that increases the calculated fuel average temperature by 25Â°C above the design bases calculation. The formation of crud is not calculated within the AREVA proprietary methodology. Instead, an upper bound of expected crud is input by the use of the crud heat transfer coefficient. The corrosion model also takes into consideration the effect of the higher thermal resistance from the crud on the corrosion rate. A higher corrosion rate is therefore included as part of the abnormal crud evaluation. A similar specific analysis is required if a plant experiences higher corrosion instead of crud.

4.2.3.1.5 Hydriding

4.2.3.1.5.1 Bases

The fuel rod is evaluated to ensure that failure will not occur due to internal cladding hydriding. [4.2-37]

4.2.3.1.5.2 Limits

For Westinghouse fuel, criteria for internal hydriding are determined in Section 3.3 of WCAP-15942-P-A[19]. For AREVA Quad Cities Unit 1 ATRIUM 10XM fuel, see ANP-3305P Rev. 2 (Reference 27), Section 3.3.4 for the applicable design limits.

4.2.3.1.5.3 Evaluations

Analysis of the SVEA-96 Optima2 fuel rod shows rod internal hydriding is negligible when manufacturing spec limits on pellet moisture, fill gas moisture, and internal clad moisture are met (Reference 19). For AREVA Quad Cities Unit 1 ATRIUM 10XM fuel, see ANP-3305P Rev. 2 (Reference 27), Table 3-1. The applicable design criteria are met.

4.2.3.1.6 Dimensional Changes

4.2.3.1.6.1 Bases

The fuel rod is evaluated to ensure that fuel rod bowing does not result in fuel failure due to boiling transition. [4.2-38]

4.2.3.1.6.2 Limits

For Westinghouse fuel, fuel rod bow design criteria are set in Section 3.3 of WCAP-15942-P-A[19]. For AREVA Quad Cities Unit 1 ATRIUM 10XM fuel, see ANP-3305P Rev. 2 (Reference 27), Section 3.3.5 for the applicable design limits.

4.2.3.1.6.3 Evaluations

For SVEA-96 Optima2 fuel, the design of the assembly and the material used indicates that rod bowing will be minimal and will not lead to fuel rod contact. The impact of bundle performance is minimal in the absence of fuel rod contact [19]. For AREVA Quad Cities Unit 1 ATRIUM 10XM fuel, see ANP-3305P Rev. 2 (Reference 27), Table 3-1. The applicable design criteria are met.

4.2.3.1.7 Internal Gas Pressure

4.2.3.1.7.1 Bases

The fuel rod is evaluated to ensure that the effects of fuel rod internal pressure during normal steady-state operation will not result in fuel failure due to excessive cladding pressure loading. [4.2-39]

4.2.3.1.7.2 Limits

For Westinghouse fuel, rod internal pressure limits are defined in Section 3.3 of WCAP-15942-P-A [19]. For AREVA Quad Cities Unit 1 ATRIUM 10XM fuel, these criteria are defined in the plant/cycle-dependent fuel rod thermal-mechanical analysis report.

4.2.3.1.7.3 Evaluations

STAV 7.2 [20], analysis of the SVEA-96 Optima2 fuel rod shows that pressure limits are not exceeded during normal operation or AOOs [19]. For AREVA Quad Cities Unit 1 ATRIUM 10XM fuel, these criteria are addressed in the plant/cycle-dependent fuel rod thermal-mechanical analysis report. Resulting applicable fuel operating limits are documented in the plant/cycle-dependent reload safety analysis report.

4.2.3.1.8 Hydraulic Loads

4.2.3.1.8.1 Bases

The fuel assembly is evaluated to ensure that interference sufficient to prevent blade insertion will not occur. [4.2-40]

4.2.3.1.8.2 Limits

For Westinghouse fuel, hydraulic load limits are defined in Section 3.2 of WCAP-15942-P-A [19]. For AREVA Quad Cities Unit 1 ATRIUM 10XM fuel, see ANF-89-98 for the applicable design limits [24].

4.2.3.1.8.3 Evaluations

Hydraulic analysis of the SVEA-96 Optima2 and AREVA Quad Cities Unit 1 ATRIUM 10XM fuel assemblies shows that hydraulic lifting of the fuel assemblies is not expected for normal operation or AOOs [19][27].

4.2.3.1.9 Control Rod Reactivity

Control rod reactivity limits are addressed in Section 4.6.

4.2.3.2 Fuel Rod Failure

Sections 4.2.3.2.1 through 4.2.3.2.3 apply to normal operation; Sections 4.2.3.2.4, 4.2.3.2.5 and 4.2.3.2.7 apply to anticipated operational occurrences; and Sections 4.2.3.2.6, 4.2.3.2.8 and 4.2.3.2.9 apply to postulated accidents. [4.2-41]

4.2.3.2.1 Hydriding

Hydriding is addressed in Section 4.2.3.1.5.

4.2.3.2.2 Cladding Collapse

4.2.3.2.2.1 Bases

The fuel rod is evaluated to ensure that fuel rod failure due to cladding collapse into a fuel column axial gap will not occur. [4.2-42]

4.2.3.2.2.2 Limits

For Westinghouse fuel, cladding collapse limits are defined in Section 3.3 of WCAP-15942-P-A [19]. For AREVA Quad Cities Unit 1 ARIUM 10XM fuel, these criteria are defined in the plant/cycle-dependent fuel rod thermal-mechanical analysis report.

4.2.3.2.2.3 Evaluations

COLLAPS - 3.3D [20] analysis of the SVEA-96 Optima2 fuel rod shows that cladding collapse is not expected to occur during normal operation or AOOs [19]. For AREVA Quad Cities Unit 1 ATRIUM 10XM fuel, these criteria are addressed in the plant/cycle-dependent fuel rod thermal-mechanical analysis report. Resulting applicable fuel operating limits are documented in the plant/cycle-dependent reload safety analysis report.

4.2.3.2.3 Fretting Wear

Fretting wear is addressed in Section 4.2.3.1.3.

4.2.3.2.4 Overheating of Cladding

Overheating of the cladding is addressed in Section 4.4.

4.2.3.2.5 Overheating of Pellets

4.2.3.2.5.1 Bases

The fuel rod is evaluated to ensure that fuel rod failure due to fuel melting will not occur. [4.2-43]

4.2.3.2.5.2 Limits

For Westinghouse fuel, cladding rod melting limits are described in Section A, part 3.4.9 of WCAP-15942-P-A [19]. The Thermal Mechanical Operating Limit (TMOL) protecting the fuel from centerline melting, is specified in the Core Operating Limits Report (COLR) for each operating cycle. For AREVA Quad Cities Unit 1 ATRIUM 10XM fuel, these criteria are defined in the plant/cycle-dependent fuel rod thermal-mechanical analysis report.

4.2.3.2.5.3 Evaluations

Westinghouse Optima2 fuel is evaluated both during normal operation and AOOs, to compare those temperatures to the melting temperature of the limiting fuel pellets. Operation of the fuel within the TMOL specified in the COLR assures that the centerline melting will not occur during normal operation and AOOs. For AREVA Quad Cities Unit 1 ATRIUM 10XM fuel, these criteria are addressed in the plant/cycle-dependent fuel rod thermal-mechanical analysis report. Resulting applicable fuel operating limits are documented in the plant/cycle-dependent reload safety analysis report.

4.2.3.2.6 Excessive Fuel Enthalpy

The potential for excessive fuel enthalpy is addressed in the control rod drop accident (CRDA) discussions in Section 4.3.2.1.4 and Section 15.4.10.

4.2.3.2.7 Pellet-Cladding Interaction

4.2.3.2.7.1 Bases

The fuel rods are evaluated to ensure that fuel rod failure due to pellet-cladding mechanical interaction will not occur. [4.2-44]

4.2.3.2.7.2 Limits

For Westinghouse fuel, pellet-clad interaction is discussed in Section 4.3.11 of WCAP-15942-P-A [19]. For AREVA Quad Cities Unit 1 ATRIUM 10XM fuel, these criteria are defined in the plant/cycle-dependent fuel rod thermal-mechanical analysis report.

4.2.3.2.7.3 Evaluations

SVEA-96 Optima2 fuel is not expected to suffer pellet-cladding interaction damage as long as the thermal-mechanical operating limit identified in the Core Operating Limits Report is not exceeded. For AREVA Quad Cities Unit 1 ATRIUM 10XM fuel, these criteria are addressed in the plant/cycle-dependent fuel rod thermal-mechanical analysis report. Resulting applicable fuel operating limits are documented in the plant/cycle-dependent reload safety analysis report.

4.2.3.2.8 Bursting

The potential for cladding perforation is addressed in the LOCA discussions in Section 15.6.5.

4.2.3.2.9 Mechanical Fracturing

4.2.3.2.9.1 Bases

The fuel assembly is evaluated under design basis earthquake and LOCA loading conditions to ensure that loss of fuel assembly coolability, and interference to the degree that control blade insertion is prevented, will not occur. [4.2-45]

4.2.3.2.9.2 Limits

For Westinghouse Optima2 fuel, satisfactory performance is established as part of the plant seismic/LOCA evaluation CENDP-288-P-A [21]. For AREVA Quad Cities Unit 1 ATRIUM 10XM fuel, Section 3.4.4 of ANP-3305P Rev. 2 (Reference 27) describes how the deformations or stresses from postulated accidents (including design basis earthquake and LOCA loading conditions) are limiting according to requirements of ASME code and SRP Section 4.2. The limits for each ATRIUM 10XM structural component are derived from analyses and/or component load tests.

4.2.3.2.9.3 Evaluations

For Westinghouse SVEA-96 Optima2 fuel, the analyses reported in the UFSAR Chapter 15 and the seismic/LOCA Report [21] demonstrate the ability of the SVEA-96 Optima2 fuel to withstand the combined event. For AREVA Quad Cities Unit 1 ATRIUM 10XM fuel, Table 3-1 of ANP-3305P Rev. 2 (Reference 27) shows that the results of the fuel assembly and fuel channel meet the applicable postulated accident criteria (including design basis earthquake and LOCA loading conditions). SVEA-96 Optima2 fuel and AREVA ATRIUM 10XM fuel are not expected to violate fuel coolability or control rod interference limits under normal operation, AOOs, or postulated accidents.

4.2.3.3 Fuel Coolability

This Section applies to postulated accidents. [4.2-46]

4.2.3.3.1 Cladding Embrittlement

The potential for cladding perforation is addressed in the LOCA discussions in Section 15.6.5.

4.2.3.3.2 Violent Expulsion of Fuel

The potential for violent expulsion of fuel is addressed in the CRDA discussions in Section 4.3.2.1.4 and Section 15.4.10.

4.2.3.3.3 Generalized Cladding Melting

Generalized cladding melting is bounded by the cladding embrittlement criteria of Section 4.2.3.3.1. [4.2-47]

4.2.3.3.4 Fuel Rod Ballooning

The potential for cladding perforation is addressed in the LOCA discussions in Section 15.6.5.

4.2.3.3.5 Structural Deformation

Structural deformation is addressed in Section 4.2.3.2.9.

4.2.3.4 Safeguard Aspects of Burnable Neutron Absorber

4.2.3.4.1 Thermal Margin

The thermal margin is defined by the difference between the design operating values and the steady-state operating limits which are provided in the Core Operating Limits Report for each fuel cycle. The reactivity available for power shaping by control rods is more than adequate, and there should be no difficulty in satisfying the established local limits throughout the cycle at the maximum power level anticipated. [4.2-48]

4.2.3.4.2 Shutdown Margin

The primary design requirement for the gadolinia reactivity effect is that it produces an adequate shutdown margin. Thus, for design purposes, it is required that k_{eff} does not exceed the design limit stated in Section 4.3.2.1.3 with the control rod of maximum worth fully withdrawn and all others fully inserted, for a core temperature and an exposure chosen to maximize k_{eff}. [4.2-49]

The shutdown margin is determined by using the BWR simulator code (see Section 4.3.3) to calculate the core multiplication at selected exposure points with the maximum worth rod fully withdrawn. The shutdown margin value is obtained by subtracting the controlled k_{eff} with the maximum worth rod fully withdrawn from the critical k_{eff}. [4.2-50]

Items which may affect the evaluation of reactor shutdown margin for a given cycle are: [4.2-51]

- For reloads, an early termination or an extension of the previous cycle; and
- The inability to insert a control rod to the fully-inserted position.

4.2.3.4.3 Transients and Excursions

Results of the transients and excursions remain unchanged with the replacement of absorber curtains (previously used in other BWRs) by a gadolinia burnable absorber. Of primary importance in the rapid excursions is the presence of a strong Doppler effect to compensate for the excess reactivity input. For slower transients, the moderator void coefficient assumes a major role. None of the reactivity coefficients associated with the fuel lattice are materially affected by the change in control augmentation method from curtains to gadolinia absorber. [4.2-52]

Gadolinia is held in solid solution by the UO_2. Thus the initially chemically uniform gadolinia-bearing pellets remain so at all exposures, because neutron absorption in Gd-155 or -157 produces stable isotopes Gd-156 and -158, respectively. Initially, the gadolinia is highly self-shielded. During irradiation, the isotopic distribution varies radially in the pellet, nearly as a step function with an essentially zero concentration of Gd-155 and -157 outside a certain radius and a natural percentage of these isotopes inside that radius. Because no chemical concentration gradients exist in the pellet, net migration of gadolinia in normal temperature gradients has not been detected in any of the post-irradiation examinations to date. Either the migration does not occur or is limited to amounts below the detection threshold. Any dispersal of the solid solution into the moderator caused by an excursion would only reduce the self-shielding, causing an increase in the local neutron absorption and producing a loss in reactivity. There seems to be no mechanism which could cause the control effectiveness of the gadolinia to vary in such a way as to compromise safety.

4.2.3.4.4 Absorber Omission and Fuel Loading Errors

The safety effect of the omission of gadolinia during fuel fabrication, and the consequences of fuel assembly misorientation during initial fuel loading, have been considered. [4.2-53]

Quality control procedures assure production of fuel in conformance with design to prevent the inadvertent omission of gadolinia. The inspections performed on gadolinia-bearing fuel are described in Section 4.2.4.3. The safety aspects of fuel storage are addressed in Section 9.1. In the event of substantial gadolinia omission from the fuel assemblies, the shutdown margin check would expose the abnormal condition.

The second procedural error which has been studied is the misorientation of a fuel assembly. Design features of the upper casting make it possible to confirm visually that fuel orientation is correct, as discussed in Section 4.2.2.1. If, however, through an error, a fuel assembly were installed with a 90° or 180° rotation from the proper location, no fuel damage would be expected to occur during the subsequent power operation, even if the misoriented assembly were operating at the maximum permitted power. For Westinghouse reload cores, the maximum fuel rod power is calculated during each cycle to assure that MCPR remains greater than or equal to the Fuel Cladding Integrity Safety Limit MCPR during normal operation and for all Anticipated Operational Occurrences.

For AREVA reload cores, a similar process is used to calculate a delta CPR due to the misorientation event with the expectation that the SLMCPR will remain protected. However, AREVA's NRC approved methodology identified the fuel loading error, which includes assembly misorientation as an infrequent event. The acceptance criterion for this event is that a small fraction of the 10 CFR 50.67 offsite dose criteria be conservatively satisfied. Consequently, if fuel integrity cannot be confirmed then a dose evaluation may be used to verify the underlying criterion is met.

4.2.4 Testing and Inspection Plan

4.2.4.1 Fuel Assemblies

Rigid quality control requirements are enforced at every stage of fuel manufacturing to ensure that the design specifications are met. Written manufacturing procedures and quality control plans define the steps in the manufacturing process. Fuel cladding is subjected to 100% dimensional inspection and ultrasonic inspection to reveal defects in the cladding wall. Destructive tests are performed on representative samples from each lot of tubing. Integrity of end plug welds is assured by standardization of weld processes based on radiographic and metallographic inspection of welds, and by the helium leak test of completed fuel rods. The UO_{2} powder characteristics, and pellet densities, composition, and surface finish, are controlled by regular sampling inspections. Sample dimensional measurements and visual inspections of critical areas such as fuel rod-to-rod clearances are performed after assembly and after arrival at the reactor site. [4.2-54]

Flow tests were conducted using prototype reactor hardware, in which the single-phase and two-phase flow characteristics were determined for core and vessel internal components which contribute to the core pressure drop. Fuel assembly handling tests were done to verify structural integrity. Mechanical tests and corrosion tests of the Zircaloy spacers were performed to identify design and specification requirements. Critical power tests were performed using prototype multi-rod configurations, minimum allowable coolant clearance gaps, and prototype Zircaloy spacers. Further descriptions of testing, inspection, and surveillance of new SVEA-96 Optima2 fuel can be found in Reference 19.

4.2.4.2 Control Rods

Testing and inspection for control rods are addressed in Section 4.6.

4.2.4.3 Burnable Absorber Bearing Rods

The same rigid quality control requirements observed for standard UO_2 fuel are employed in the manufacture of gadolinia-urania fuel. Gadolinia-bearing UO_2 fuel pellets of a given enrichment and gadolinia concentration are maintained in separate groups throughout the manufacturing process. The percent enrichment and gadolinia concentration characterizing a pellet group is denoted by an identification stamp on the pellet. [4.2-55]

Fuel rods are individually numbered prior to loading of fuel pellets. This is done to:

A. Identify which pellet group is to be loaded in each fuel rod;

B. Identify which position in the fuel assembly each fuel rod is to be loaded; and

C. Facilitate total fuel material accountability for a given project.

The following quality control inspections are made:

A. Gadolinia concentration in the gadolinia-uranium dioxide powder blend is verified.

B. Sintered pellet UO_2-Gd_2O_3 solid-solution homogeneity across a fuel pellet is verified by examination of metallographic specimens.

C. Gadolinia-uranium dioxide pellet identification is verified to confirm proper enrichment and gadolinia content.

D. Gadolinia-uranium dioxide fuel rod identification is checked.

Finally, an inspection is made of all assemblies and rods of a given project to assure overall accountability of fuel quantity and placement for the project.

4.2.5 References

1. Deleted.

2. Deleted.

3. "Power Cycling of Zirconium Barrier Fuel At Exposures of 22 and 44 MWd/kg U - Final Report," December 1989, General Electric Company, NEDM-31383P-6.

4. Deleted.

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17. Deleted.

18. GE Proprietary Report, NEDC-31602-15P, January 1996.

19. WCAP-15942-P-A, “Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors Supplement 1 to CENPD-287,” March 2006.

20. WCAP-15836-P-A, "Fuel Rod Design Methods for Boiling Water Reactors – Supplement 1," April 2006.

21. CENPD-288-P-A, "ABB Seismic/LOCA Evaluation for Boiling Water Reactor Fuel," July 1996.

22. Westinghouse Proprietary Report CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel," July 1996.

23. BAW-10247PA Revision 0, “Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors,” AREVA NP Inc., February 2008, and BAW-10247PA Revision 0 Errata 1P-001, (same title), AREVA Inc., June 2016.

24. ANF-89-98(P)(A) Revision 1 and Supplement 1, “Generic Mechanical Design Criteria for BWR Fuel Designs,” Advanced Nuclear Fuels Corporation, May 1995.

25. ANP-2899P Revision 0, “Fuel Design Evaluation for ATRIUM 10XM BWR Reload Fuel,” AREVA NP Inc., April 2010.

26. EMF-2158(P)(A) Revision 0, “Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation for CASMO-4/MICROBURN- B2, Siemens Power Corporation, October 1999.

27. ANP-3305P Revision 2, “Mechanical Design Report for Quad Cities and Dresden ATRIUM 10XM Fuel Assemblies,” AREVA Inc., August 2016.

28. ANP-10298P-A Revision 1, “ACE/ATRIUM 10M Critical Power Correlation,” AREVA, March 2014.

29. EMF-2209(P)(A), Revision 3, “SPCB Critical Power Correlation,” AREVA NP, September 2009.

30. EMF-2245(P)(A) Revision 0, “Application of Siemens Power Corporation’s Critical Power Correlations to Co-Resident Fuel,” Siemens Power Corporation, August 2000.

31. ANP-10307PA Revision 0, “AREVA MCPR Safety Limit Methodology for Boiling Water Reactors,” AREVA NP, June 2011.

32. Safety Evaluation By The Office of Nuclear Reactor Regulation Related To Amendment No. 251 To Renewed Facility Operating License No. DPR-19, Amendment No. 244 To Renewed Facility Operating License No. DPR-25, Amendment No. 264 To Renewed Facility Operating License No. DPR-29, and Amendment No. 259 To Renewed Facility Operating License No. DPR-30, Exelon Generation Company, LLC and MidAmerican Energy Company Dresden Nuclear Power Station Unit Nos. 1 and 2, Docket nos. 50-237, 50-249, 50-254, and 50-265, October 20, 2016.

4.3 NUCLEAR DESIGN

This section addresses the nuclear design bases (Section 4.3.1), the steady-state and dynamic nuclear characteristics of the core (Sections 4.3.2.1 and 4.3.2.2 respectively), stability (Section 4.3.2.3), nuclear design analytical methods (Section 4.3.3), and the impacts of new fuel types and new stability concerns (Section 4.3.4).

4.3.1 Design Bases

The bases for nuclear design include the nuclear design bases for the reactor and the design bases for system stability, which are discussed in Sections 4.3.1.1 and 4.3.1.2, respectively.

4.3.1.1 Reactor

A summary of the design bases for the nuclear (reactivity) aspects of the reactor is presented as follows: [4.3.1]

A. The core shall be capable of being made subcritical at any time, at any core condition, with the highest worth control rod fully withdrawn.

B. The overall moderator void coefficient is always negative over the entire operating range.

C. Reactivity coefficients generally have:

   1. A strong negative reactivity feedback under severe transient conditions consistent with the requirements of overall plant nuclear-hydrodynamic stability; and

   2. A reactivity response which regulates or damps changes in core power level and spatial distribution in a manner consistent with safe and efficient plant operation.

D. The design bases for gadolinia in the fuel are addressed in Section 4.2.1.3.

4.3.1.2 System Stability

Stability criteria are established to demonstrate compliance with 10 CFR 50, Appendix A, General Design Criterion (GDC) 12. The requirement of GDC 12 is that one of the following alternatives be satisfied: [4.3.2]

A. Power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible; or

B. Such power oscillations can be reliably and readily detected and suppressed

The plant was originally designed to satisfy the first alternative; it was expected to be stable during all normal modes of operation, i.e., it would have no inherent tendency toward oscillations of power and channel flow, either divergent or of limited amplitude. The units were also designed to be well damped over all normal modes of power and flow. These design bases were applied in the stability evaluation described in Section 4.3.2.3. [4.3.3]

However, since 1988, industry experience (see Section 4.3.4.2) indicated that this type of plant may not be stable under all operating conditions during a fuel cycle as originally expected. Core design and operating criteria were then modified to concentrate on satisfying the latter alternative of GDC 12. For Westinghouse licensed reload cores the stability criteria for power operation with possible limit-cycle oscillations are discussed in CENPD-300-P-A [28] Section 9.2 (for Westinghouse method). Fuel design thermal hydraulic compatibility is addressed in References 29 and 30. [4.3.4] For AREVA licensed reload cores (staring with Quad Cities Unit 1 Cycle 25), the stability criteria, reload-specific analyses, and establishment of appropriate operating limits/setpoints are described in Section 4.3.4.2. The fuel design thermal hydraulic compatibility is addressed in Reference 36.

4.3.2 Description

Quad Cities Units 1 and 2 reactors are light-water moderated boiling water reactors (BWRs), fueled with slightly enriched uranium dioxide (UO_2). At operating conditions, the moderator is permitted to boil, producing a spatially varying density of steam voids within the core. The use of water as moderator produces a neutron energy spectrum such that the fissions are produced principally by thermal neutrons. Gadolinia (Gd_2O_3) is used as a burnable neutron absorber in the fuel, as discussed in Section 4.2. [4.3.5]

BWR fuel assemblies are hydraulically isolated from each other by Zircaloy channels, which allow flow mixing only outside the active fuel region of the core. Evaluation of core thermal-hydraulic stability and operational provisions to assure compliance with stability design bases are discussed in Section 4.3.2.3.2 and 4.3.4.2.

Burnup capability of UO_2 fuel is extended through the use of gadolinia Gd_2O_3 burnable absorber material in the fuel rods, which absorbs neutrons early in life to reduce assembly reactivity and is depleted through neutron absorption to allow access to relatively fresh fissile material at the end of the first cycle.

Gadolinia in varying amounts is present in all fuel assemblies. An improvement in the transverse power flattening can be expected by placing assemblies with larger gadolinia contents in the central zone. The burnable absorber is mixed uniformly with UO_2 in some fuel pellets in selected lengths of a few rods in the fuel assemblies. The absorber is initially in a highly self-shielded configuration, leading to a more linear rate of change of its control effect than would be the case for lower concentrations in more rods per assembly.

The following are the observed effects of the addition of small amounts of Gd_2O_3 to UO_2. Both the conductivity and melting temperature of the mixture are affected. Below 1800°C the conductivity is reduced, but above 1800°C there is essentially no effect. At no temperature is the conductivity of the mixture less than the minimum conductivity of pure UO_2. Melting temperature of the mixture is below that of pure UO_2. However, the combined effect of these changes does not cause the fuel rods containing gadolinia to approach a molten condition at core power outputs which would normally cause melting in the highest powered pure UO_2 rods.

During postulated severe transients such as a control rod drop, there would be no melting of the Gd_2O_2-UO_2 mixture at exposures low enough for the gadolinia to have an appreciable control effect. This is a result of the very low relative power of the gadolinia-bearing rods. Because the severe transients are rapid and of short duration, and because the fuel rods containing the control material do not sustain melting, changes in the control worth of the gadolinia due to migration inside the pellet would not occur. In postulated excursions more severe than those employed as a design basis, any dispersal of the gadolinia-bearing UO_2 into the water would increase its control effect due to the reduction in self-shielding.

The presence of U-238 in the uranium dioxide fuel has several major effects. First, it leads to the production of significant quantities of plutonium during core operation. This plutonium contributes both to fuel reactivity and to power production of the reactor, and changes the delayed neutron fraction with exposure. (The neutron lifetime during hot operating conditions is approximately 40 microseconds and is relatively unaffected by fuel exposure. The infinite lattice effective delayed neutron fraction decreases from 0.0072 at zero exposure to approximately 0.0056 at 10,000 MWd/t, where "t" denotes short ton.) In addition, the direct fissioning of U-238 by fast neutrons yields approximately 7% of the total power. Finally, the U-238 contributes a strong negative Doppler coefficient of reactivity, which improves the inherent or self-regulating response of the reactor and limits the peak power in excursions.

The strong negative moderator void reactivity effect contributes to the overall plant thermal hydraulic stability and to the damping of xenon oscillations.

4.3.2.1 Core Steady-State Characteristics

The design of the BWR core and fuel is based on a proper combination of design variables, such as moderator-to-fuel volume ratio, core power density, thermal hydraulic characteristics, fuel exposure level, nuclear characteristics of the core and fuel, heat transfer parameters, flow distribution, moderator void content, heat flux, and operating pressure. These variables are dynamic functions of operating conditions. However, design analyses and calculations verified by comparison with data from operating plants are usually performed for specific steady-state conditions. This section addresses steady-state analyses for the fuel cycle, local power peaking, reactivity control, and control rod worths. [4.3-6]

4.3.2.1.1 Fuel Cycle

The information on each new fuel cycle is provided in the corresponding cycle-specific licensing documents. [4.6-7]

For each reload core, the enrichments in the fuel assemblies are chosen to provide excess reactivity sufficient to overcome the negative reactivity effects of core neutron leakage, moderator heating and boiling, fuel temperature rise, xenon and samarium poisoning, and fuel depletion. During fuel burnup, control rods are used in part to counteract the power distribution effect of steam voids. The combined use of both the control rod and moderator void distributions provides the BWR design with considerable flexibility to control power distribution. Power distributions during the course of the cycle cause changes in fuel burnup and isotopic composition throughout the core, which in turn affect the reactivity and power distributions later in the cycle. This phenomenon permits using the control rods to shape the fuel burnup and isotopic composition early in the fuel cycle to counteract the effect of moderator voids on the axial distribution toward the end of a fuel cycle, when few control rods remain in the core. [4.3-8]

4.3.2.1.2 Local Power Peaking

The enrichment distribution is selected to reduce the relative peak-to-average fuel rod powers within each assembly in locations where the axial power peak is expected to occur. The enrichment distribution for the reload fuel assemblies are provided in cycle specific bundle design reports. Design practices include the generation of fuel bundle nuclear configurations for each application within the mechanical parameters of the generic design analysis reported in Section 4.2. Fuel rod enrichment and burnable absorber loadings are designed for each operating cycle. [4.3-9]

4.3.2.1.3 Reactivity Control

The excess reactivity designed into the core is controlled by a control rod system supplemented by burnable neutron absorbers in the fuel. [4.3-10]

The control rod system is designed to provide adequate control of the maximum excess reactivity anticipated during the fuel cycle operation. Shutdown capability is evaluated assuming a xenon-free core at ambient temperature, which represents the condition of maximum fuel reactivity.

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition. [4.3-11]

The SHUTDOWN MARGIN limitation is a restriction to be applied principally to a new refueling pattern. Satisfaction of the limitation must be determined at the time of loading and must be such that it will apply to the entire subsequent fuel cycle. This determination is provided by core design calculations and administrative control of fuel loading patterns. These procedures include restrictions to allow only those intermediate fuel assembly configurations that have been shown to provide the required SHUTDOWN MARGIN.

Since core reactivity values will vary through core life as a function of fuel depletion and poison burnup, the demonstration of SHUTDOWN MARGIN will be performed during xenon free conditions and adjusted to the most reactive temperature $\geq 68^{o}$F to accommodate the current moderator temperature. The generalized form is that the reactivity of the core loading will be limited so the core can be made subcritical by at least R + 0.38% Δ-k/k or R + 0.28% Δ-k/k, as appropriate, with the strongest control rod fully withdrawn and all others fully inserted. Two different values are supplied in the Limiting Condition for Operation to provide for the different methods of determination of the highest control rod worth, either analytically or by test. This is due to the reduced uncertainty in the SHUTDOWN MARGIN test when the highest worth control rod is determined by demonstration. When SHUTDOWN MARGIN is determined by calculations not associated with a test, additional margin must be added to the specified SHUTDOWN MARGIN limit to account for uncertainties in the calculation.

The value of R in units of % Δk/k is the difference between the calculated beginning-of-life core reactivity, at the beginning of the operating cycle, and the calculated value of maximum core reactivity at any time later in the operating cycle, where it would be greater than at the beginning. R must be a positive quantity or zero and a new value of R must be determined for each new fuel cycle.

The value of % Δk/k in the above expression is provided as a finite, demonstrable, subcriticality margin. This margin is verified using an in-sequence control rod withdrawal at the beginning-of-life fuel cycle conditions. This assures subcriticality with not only the strongest fully withdrawn but at least an R + 0.28% Δk/k (or 0.38% Δk/k) margin beyond this condition. This reactivity characteristic has been a basic assumption in the analysis of plant performance and can be best demonstrated at the time of fuel loading, but the margin must also be determined anytime a control rod is incapable of insertion following a scram signal. Any control rod that is immovable as a result of excessive friction or mechanical interference, or is known to be unscrammable, per Specification 3.1.3, is considered to be incapable of insertion following a scram signal. It is important to note that a control rod can be electrically immovable, but scrammable, and no increase in SHUTDOWN MARGIN is required for these control rods.

During MODE 5, adequate SDM is required to ensure that the reactor does not reach criticality during control rod withdrawals. An evaluation of each in-vessel fuel movement during fuel loading (including shuffling fuel within the core) is required to ensure adequate SDM is maintained during refueling. This evaluation ensures that the intermediate loading patterns are bounded by the safety analyses for the final core loading pattern. For example, bounding analyses that demonstrate adequate SDM for the most reactive configurations during the refueling may be performed to demonstrate acceptability of the entire fuel movement sequence. These bounding analyses include additional margins to the associated uncertainties. Spiral offload/reload sequences inherently satisfy the SDM, provided the fuel assemblies are reloaded in the same configuration analyzed for the new cycle. Removing fuel from the core will always result in an increase in SDM.

In order to assure that the basic design criterion and the LCO ($k_{eff} < 1$) are always satisfied, a 0.01 Δk design margin is adopted. Thus, the design goal is $k_{eff} \le 0.99$ with the rod of highest worth fully withdrawn. This target assures that the unit can be shut down by control rods alone. This means reload cores are designed with margin to assure adequate shutdown margin capability throughout the cycle; the shutdown margin design goal is (at least) 1% Δk. [4.3-12]

In addition to the control rod shutdown requirements, the standby liquid control system (Section 9.3.5) provides sufficient reactivity control to shut down the unit at any time independently of control rod action.

The concentration of gadolinia in the fuel assemblies is selected so that the high cross-section isotopes, Gd-155 and Gd-157, will be substantially depleted by the end of each cycle. The irradiation products of this process are other gadolinium isotopes having low cross-sections. Thus, the control augmentation effect diminishes on a predetermined schedule without changes in the chemical composition of the fuel or the physical makeup of the core.

4.3.2.1.4 Control Rod Worth

In an operating reactor there is a spectrum of possible control rod worths, depending on the reactor state and on the control rod patterns chosen for operation. Control rod withdrawal sequences and patterns are selected prior to reactor startup to achieve optimal core performance, and simultaneously, low individual rod and notch worths. [4.3-13]

The fractional control rod density versus average moderator density for the beginning of life conditions is shown in Figure 4.3-1. This density distribution is based on control rod patterns that provide optimal power distributions and minimum rod worths.

Distributed control rod patterns are used. The operating procedures to establish such patterns are supplemented by the rod worth minimizer (RWM), which prevents rod withdrawals that would yield a rod worth greater than that permitted by the prescribed rod withdrawal sequence. The RWM is described in the following paragraphs and in Section 7.7.

[START HISTORICAL INFORMATION]

Figure 4.3-2 shows the maximum of three classes of rod worths as a function of moderator density for fresh fuel. The upper curve is an envelope of the maximum possible control rod worths with multiple errors in the control rod withdrawal sequence. The central curve is an envelope of the maximum rod worths with a single error in the withdrawal sequence. The lower curve is an envelope of the maximum rod worths with a normal (error free) withdrawal sequence. A peak value of less than 0.01 Δk is seen on this envelope. Only a few rods have worths as high as the envelope; typical rod worths are about half the magnitude of the envelope. Operating procedures, supplemented by the RWM, prevent the withdrawal of rods with worths higher than the lower curve of Figure 4.3-2.

[END HISTORICAL INFORMATION]

Most BWR reactivity insertion scenarios result in a relatively slow rate of reactivity insertion and pose no threat to the reactor system. However, the rapid removal of a high worth control rod could result in a potentially significant power excursion. The accident which has been chosen to encompass the consequences of reactivity excursions is the control rod drop accident (CRDA), i.e., the free fall of any rod to the position of its drive. [4.3-14]

In light-water reactors with UO_2 fuel, a large fraction of the energy generated during a rapid power excursion is deposited in the fuel and then released to the rest of the system through heat transfer mechanisms. If the fuel energy densities are high enough, there may be extensive fuel melting, causing a rapid increase in internal fuel rod pressure, followed by rapid fragmentation and dispersal of fuel and cladding into the coolant. Consequently, rapid heat transfer would occur from finely dispersed molten UO_2 to the coolant. The associated conversion of heat to mechanical energy, in sufficient quantity, could disarrange the reactor core or breach the primary coolant system.

Available experimental information concerning fuel failure thresholds indicates that the immediate fuel dispersal threshold occurs at a fuel enthalpy content of about 425 cal/g. [4.3-15]

In general, failure consequences for UO_2 fuel are insignificant below 300 cal/g for both irradiated and unirradiated fuel rods. Therefore, a limit of 280 cal/g on the peak fuel enthalpy, i.e., the maximum calculated radial average energy density at any axial fuel location in any fuel rod, ensures that core damage resulting from any postulated reactivity excursion would be minimal and that both short-term and long-term core cooling capability would not be impaired. (Regulatory Guide 1.77, May 1974). [4.3-16]

To implement the preceding limitation, the control rod withdrawal sequences are designed to limit rod worths so that a CRDA would result in a peak fuel enthalpy of not more than 280 cal/g. Generic CRDA analyses indicated that peak fuel enthalpies would not exceed 280 cal/g if the rod worths were kept within certain limits. See Section 15.4 for details of analysis of this event. [4.3-17]

The effects of fuel densification are addressed in Section 4.2.1.1. Analyses by Westinghouse and AREVA have indicated that the CRDA analyses remain valid even with the additional consideration of fuel densification. [4.3-18]

Quad Cities Units 1 and 2 use the RWM, in conjunction with sequences designed to ensure that a postulated CRDA will not result in a peak fuel enthalpy greater than 280 cal/g, for controlling the rod patterns below 10% rated power. Above 10% power, even multiple operator errors will not create rod worths high enough to produce a peak fuel enthalpy of 280 cal/g[35]. Therefore, the RWM is not required to be operable when the reactor is above 10% power. For the above analyses, it is assumed that the control rod drops at the maximum velocity determined from experimental data (99.9% confidence limit). [4.3-19]

Control rod drop analyses are discussed in Section 15.4.10.

4.3.2.2 Core Nuclear Dynamic Characteristics

The performance characteristics of the core provide a nuclear dynamic response which: [4.3-20]

A. Has a strong negative reactivity feedback under severe transient conditions;

B. Contributes negative reactivity feedback consistent with the requirements of overall plant nuclear-hydrodynamic stability; and

C. Regulates or damps changes in core power level and spatial distribution in a manner consistent with safe and efficient plant operation.

Characteristic A, through the Doppler and moderator reactivity coefficients which are negative during power operation, provides shutdown mechanisms in the event of a power excursion. Characteristics B and C assure that there are no inherent tendencies for undamped oscillations during normal modes of operation.

4.3.2.2.1 Reactivity Coefficients

The discussion of reactivity coefficients in this section is specifically directed to GE methods and results and is presented for illustrative purposes.

The dynamic behavior of the core is characterized in terms of the following reactivity coefficients: [4.3-21]

A. The fuel temperature or Doppler coefficient,

B. The moderator temperature coefficient, and

C. The moderator void coefficient.

These three coefficients are collectively termed the power coefficient.

4.3.2.2.1.1 Doppler Coefficient

The Doppler coefficient provides an inherent mechanism for terminating nuclear transients and is of prime importance in reactor safety. The Doppler coefficient is a measure of the reactivity change associated with an increase in the absorption of resonance-energy neutrons caused by a change in the temperature of the absorbing material. For most structural and moderator materials, this effect is not significant, but in U-238 and Pu-240, an increase in temperature produces a comparatively large increase in the neutron absorption cross-section, causing a significant decrease in reactivity. The Doppler reactivity coefficient provides instantaneous negative reactivity feedback to any rise in fuel temperature, on either a gross or local basis. In BWR fuel, approximately 96% of the uranium in UO_2 is U-238. The magnitude of the Doppler coefficient is therefore inherent in the fuel design and does not vary significantly among BWR designs. [4.3-22]

Although the reactivity change caused by the Doppler effect is small compared to other power-related reactivity changes during normal operation, it becomes very important during postulated rapid power excursions in which large fuel temperature changes occur. The most severe postulated power excursions are associated with the CRDA. Local Doppler feedback resulting from the associated temperature rise of 3000 — 5000°F is available to terminate the power excursion.

[START HISTORICAL INFORMATION]

The Doppler reactivity decrement is derived directly from the lattice calculations by performing two calculations using two different fuel temperatures but otherwise identical input parameters. The results are used to determine the proportionality constant C_{DOP} from: [4.3-23]

$$\left(\frac{\Delta k}{k}\right)_{DOP} = C_{DOP}\left(\sqrt{T} - \sqrt{T_o}\right)$$

where:

$$\left(\frac{\Delta k}{k}\right)_{DOP} = \frac{k - k_o}{k_o}$$

and k and k_o are the neutron multiplication factors at the fuel temperatures T and T_o, respectively.

The Doppler coefficient is then generated using:

$$\text{Doppler Coefficient} = \left[\frac{1}{k}\frac{dk}{dT}\right]_{DOP} = \frac{C_{DOP}}{\left[1 + C_{DOP}\left(\sqrt{T} - \sqrt{T_o}\right)\right] 2\sqrt{T}}$$

[END HISTORICAL INFORMATION]

The effectiveness of the Doppler coefficient in terminating rapid reactivity transients can be seen in the analysis of the rod withdrawal error incident in Section 15.4.2, cold water insertion accidents in Sections 15.1.1 and 15.5.1, and the CRDA in Section 15.4.10. [4.3-24]

Figures 4.3-3 and 4.3-4 show typical regions of Doppler coefficient versus temperature. Figure 4.3-3 is for an exposure of 200 MWd/t and Figure 4.3-4 is for an exposure of 15,000 MWd/t.

4.3.2.2.1.2 Moderator Temperature Coefficient

The moderator temperature coefficient represents a change in the moderating power of the water associated with a change in water temperature. The moderator temperature coefficient is of operational importance primarily in the reactor startup range. Once the reactor reaches the power-producing range, boiling begins and the moderator temperature remains essentially constant. The moderator temperature coefficient is negative during power operation.

The values of moderator temperature coefficient encountered in current BWR lattices are insignificant from a safety point of view. Typically, the moderator temperature coefficient ranges from +4 x 10^{-5} Δk/k°F to -14 x 10^{-5} Δk/k°F during startup and power operation conditions, depending on the base temperature and core exposure. The small magnitude of this coefficient relative to that associated with steam voids, combined with the relatively long time-constant associated with the transfer of heat from the fuel to the coolant, makes the reactivity contribution of the moderator temperature coefficients small.

Current core design criteria do not impose explicit limits on the value of the moderator temperature coefficient. Design control over the moderator temperature coefficient is exercised by applying a design limit to the moderator void coefficient. This limit places a control over the water-to-fuel ratio of the lattice, which effectively limits the moderator temperature coefficient.

[START HISTORICAL INFORMATION]

The moderator temperature coefficient is mathematically represented by: [4.3-25]

$$\text{Moderator Temperature Coefficient} = \frac{1}{k_{eff}}\frac{dk_{eff}}{dT}$$

where:

T = Average moderator temperature
k_{eff} = Effective neutron multiplication factor

[END HISTORICAL INFORMATION]

The moderator temperature coefficient becomes less negative with fuel depletion, reaching its least negative value at the end of each fuel cycle. The principal cause of the positive trend of the moderator temperature coefficient with core life is the reduction in control rod fraction toward the end of cycle.

4.3.2.2.1.3 Moderator Void Coefficient

Moderator voids provide a large inherent reactivity feedback mechanism. In the fuel design, the region inside the fuel channel is undermoderated which results in a negative moderator void coefficient. This means that an increase in moderator void content inside the fuel channel decreases the reactivity. This reactivity decrease results from a reduction in neutron slowing-down (or moderation) due to the decrease in the water-to-fuel ratio. This negative void coefficient: [4.3-26]

A. Permits control of core power through core flow regulation;

B. Flattens the core radial power distribution;

C. Prevents oscillation due to spatial xenon changes; and

D. Provides an inherent mechanism for limiting core power increases.

[START HISTORICAL INFORMATION]

The moderator void coefficient is mathematically represented by: [4.3-27]

$$\text{Moderator Void Coefficient} = \frac{1}{k_{eff}} \frac{dk_{eff}}{dV}$$

where:

V = In-channel moderator void fraction (averaged over the core)

The moderator void coefficient becomes less negative with fuel depletion. The cause is similar to that for the moderator temperature coefficient discussed in Section 4.3.2.2.1.2.

The moderator void coefficient as a function of moderator void fraction can be calculated using a point model. The following equation from NEDO-20964[3] (NEDO-20964 models are representative for BWR core behavior) simulates the total reactor, which is made up of a number of different fuel assembly types (lattice designs) at characteristic exposures: [4.3-28]

$$k_{eff}(E,V) = \frac{\sum_{i=1}^{N} \left[k_{\infty i}^{uc}\ (E_i, V) \cdot VF_i \cdot (1 - CF) + k_{\infty i}^{c}\ (E_i, V) \cdot VF_i \cdot CF\right]}{1 + M_T^2\ (V) \cdot B_g^2\ (V) + \Delta k_f} \qquad (4.3\text{-}1)$$

where:

E_i = Average exposure of the ith fuel type

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V = Average in-channel moderator void fraction in the core

k_∞^{uc} = k_∞ uncontrolled for the ith fuel type

k_∞^{c} = k_∞ controlled for the ith fuel type

CF = Control fraction (controlled fuel assemblies/total fuel assemblies)

N = Number of fuel types for the moderator void coefficient calculation (different batches of the same lattice design may be considered as separate fuel types for the moderator void coefficient calculation if their variations in k_∞ with moderator void fraction are different)

VF_i = Volume fraction of the ith fuel type

M_T^2 = M^2 (migration area) volume-weighted by control fraction and fuel types

B_g^2 = Geometric buckling

Δk_f = Correction factor to preserve criticality

Equation 4.3-1 assumes that each fuel type, controlled and uncontrolled, is at the core-average moderator void fraction so that the k_∞ values are only weighted by relative fuel type volume. Each fuel type is assumed to have the same control fraction.

The moderator void reactivity, or the reactivity change associated with the change from the critical moderator void fraction (V_C) to a moderator void fraction V, can be calculated using:

$$\Delta k(V) = k_{eff}(V) - k_{eff}(V_C)$$

where k_{eff} (V) and k_{eff} (V_C) are calculated using Equation 4.3-1, while holding the control fraction constant for any given critical condition corresponding to V_C. A curve of void reactivity is obtained by parametrically varying V in the neighborhood of the value V_C. The derivative of the void reactivity curve at the critical moderator void fraction is the moderator void coefficient.

Another method is as follows. Equation 4.3-1 is differentiated with respect to V to obtain the moderator void coefficient ($1/k_{eff}$) (dk_{eff}/dV). Since k_{eff} = 1, the ($1/k_{eff}$) term can be disregarded. The resulting equation is:

$$\frac{dk_{eff}}{dV} = \frac{\sum_{i=1}^{N} \frac{dk_{\infty i}^{uc}}{dV} VF_i \left(1-CF\right) + \frac{dk_{\infty i}^{c}}{dV} VF_i \cdot CF}{1+M_T^2(V)\cdot B_g^2(V)+\Delta k_f} + \sum_{i=1}^{N}\left[k_{\infty i}^{uc} VF_i \left(1-CF\right) + k_{\infty i}^{c} VF_i \cdot CF\right]\frac{d}{dV}\frac{1}{1+M_T^2\cdot B_g^2+\Delta k_f} \tag{4.3-2}$$

Lattice data for the point-model simulation is obtained by performing exposure accumulation steps in increments small enough so that the fuel assembly power distributions do not vary significantly from the beginning to the end of the step. Exposure accumulation is performed in the uncontrolled state at a fixed in-channel moderator void fraction (normally 40%), with no void fraction in the water gaps and water rods, at a constant pressure corresponding to saturation conditions at 286°C (546.8°F).

At selected exposures, lattice calculations are performed for in-channel moderator void fraction values of 0% and 70%. The result of the three calculations is k_∞ at three instantaneous moderator void fraction values: 0, average, and 70%. For any exposure E, k_∞ (E, V) is least-squares fit to a quadratic in V; derivatives are then calculated using term-by-term differentiation (see NEDO-20964[3]).

Expressions for controlled k_∞ are determined by adding a control blade to the uncontrolled lattice at the desired exposure intervals. The derivatives of both controlled and uncontrolled k_∞ expressions are applied in Equation 4.3-2 to determine the moderator void coefficient.

[END HISTORICAL INFORMATION]

4.3.2.3 Stability

Sections 4.3.2.3 through Section 4.3.3 discuss original analyses of thermal hydraulic stability. Section 4.3.4.2 contains the most current methodology for stability requirements. Operational stability concerns are addressed in two separate areas. Stability analyses are performed to assure adequate protection against potentially damaging power oscillations during normal operation. These analyses generally demonstrate that the core exhibits a high degree of inherent power oscillation damping over most of the operating power-flow domain. These analyses, which are performed for each operating cycle, identify operating regimes in which oscillations are more likely.

Stability surveillance is performed to assure that potentially damaging power oscillations are detected and suppressed before fuel damage can occur. Operating regimes in which heightened surveillance is required are identified in the response to Reference 15.

The new stability concerns as a result of the LaSalle Unit 2 instability event of March 9, 1988 are addressed in Section 4.3.4.2.

4.3.2.3.1 Design Basis

The design basis for stability is contained in Section 4.3.1.2.

4.3.2.3.2 Description and Design Evaluation

The following discussions on stability are based on the original design basis of no inherent tendency toward oscillations (see Section 4.3.1.2) and are presented here for reference. The new design and operating criteria since IE Bulletin 88-07 are based on detection and suppression of oscillations as addressed in GDC 12 (see Section 4.3.1.2). Satisfaction of GDC 12 is achieved by implementing the “Option III” long term solution discussed in NEDO 31960 (see Section 4.3.4.2).

4.3.2.3.2.1 Introduction

A BWR unit consists of many interacting processes and associated control systems. A BWR process is self-regulating if it exhibits a negative reactivity feedback effect. In a BWR, when a control rod is withdrawn, core power increases due to a positive reactivity addition. This causes increased boiling, which increases the steam volume in the core, resulting in decreased neutron moderation. This decreases reactivity to counteract the added reactivity that results from the withdrawn control rod. Thus a rise in core power is limited by the negative feedback effect of the increased steam volume and serves as a self-regulating mechanism. A secondary inherent negative feedback effect, the Doppler reactivity effect, also occurs as the fuel temperature increases with power level. [4.3-29]

For the present discussion on feedback processes and control systems, the following definition of stability is used: a system is stable if, following a disturbance, the transient settles to a steady, noncyclic state. A system may be acceptably safe, even if oscillatory, provided the limit cycle of the oscillations is less than a prescribed magnitude.

An unstable process can be stabilized by a control system or by operator intervention. In general, however, it is preferable that a process with inherent feedback be designed to be stable by itself before it is combined with other processes and control systems. The design of the BWR is based on this premise, that individual system components are stable.

In the design of BWR systems, the following four types of stability are considered:

1. Channel hydrodynamic stability;

2. Reactor core (reactivity) stability;

3. Total system stability; and

4. Xenon spatial stability.

Items 1 and 2 are concerned with core dynamics and core design. These two types of stability are examined utilizing a linearized analytical model. First, the hydrodynamic channel stability of one type of channel operating in parallel with other channels in the core is considered, as flow oscillations may impede heat transfer to the moderator and/or drive the reactor into power oscillations. Second, the reactivity-feedback stability of the entire reactor core, which helps prevent power oscillations, is studied. Criteria have been established to ensure hydrodynamic stability of the channels and reactivity-feedback stability of the core.

Item 3 is concerned with total system dynamics, as the dynamics of the control systems, combined with those of the basic process, determine the dynamics of the entire reactor system. A time domain analysis, compatible with the frequency domain model, is applied to evaluate the total system stability. A stable system is analytically demonstrated if no inherent limit-cycle or divergent oscillation develops within the system as a result of step disturbances of any critical variable, such as steam flow, pressure, neutron flux, or recirculation flow.

Item 4 considers the operational problem of spatial xenon stability. Perturbations of reactor power level result in power distribution changes in the core due to the effects of xenon. The inherent nuclear characteristics of the core lead to strong damping of such disturbances, provided by the operating power coefficient of the core.

The criteria for the above four types of stability are stated in terms of two compatible parameters. First, the decay ratio x_{2}/x_{0} is defined as the ratio of the magnitudes of the second overshoot to the first overshoot resulting from a step perturbation. This may be seen as a graphic representation of the physical responsiveness of the system, which is readily evaluated in a time domain analysis. Second, in the frequency domain interpretation, the damping coefficient delta-$_{n}$ is defined as corresponding to the pair of poles closest to the jomega-axis in the *s*-plane for the system closed-loop transfer function, where jomega is the imaginary part of the complex frequency parameter *s*. The decay ratio and the damping coefficient have a direct relationship, as illustrated in Figure 4.3-5.

4.3.2.3.2.2 Reactor Core and Channel Hydrodynamic Stability Model Description

The Westinghouse stability methodology, which is described in References 28 through 33, is based on the RAMONA-3 computer code. The code is used to predict hydrodynamic channel instability and coupled neutronic and thermal-hydraulic stability for a full reactor core.

RAMONA-3 is a three-dimensional, transient, coupled neutronic and thermal-hydraulic code that explicitly models each fuel type in the reactor core. The code is comprised of a neutron kinetics model, a thermal hydraulic model, a steam line model, and several special models to represent the recirculation pumps, the jet pumps, the steam separator, the feedwater sparger, and various plant control and protection systems.

RAMONA has been validated by comparison to plant data from stability testing and from instability events. The application of the methodologies for implementing BWROG long-term solution Option III (detect and suppress), and the backup stability protection in the event the Option III OPRM hardware is unavailable, are described in References 30, 31, 32 and 33.

For AREVA licensed reload cores (starting with Quad Cities Unit 1 Cycle 25), AREVA uses the NRC approved STAIF methodology described in Reference 37 to predict core and channel instability for reload cores.

Cycle-specific setpoints for the Option III OPRM hardware system are provided consistent with the requirements of the BWROG topical report NEDO-32465-A (Reference 31) using the NRC approved RAMONA5-FA methodology (Reference 38). The two recirculation pump trip response for the OPRM setpoint calculation is determined using the AREVA BWR core simulator code, MICROBURN-B2 (Reference 27).

4.3.2.3.2.3 Total System Analytical Stability Model Description

The system transient and stability analyses reported in this section were performed using GE methods to demonstrate adequate system performance during potentially destabilizing control system actions. Although core and fuel response may not agree explicitly with the results of these analyses, system response remains unchanged and the results of these analyses are still valid. [4.3-32]

Refer to Section 15.0 for system transient analyses supporting the establishment of operating limits.

The total system model considers the entire reactor system (including the neutronics, heat transfer, and hydraulics aspects) as well as associated control systems (such as the flow controller, pressure regulator, feedwater controller, etc.). Although the control systems may be stable when analyzed individually, final control system settings must be made in conjunction with the operating reactor so that the entire system is stable. The model yields results which are essentially equivalent to those achieved with the core model, and allows the addition of the controllers which have adjustable features for attaining desired system performance within the inherent capabilities of the channel and core behavior. [4.3-33]

The model incorporates the dynamic equations which represent the BWR system in the time domain. The variables such as steam flow, pressure, etc., are represented as a function of time. The model is shown in block diagram form in Figure 4.3-8. Many of the blocks are extensive systems in themselves. The recirculation flow model is shown in greater detail in Figure 4.3-9. The model is constantly being improved as pertinent new experimental or reactor operating data are obtained.

4.3.2.3.3 Ultimate Performance Limit Criteria and Conformance

4.3.2.3.3.1 Criteria Definition

The Ultimate Performance Criteria are based on avoiding any inherent instability of total or component systems, whether manifest as a divergent oscillation or a limit-cycle oscillation. The pertinent systems are analytically evaluated for compliance with these criteria, described as follows. [4.3-34]

The assurance that the total unit is stable, and therefore has sufficient safety margin, shall be demonstrated analytically when the decay ratio x_2/x_0 is less than 1.0, or equivalently, when the damping coefficient delta-$_n$ is greater than zero, for each type of stability discussed. These limits are summarized in Table 4.3-1.

Special attention should be given to differentiate between inherent system limit cycles and small, acceptable, limit cycles which are caused by physical nonlinearities (dead-band, friction, etc.) in real control systems. The latter are always present, even in the most stable reactors, and are not representative of inherent hydrodynamic or reactivity instabilities in the reactor.

These criteria shall be satisfied for the unit for all attainable operating conditions. For stability purposes, the most severe condition used for applying these criteria was assumed to correspond to natural circulation flow at a power representing the rod block power limit condition.

In the following subsections 4.3.2.3.3.2 and 4.3.2.3.3.3 that address conformance to the Ultimate Performance Criteria, the values presented are typical values and do not necessarily represent current data.

4.3.2.3.3.2 Channel Hydrodynamic Conformance to the Ultimate Performance Criteria

The following results in Section 4.3.2.3.3.2 are based on the Quad Cities initial core.

The channel hydrodynamic performance calculation yielded the following results: [4.3-35]

Channel Hydrodynamic Performance
Natural Circulation at Maximum Power

Decay Ratio, x_2/x_0:	<0.01
Resonant Frequency, Hz:	0.318

In this most responsive attainable mode, rod block power during natural circulation flow, the most responsive channel was in conformance with the Ultimate Performance Criteria of <1.0 decay ratio. Therefore the channel performance over the entire range of attainable operation was well within stability limits.

4.3.2.3.3.3 Reactor Core Conformance to the Ultimate Performance Criteria

The following results in Section 4.3.2.3.3.3 are based on the Quad Cities initial core.

The reactor core performance calculation for the limiting condition corresponding to natural circulation flow and rod block power at that flow condition yielded the following results: [4.3-36]

(Initial Core) Reactor Core Performance
Natural Circulation at Maximum Power

Decay Ratio, x_2/x_0:	0.59
Resonant Frequency, Hz:	0.37

Figure 4.3-10 illustrates the variation of the decay ratio with flow and power. The calculated values showed the reactor to be in compliance with the Ultimate Performance Criteria in this most responsive attainable mode.

Cycle-specific Reactor Core Decay Ratio calculations are located in the cycle-specific licensing documents.

4.3.2.3.3.4 Total System Conformance to the Ultimate Performance Criteria

The results in Section 4.3.2.3.3.4 are based on the Quad Cities initial core.

The time response calculated for a step disturbance of each of three parameters described in the following section resulted in no instability, including limit-cycle operation, thus confirming that the decay ratio is less than 1.0, in conformance with the Ultimate Performance Limit Criteria. Figures 4.3-11 and 4.3-12 give the response of the system to a 10-psi pressure setpoint change; Figures 4.3-13 and 4.3-14 to a $0.10-rod notch reactivity change; and Figures 4.3-15 and 4.3-16 to a 6-inch water level setpoint change. The initial operating condition for each transient was at rod block power and natural circulation flow. The same transients initiated from full power operation were even more stable, as described in Section 4.3.2.3.4.4. [4.3-37]

4.3.2.3.4 Operational Design Guide and Conformance

4.3.2.3.4.1 Design Guide Limit

The following in Section 4.3.2.3.4.1 is based on the Quad Cities initial core.

Although the absolute stability of each Quad Cities unit is assured by the Ultimate Performance Limit Criteria described in Section 4.3.2.3.3, it is the practice to design to a level of operational excellence that will allow normal maneuvering and control with no underdamped response. Therefore, after meeting the Ultimate Performance Limit Criteria for stability over all attainable operating transients, the unit was analyzed to conform with the following Operational Design Guide Limits over all normally expected operating conditions. The Operational Design Guide Limits correspond to a decay ratio of <=0.25 for both reactor core performance and total system performance. An individual component or system may have a decay ratio in excess of 0.25, provided that the total system performance ratio is <=0.25. [4.3-38]

The Operational Design Guide analysis for transient performance is conducted for the total system, the reactor core, and the channel hydrodynamics, utilizing the same analytical methods as described in Sections 4.3.2.3.2.2 and 4.3.2.3.2.3.

The operational dynamic characteristics for each Quad Cities unit was demonstrated analytically to be within the Operational Design Guide Limits listed in Table 4.3-2 for all expected power and flow conditions to be encountered in normal operation. The expected most limiting condition corresponds to that attained starting from rated power and flow and reducing flow, potentially to natural circulation, with a corresponding power reduction. The power and flow condition at which the above limits are analytically attained are recognized procedurally as the operational boundary for normal manual or automatic control.

The total system time domain analysis evaluates the decay ratio response of the plant in terms of the primary response variables associated with the perturbed parameter. Thus, for a control rod change the primary response variables are the neutron flux and vessel steam flow; for the pressure setpoint change they are the vessel pressure and vessel steam flow; for the reactor water level setpoint change they are the reactor water level and vessel feedwater flow; and for a load demand perturbation they are core inlet flow and vessel steam flow.

The channel hydrodynamic Operational Design Guide Limit presented in Table 4.3-2 allows locally more responsive operation than is allowed for the complete core or the total system. This is justified by the fact that the response of an individual component can be less damped than the total system, as long as total performance is uncompromised and local transients are not harmful. Core stability and total system stability can both be satisfied in the presence of a highly responsive, but stable, channel. Because of the short period of natural resonance relative to the slow response of heat transfer, the local channel transients will not be manifest as significant local heat flux transients.

4.3.2.3.4.2 Channel Hydrodynamic Conformance to the Operational Design Guide

The following in Section 4.3.2.3.4.2 is based on the Quad Cities initial core.

The channel hydrodynamic performance calculations yielded the following results for rated operating conditions and for natural circulation conditions at the corresponding nominal power. (The numerical values cited below are typical values, but do not necessarily reflect current data. New conformance data are generated for each unit on a cycle basis.) [4.3-39]

Channel Hydrodynamic Performance

	Rated Conditions	Natural Circulation
Decay Ratio, x_2/x_0:	<0.01	0.28
Frequency, Hz:	0.538	0.342

The most responsive channel is, therefore, in conformance with the Operational Design Guide of <=0.5 decay ratio for channel performance.

4.3.2.3.4.3 Reactor Core Conformance to the Operational Design Guide

The results in Section 4.3.2.3.4.3 are based on the Quad Cities initial core.

Figure 4.3-10 shows that the calculated decay ratio x_2/x_0 varies along the rated power-flow control line from 0.01 at rated power and flow to 0.59 at that power level corresponding to natural circulation. At the end of life (EOL), the flow control range from rated power and flow to that point on the flow control line corresponding to 65% power does not violate the Operational Design Guide Limits. Although the Operational Design Guide Limit is violated for some power levels, the Ultimate Performance Limit (x_2/x_0 <1.0) is always achieved. [4.3-40]

Since Technical Specifications do not allow continued operation on natural circulation, combinations of low flow and high power sufficient to produce high decay ratios are not permitted. Furthermore, Quad Cities has implemented the requirements of IE Bulletin 88-07, Supplement 1, which prohibits normal operation in the power/flow regions known to be susceptible to core instabilities. [4.3-41]

4.3.2.3.4.4 Total System Conformance to the Operational Design Guide

The following in Section 4.3.2.3.4.4 is based on the Quad Cities initial core.

For normal operating modes, the time response of each of the primary response variables of the reactor system to small step disturbances must show a decay ratio of <=0.25 in order to satisfy the Operational Design Guide Limit. Each of the following disturbances were analytically imposed, one at a time, using the model previously described for time domain analysis: [4.3-42]

A. A pressure setpoint change of 10 psi;

B. A control rod position change equivalent to a $0.10 reactivity change;

C. A recirculation flow change equivalent to a power change of 10% of starting point; and

D. A reactor water level setpoint change of 6 inches.

The calculated responses of the primary variables to the pressure setpoint change of 10 psi and to the load demand perturbation are shown in Figures 4.3-17 through 4.3-20 for rated power operating conditions. Figures 4.3-21 and 4.3-22, 4.3-23 and 4.3-24, and 4.3-25 and 4.3-26 give the response to a pressure setpoint change, a rod notch change, and a level setpoint change, respectively, at the nominal power corresponding to the rated power-flow control path at natural circulation flow. In all cases, the decay ratio of each of the primary response variables is less than 0.25, thus indicating proper dynamic damping for expected normal operating conditions, in conformance with the Operational Design Guide. In this case, the total system response does not constrain the operating range. The reactor core performance analysis provides the only limiting constraint on normal operating range, as indicated in Section 4.3.2.3.4.3.

4.3.2.3.4.5 Xenon Stability Operational Design Guide and Conformance

In addition to the above guides, attention is also given to xenon induced disturbances, especially the effects of these disturbances on the flux distribution. [4.3-43]

AREVA has generically addressed xenon oscillations in Reference 38.

Operating experience at over 30 BWRs has shown that large BWRs are inherently stable against xenon-induced power oscillations. Large load changes have resulted in power distribution changes, in the vertical direction only, that are highly damped. The axial power distribution attains a steady-state condition 15 to 24 hours after the initiating disturbance.

4.3.2.3.5 Summary

The stability of the unit (including the basic process, associated equipment, and control systems) was evaluated by an extensive plant analytical simulation model. Selected near-step perturbations were introduced into the unit during startup testing to demonstrate the acceptable time response behavior of the reactor system at various conditions of operation. Compliance with the Ultimate Performance Limit Criteria was demonstrated at the attainable operational extremes and was evidenced by the absence of divergent oscillations or limit-cycle oscillations, excluding those minor fluctuations induced by the controller dead-band characteristics. [4.3-44]

Compliance with the Operational Design Guide at selected normal operating conditions was assessed, although approximately insofar as the near step perturbation obtainable approximates the analytical step disturbance. This assures desirable control and acceptable operating characteristics in all modes to be encountered. A detailed description of the GE BWR system stability analysis may be found in topical reports APED-5652[6] and APED-5640.[7]

Quad Cities Station has implemented the requirements of IE Bulletin 88-07, Supplement 1 and Reference 21 to ensure that adequate measures are taken to prevent the occurrence of uncontrolled power oscillations during all modes of BWR operation. See Section 4.3.4.2. [4.3-45]

4.3.3 Analytical Methods

Westinghouse nuclear evaluations are performed using an NRC approved lattice physics code and core simulator. See Reference 34 for a detailed description of the WEC nuclear design codes. [4.3-46]

For AREVA licensed reload cores (staring with Quad Cities Unit 1 Cycle 25), nuclear evaluations are performed using an NRC approved lattice physics code and core simulator. Reference 27 provides a detailed description of these AREVA nuclear design codes.

Since the BWR core contains hundreds of fuel assemblies of various designs and in various control states, moderator void conditions, and accumulated exposures, the nuclear evaluations are best addressed as two parts: lattice analysis and core analysis.

Most of the lattice analyses are performed during the fuel assembly design process. The fuel assembly modeling is further divided into two stages: the fuel rod cell and external region modeling using transport theory methods, and the coarse-mesh fuel assembly modeling based on cell homogenization and diffusion theory methods. The results of these single assembly calculations are reduced to "libraries" of lattice reactivities, relative rod powers, and few-group cross-sections as functions of instantaneous moderator void fraction, exposure, exposure-void history, control state, and fuel and moderator temperatures for use in the core analysis. These analyses are dependent upon fuel lattice parameters only, and therefore, are valid for all plants and cycles to which they are applied.

The lattice analyses of Quad Cities Units 1 and 2, which contain several types of fuel designs (see Section 4.2.2.1), are performed using PHOENIX for WEC (Reference 34) and CASMO-4 (Reference 27) for AREVA methods.[26]

The core analysis is unique for each cycle. It is performed prior to the cycle loading to demonstrate that the core meets all applicable safety limits. The principal tool used in the core analysis is a three-dimensional BWR simulator code. This code performs coupled nuclear and thermal-hydraulic calculations based on a coarse-mesh nodal approximation to the one-group, steady-state neutron diffusion equation. Neutron parameters are obtained from the lattice analysis output, parametrically fitted as a function of moderator density, exposure, control, and moderator density history for each fuel type. The BWR simulator code computes power distributions, exposure, and reactor thermal-hydraulic characteristics as a function of spatially varying moderator voids, control rod positions, fuel loading patterns, burnable poisons, coolant flow, and other design and operational variables. The BWR simulator code includes the Doppler reactivity effect as a function of effective average fuel temperature, and the effect of xenon poisoning.

For Westinghouse reloads, the core analysis is performed using an NRC approved lattice physics code (PHOENIX) and core simulator (POLCA). See Reference 34 for a detailed description of the WEC core design codes.

For AREVA licensed reload cores (starting with Quad Cities Unit 1 Cycle 25), lattice physics calculations are performed with CASMO-4, which provides local pin cell evaluations and a two-dimensional flux solution for the fuel lattice. When the lattice includes gadolinia burnable absorber material, self-shielded gadolinium cross sections are calculated internally by CASMO-4. Lattice output from CASMO-4 is translated into global core performance predictions by the MICROBURN-B2 three-dimensional core simulator[27].

4.3.4 Changes

4.3.4.1 New Fuel Type

The following includes current and new fuel types.

The SVEA-96 Optima2 fuel bundle consists of 96 rods, arranged in four 5x(5-1) sub-bundles. The sub-bundles are separated by a cruciform internal structure (water cross) in the channel. Each sub-bundle is assembled as a separate unit with its own top and bottom tie plates. The sub-bundles are inserted into the channel from the top and are supported at the bottom by a stainless steel inlet piece bolted to the channel. The inlet piece consists of a transition piece and bottom support with an integrated debris filter. The water cross has a square central channel and smaller water channels in each of the four wings to accommodate non-boiling water during operation. The design includes eight 2/3-length and four 1/3-length partial length rods. The positions of the part-length rods have been chosen to maximize shutdown margin and to optimize the critical power performance. The fuel assembly is lifted by a handle connected to the top end of the channel, and is supported against adjacent assemblies in the core by a double leaf spring. Further description of the Westinghouse SVEA-96 Optima2 fuel is contained in Section 4.2.2.1. [4.3-50]

For AREVA licensed reload cores (starting with Quad Cities Unit 1), the AREVA ATRIUM 10XM reload fuel will be used. This ATRIUM 10XM fuel bundle geometry consists of a 10x10 fuel lattice with a square internal water channel that displaces a 3x3 array of rods. Twelve of the ninety-one fuel rods are part-length fuel rods (PLFRs). The number and length of the PLFRs have been chosen to minimize two-phase pressure drop; positions of the PLFRs have been chosen to maximize shutdown margin and to optimize the critical power performance. The fuel assembly is lifted by a handle connected to the top end of the water channel. Further description of the ATRIUM 10XM fuel design is provided in Reference 39.

4.3.4.2 Current Resolution of Stability Concerns

On March 9, 1988, LaSalle County Station Unit 2 underwent a dual recirculation pump trip event. After the pump trip, while on natural circulation, the unit experienced an excessive neutron flux oscillation. The event was described in NRC Information Notice No. 88-39, "LaSalle Unit 2 Loss of Recirculation Pumps With Power Oscillation Event" dated June 15, 1988. [4.3-51]

The NRC concluded that this event raised generic questions and issued IE Bulletin No. 88-07 which requested that holders of operating licenses for BWRs ensure that adequate operating procedures and instrumentation are available and adequate operator training is provided to prevent the occurrence of uncontrolled power oscillations during all modes of operation.

In IE Bulletin No 88-07, Supplement 1, the NRC provided additional information concerning power oscillations in BWRs and requested that addressees take action to ensure that the safety limit for the plant minimum critical power ratio (MCPR) is not violated.

In Generic Letter 94-02, the NRC requested that manual operator actions be implemented until permanent hardware was installed as a solution to the issue of detection and suppression of neutron flux oscillations. These manual actions (Actions 1.a and 1.b of GL 94-02) were committed by ComEd[20]. The Quad Cities procedures specify an immediate reactor scram or immediate exit for specific operating regions and plant conditions that bound the requirements of GL 94-02, Actions 1.a and 1.b. Permanent, fully automatic oscillation detection and suppression hardware has been installed and therefore these manual actions are considered an alternate to an armed OPRM system.

Commonwealth Edison Company performed the actions requested by IE Bulletin 88-07 and 88-07, Supplement 1 for Quad Cities.

In Generic Letter 94-02, the NRC requested a review of existing procedures and training programs and modification as appropriate to strengthen administrative provisions intended to avoid power oscillations or to detect and suppress them if they occur prior to implementation of long-term solutions. [4.3-52]

Commonwealth Edison Company performed the actions requested by GL 94-02 for Quad Cities.

Commonwealth Edison Company joined the BWR Owners' Group (BWROG) program which is developing generic long-term solutions to the stability issue. The BWROG program developed a design and evaluation methodology to analyze thermal-hydraulic stability and identified several viable approaches to the long-term resolution of the stability issue. Details of this methodology and examples of the Option III solution concept that EGC will implement are discussed in NEDO-31960-A(9),NEDO-31960 Supplement 1(21), and CENPD-400-P-A, Rev 6(22).

For AREVA licensed reload cores with ATRIUM 10XM fuel (starting with Quad Cities Unit 1 Cycle 25), BAW-10255PA Revision 2 (Reference 38) is also used to analyze thermal hydraulic stability.

The solution to the instability problem implemented at Quad Cities is the installation of the ABB designed Oscillation Power Range Monitor (OPRM), see also Section 7.6.1.5.5. [4.3-53]

The OPRM system is designed to initiate a reactor scram via RPS trip logic and provide alarm indication upon detection of core power oscillations prior to exceeding the Minimum Critical Power Ratio (MCPR) safety limit. The OPRM augments the original Reactor Protection System (RPS) functions by adding the suppress function of thermal hydraulic oscillations in the reactor core and does not remove or replace any existing RPS functions. The OPRM utilizes the Local Power Range Monitor (LPRM) signals to detect core instabilities using the Period Based algorithm. Also, the OPRM uses Amplitude and Growth Rate algorithms which are implemented for defense in depth but are not relied upon for detecting instabilities. If an unacceptable oscillation is detected by any of these algorithms, a trip signal will be generated by the OPRM.

4.3.5 References

1. Deleted.

2. Deleted.

3. "Generation of Void and Doppler Reactivity Feedback for Application to BWR Design," General Electric Company, December 1975, NEDO-20964.

4. J.A. Miller to H.E. Bliss, "GE PRC 86-07 Limiting Control Rod Sequence for CRDA," General Electric Company, GE letter G-EBO-7-190, May 6, 1987.

5. Neal, L.G. and Zivi, S.M. "The Stability of Boiling-Water Reactors and Loops," Nuclear Science and Engineering, Vol. 30, p. 25, 1967.

6. "Stability and Dynamic Performance of the General Electric Boiling Water Reactor," General Electric Company, April 1969, APED-5652.

7. "Xenon Considerations in Design of Large Boiling Water Reactors," General Electric Company, June 1968, APED-5640.

8. Deleted.

9. "BWR Owner's Group Long-Term Stability Solutions Licensing Methodology," General Electric Company, November 1995, NEDO-31960-A.

10. Deleted.

11. Deleted.

12. Deleted.

13. Deleted.

14. Deleted.

15. NRC Generic Letter 94-02, "Long-Term Solutions and Upgrade of Interim Operating Recommendations for Thermal-Hydraulic Instabilities in Boiling Water Reactors", July 1994.

16 Deleted.

17. Deleted.

18. Deleted.

19. Deleted.

20. John C. Brons to William T. Russell, "Dresden Station Units 2 and 3, Quad Cities Station Units 1 and 2, LaSalle County Station Units 1 and 2, Response to Generic Letter 94-02 (BWR Stability), NRC Dockets 50-237 and 50-249, NRC Dockets 50-254 and 50-265, NRC Dockets 50-373 and 50-374," Commonwealth Edison Company, September 9, 1994.

21. "BWR Owner's Group Long-Term Stability Solutions Licensing Methodology", November 1995, NEDO-31960-A Supplement 1.

22. Generic Topical Report for the ABB Option III Oscillation Power Range Monitor (OPRM), May 1995, CENPD-400-P-A, rev. 6.

23. Deleted.

24. Deleted.

25. Deleted.

26. Letter to U. S. Nuclear Regulatory Commission, "Notification of Intent to Perform Analyses Using Vendor Safety Analysis Codes," Letter RS-03-174, September 19, 2003.

27. "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2," EMF-2158(P), Revision 0, October 1999.

28. "Reference Safety Report for Boiling Water Fuel," Westinghouse Topical Report CENPD-300-P-A, July 1996.

29. "Thermal-Hydraulic Stability Methods for Boiling Water Reactors," Westinghouse Topical Report CENPD-294-P-A, July 1996.

30. "Thermal-Hydraulic Stability Methodology for Boiling Water Reactors," Westinghouse Topical Report CENPD-295-P-A, July 1996.

31. "Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications," NEDO-32465-A, August 1996.

32. "Plant-Specific Regional Mode DIVOM Procedure Guideline," GENE Letter OG04-0153-260, June 15, 2004.

33. "Backup Stability Protection (BSP) for Inoperable Option III Solution," OG 02-0119-260, July 17, 2002.

34. "The Advanced PHOENIX and POLCA Codes for Nuclear Design of Boiling Water Reactors," Westinghouse Topical Report CENPD-390-P-A, December 2000.

35. Westinghouse Topical Report CENPD-284-P-A, "Control Rod Drop Accident Analysis Methodology for Boiling Water Reactors: Summary and Qualification," July 1996.

36. ANP-3287P Revision 1, Quad Cities Units 1 and 2 Thermal-Hydraulic Design Report for ATRIUM™ 10XM Fuel Assemblies, AREVA, November 2014.

37. EMF-CC-074(P)(A) Volume 4 Revision 0, "BWR Stability Analysis: Assessment of STAIF with Input from MICROBURN-B2", Siemens Power Corporation, August 2000.

38. BAW-10255PA Revision 2, "Cycle-Specific DIVOM Methodology Using the RAMONA5-FA Code", AREVA NP, May 2008.

39. ANP-3305P, Revision 2, "Mechanical Design Report for Quad Cities and Dresden ATRIUM 10XM Fuel Assemblies," AREVA, August 2016.

Table 4.3-1

ACCEPTABLE ULTIMATE PERFORMANCE LIMITS

Type of Dynamic Performance	Ultimate Performance Limit Criteria
Channel hydrodynamic stability	$x_2/x_0 < 1$ delta-$_n > 0$
Reactor core (reactivity) stability	$x_2/x_0 < 1$ delta-$_n > 0$
Total system stablility	$x_2/x_0 < 1$ delta-$_n > 0$

Where x_2/x_0 is the decay ratio and delta-$_n$ is the damping coefficient.

Table 4.3-2

ACCEPTABLE OPERATIONAL DESIGN LIMITS*

Type of Dynamic Performance	Operational Design Guide Limit
Channel hydrodynamic performance	$x_2/x_0 \leq 0.5$ $\delta_n \geq 0.11$
Reactor core (reactivity) performance	$x_2/x_0 \leq 0.25$ $\delta_n \geq 0.22$
Total system performance	$x_2/x_0 \leq 0.25$ $\delta*_n \geq 0.22$

Where x_2/x_0 is the decay ratio and $\delta*_n$ is the damping coefficient.

* Historical information based on initial core GE modeling. Superseded by NRC Generic Letter 94-02[16].

4.4 THERMAL AND HYDRAULIC DESIGN

4.4.1 Design Bases

The design basis for the thermal and hydraulic characteristics of the core is to ensure, in conjunction with the fuel system design, the plant equipment characteristics, the nuclear instrumentation, and the reactor protection system, that no fuel damage will occur during normal operation or operational transients caused by any reasonably expected single operator error or single equipment malfunction. Fuel damage is defined in Section 4.2.1.1. [4.4-1]

The above design basis is used both for the core design and for the determination of operating limits.

4.4.1.1 Fuel Damage Limits

There are two principal mechanisms which could cause fuel damage in reactor transients, each with a corresponding design limit to ensure that fuel damage would not occur. The fuel damage limit to prevent cladding overheating due to inadequate cooling is conservatively defined as the onset of transition boiling, whereas the fuel damage limit to prevent cladding overstraining due to UO_2 pellet expansion is defined as 1% strain of the Zircaloy cladding. These are discussed in detail in Section 4.2.1.1. [4.4-2]

These fuel damage limits are also employed in the development of operating limits to control reactor operation. Additional evaluations of the fuel rod thermal and mechanical performance during normal operation and anticipated operational occurrences (AOOs) are performed. An AOO is an incident of moderate frequency, i.e., greater than once per 20 years for a particular reactor.

Design analyses protect fuel integrity by assuring compliance with limitations against cladding overheat, cladding overstress, and fuel pellet overheat. Cladding overheat is avoided by operating within steady-state MCPR limitations as determined from the analysis of limiting system transients as described in UFSAR Chapter 15. Cladding overstress and fuel pellet overheat are avoided by operating within transient LHGR limitations as determined from the fuel rod mechanical design analysis as described in UFSAR Section 4.2.

4.4.1.2 Design Criteria, Operating Basis and Operating Limits

4.4.1.2.1 Design Criteria

The design criteria developed to implement the preceding design bases are discussed in the following paragraphs. The conditions addressed here correspond to reactor pressures at or above 685 psig and core flows at or above 10% of rated. The cases of reactor pressures below 685 psig or core flows below 10% of rated are addressed in Section 4.4.4.2.2. [4.4-3]

4.4.1.2.1.1 Minimum Critical Power Ratio

The onset of transition boiling results in a decrease in heat transfer from the cladding, and hence an elevated cladding temperature, and the possibility of fuel damage. However, the attainment of critical power, or transition boiling, is not a directly observable event in an operating reactor. Margin to transition boiling is calculated from plant operating parameters such as core power, core flow, feedwater temperature, and core power distribution. The margin for each fuel assembly is characterized by the critical power ratio (CPR), which is the ratio of the fuel assembly power which would produce onset of transition boiling divided by the actual fuel assembly power. The minimum (most limiting) value of this ratio among all fuel assemblies in the core is the minimum critical power ratio (MCPR). [4.4-3a]

4.4.1.2.1.2 Fuel Cladding Integrity Safety Limit Minimum Critical Power Ratio

The fuel cladding is one of the physical barriers which separate radioactive materials from the environs. The integrity of this barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use-related cracking may occur during the life of the cladding, fission product migration from such cracking is gradual and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the protection system safety settings. While fission product migration from thermally caused cladding perforations is just as measurable as that from use-related cracking, the occurrence of such cladding perforations signals a threshold beyond which still greater thermal stresses may cause gross, rather than gradual, cladding deterioration. Therefore, to prevent the possibility of sudden fuel damage, the Fuel Cladding Safety Limit MCPR is defined with margin to the conditions which would produce onset of transition boiling (MCPR of 1.0). These conditions represent a significant departure from the condition intended by design for planned operation. Therefore, the Fuel Cladding Integrity Safety Limit MCPR is established such that no fuel damage shall be calculated to result from an AOO.

Because fuel damage by overheating of cladding, defined in Section 4.2.1.1 as onset of transition boiling, is not directly observable, a conservative step-back approach is used to establish this safety limit such that the calculated MCPR for any AOO is no less than the Fuel Cladding Integrity Safety Limit MCPR.

The Fuel Cladding Integrity Safety Limit MCPR has sufficient conservatism to assure that in the event of an AOO initiated from the normal operating condition, more than 99.9% of the fuel rods in the core are expected to avoid boiling transition. The margin between MCPR of 1.0 (onset of transition boiling) and the Fuel Cladding Integrity Safety Limit MCPR is derived from a detailed statistical analysis considering all of the uncertainties in monitoring the core operating state, including uncertainty in the boiling transition correlation. Because the boiling transition correlation is based on a large quantity of full-scale data, there is very high confidence that operation of a fuel assembly at the condition of MCPR equal to the Fuel Cladding Integrity Safety Limit MCPR would not produce boiling transition.

Even if boiling transition were to occur, cladding perforation would not be expected. Cladding temperatures would increase to approximately 1100°F, which is below the perforation temperature of the cladding material. This has been verified by tests in the General Electric Test Reactor (GETR), where similar fuel operated above the critical heat flux for a significant period of time (30 minutes) without cladding perforation.

If reactor pressure should ever exceed the range of applicability of the critical power correlation during normal power operation, it would be assumed that the Fuel Cladding Integrity Safety Limit MCPR was violated.

Westinghouse utilizes the D4 correlation (References 3, 4, and 5) with correction factors of mass flux, exit pressure, inlet enthalpy and boiling length to determine CPR. For Quad Cities Unit 1 reload core analyses, AREVA utilizes the advanced ACE critical power correlation (Reference 28) for ATRIUM 10XM fuel, with input parameters of mass flux, pressure, inlet enthalpy, axial and local power distributions. For the co-resident Optima2 fuel, the SPCB critical power correlation (Reference 29) is used. Reference 30 describes the approved AREVA methodology for applying AREVA critical power correlations to co-resident fuel.

4.4.1.2.1.3 Maximum Linear Heat Generation Rate

In addition to the boiling transition limit (MCPR limit), operation is constrained to a maximum linear heat generation rate (MLHGR) stated in Section 4.4.2.2. This constraint is established to provide adequate safety margin to 1% plastic strain for abnormal operating transients initiated from high power conditions. [4.4-5]

A safety margin equivalent to the 1% plastic strain for abnormal operating transients from high power conditions is provided for low power conditions in the form of the power and flow dependent LHGR adjustment factors.

4.4.1.2.2 Operating Basis

Based on the preceding design criteria, the operating basis for the thermal and hydraulic characteristics of the core design is to control the local power density to levels such that the fuel assembly power is maintained less than the critical power. [4.4-6]

The basis of the steady-state MCPR and MLHGR limits is to provide sufficient margin to accommodate uncertainties and to ensure that the fuel damage limits would not be exceeded during transients caused by any reasonably expected single operator error or single equipment malfunction.

4.4.1.2.3 Operating Limits

The following operating limits are used during normal steady-state operation: the MCPR is maintained greater than or equal to the MCPR limit specified in the Core Operating Limits Report (COLR) and the linear heat generation rate (LHGR) does not exceed the MLHGR for the fuel type (see Section 4.4.2.2). The APLHGR is maintained less than or equal to the MAPLHGR limits specified in the COLR. Note that the above statement does not specify the operating power nor does it specify peaking factors; these parameters are controlled by the operator subject to a number of constraints including the thermal limits given above. [4.4-7]

4.4.2 Description of the Thermal and Hydraulic Design of the Reactor Core

4.4.2.1 Fuel Cladding Integrity Safety Limit Minimum Critical Power Ratio

The value of the Fuel Cladding Integrity Safety Limit MCPR is generated by statistical analyses of the core near the limiting MCPR condition, as described in Section 4.4.4.1.1. This safety limit applies not only to core-wide transients, but also can be conservatively applied to the localized rod withdrawal error transient and fuel loading error. [4.4-8]

Calculation of the operating limit MCPR (defined in Section 4.4.1.2.3) is addressed in Section 4.4.4.1.2.

4.4.2.2 Operating Limit Maximum Linear Heat Generation Rate

The MLHGR for normal steady-state operation is listed in the Core Operating Limits Report (COLR) for the various fuel types. [4.4-9]

Exposure-dependent limits are also specified for the maximum average planar linear heat generation rate (MAPLHGR) for every fuel assembly type. These MAPLHGR limits ensure that fuel operation would remain within the limits specified by 10 CFR 50.46 under accident conditions. The LHGR is monitored to ensure the thermal mechanical integrity of the fuel.

4.4.2.3 Core Coolant Flow Distribution and Orificing Pattern

The flow distribution to the fuel assemblies and bypass flow paths is calculated on the assumption that the pressure drop across all fuel assemblies and bypass flow paths is the same. This assumption has been confirmed by measuring the flow distribution in BWRs. The components of fuel assembly pressure drop considered are listed in Section 4.4.2.4. Pressure drop measurements made in operating reactors confirm that the total measured core pressure drop and calculated core pressure drop are in good agreement. There is reasonable assurance, therefore, that the calculated flow distribution throughout the core is in close agreement with the actual flow distribution of an operating reactor. [4.4-10]

An iteration is performed on flow through each flow path (fuel assemblies and bypass flow paths), which equates the total differential pressure (plenum to plenum) across each path and matches the sum of the flows through each path to the total core flow. The total core flow less the control rod cooling flow enters the lower plenum. A fraction of this passes through various bypass flow paths. The remainder passes through the orifice in the fuel support piece (experiencing a pressure drop) where some of the flow exits through the fit-up between the fuel support and the lower tieplate and through the lower tieplate holes into the bypass flow region. All initial and reload core fuel assemblies have lower tieplate holes. The majority of the flow continues through the lower tieplate (experiencing a pressure drop) where some flow exits through the flow path defined by the fuel channel and lower tieplate into the bypass region.

Heat balances on the active coolant (i.e., coolant in the channel except in the water rods) are performed nodally for each fuel assembly. Fluid properties are expressed as the fuel assembly average at the particular node of interest and are based on ASME Steam-Water Properties.

The relative radial and axial power distributions are used with the fuel assembly flow to determine the axial coolant property distribution, which gives sufficient information to calculate the pressure drop components within each fuel assembly type. When the equal pressure drop criterion described above is satisfied, the flow distributions are established.

When fuel designs with different hydraulic characteristics are used for reload cores, hydraulic compatibility is evaluated to assure that all of the fuel types in the core receive adequate cooling for their power requirements.

4.4.2.4 Core Pressure Drop and Hydraulic Loads

[Start of HISTORICAL INFORMATION]

The components of fuel assembly pressure drop considered are frictional, local, elevation, and acceleration pressure drops. The models used for calculating these pressure drop components are described in the following subsections. The technical discussions in the following subsections 4.4.2.4.1 through 4.4.2.4.4 are applicable to GE methods for calculation of pressure drop and hydraulic loads.

4.4.2.4.1 Frictional Pressure Drop

Frictional pressure drop is calculated with a basic model as follows:

$$\Delta P_f = \frac{w^2}{2 g \rho} \frac{fL}{D_H A_{ch}^2} \phi_{TPF}^2$$

where

ΔP_{f}	=	frictional pressure drop, psi
w	=	mass flow rate
g	=	acceleration of gravity
ρ	=	average nodal liquid density
D_{H}	=	channel hydraulic diameter
A_{ch}	=	channel flow area
L	=	incremental length
f	=	friction factor
ϕ_{TPF}	=	two-phase friction multiplier

The formulation for the two-phase multiplier is based on data taken from prototypical BWR fuel assemblies.

4.4.2.4.2 Local Pressure Drop

The local pressure drop is defined as the irreversible pressure loss associated with a change in flow area, which occurs in locations such as the orifice, lower tieplate, and spacers of a fuel assembly.

The general local pressure drop model is similar to the frictional pressure drop and is given by:

$$\Delta P_L = \frac{w^2}{2g\rho}\frac{K}{A^2}\Phi^2_{TPL}$$

where

ΔP_{L} = local pressure drop, psi

K = local pressure drop loss coefficient

A = reference area for local loss coefficient

ϕ_{TPL} = two-phase local multiplier

and w, g, and ρ are defined in Section 4.4.2.4.1. Empirical constants were added to fit the results to data taken for the specific designs of the BWR fuel assembly, as discussed in Section 4.4.2.5.1.

4.4.2.4.3 Elevation Pressure Drop

The elevation pressure drop is based on the relation:

$$\Delta P_E = \overline{\rho}\ \Delta L$$

$$\overline{\rho} = \rho_L(1-\alpha) + \rho_G\alpha$$

where

ΔP_{E} = elevation pressure drop, psi

ΔL = incremental length

$\overline{\rho}$ = average mixture density

α = nodal average void fraction

ρ_{L}, ρ_{G} = saturated liquid (water) and vapor density, respectively

The void fraction model uses an empirically fit constant to predict a large block of steam void fraction data.

4.4.2.4.4 Acceleration Pressure Drop

A reversible pressure change occurs when a flow area change is encountered, and an irreversible loss occurs when the fluid is accelerated through the boiling process. The basic formulation for the reversible pressure change resulting from a flow area change in the case of single-phase flow is given by:

$$\Delta P_{ACC} = (1-\sigma_A^2)\,\frac{w^2}{2 g_c\ \rho_L A_2^2}$$

$$\sigma_A = \frac{A_2}{A_1} = \frac{\text{final flow area}}{\text{initial flow area}}$$

where

ΔP_{ACC} = acceleration pressure drop

A_2 = final flow area

A_1 = initial flow area

ρ_L = liquid (water) density

In the case of two-phase flow, the liquid density is replaced by a density ratio so that the reversible pressure change is given by:

$$\Delta P_{ACC} = (1-\sigma_A^2)\,\frac{w^2 \rho_H}{2 g_c\, \rho_{KE}^2 A_2^2}$$

where

$$\frac{1}{\rho_H} = \frac{x}{\rho_G} + \frac{(1-x)}{\rho_L}, \text{homogeneous density}$$

$$\frac{1}{\rho_{KE}^2} = \frac{x^3}{\rho_G^2 \alpha^2} + \frac{(1-x)^3}{\rho_L^2 (1-\alpha)^2}, \text{kinetic energy density}$$

α = void fraction at A_2

x = steam quality at A_2

and other terms are as previously defined. The basic formulation for the acceleration pressure change due to density change is given in Section 4.2.4.4 of GESTAR II^[1] and is evaluated at the inlet and outlet of each axial node. The total acceleration pressure drop in BWRs is on the order of a few percent of the total pressure drop.

4.4.2.5 Correlation with Physical Data

General Electric Company has obtained substantial amounts of physical data in support of the pressure drop and thermal-hydraulic loads discussed in Section 4.4.2.4. Correlations have been developed to fit these data to the formulations discussed. [4.4-13]

4.4.2.5.1 Pressure Drop Correlations

General Electric Company has taken significant amounts of frictional pressure drop data in multi-rod geometries representative of BWR fuel assemblies and correlated both the friction factor and two-phase multipliers on a best fit basis using the pressure drop formulations presented in Sections 4.4.2.4.1 and 4.4.2.4.2. Tests are performed in single-phase water to calibrate the orifice and the lower tieplate, and in both single- and two-phase flow to arrive at best fit design values for spacer and upper tieplate pressure drop. The range of test variables is specified to include the range of interest to BWRs. New data are taken whenever there is a significant design change to ensure the most applicable methods are in use at all times.

Applicability of the single-phase and two-phase hydraulic models discussed in Sections 4.4.2.4.1 and 4.4.2.4.2 for fuel designs as described in NEDE-31152-P[2] was confirmed by full-scale prototype flow tests.

4.4.2.5.2 Moderator Void Fraction Correlation

The moderator void fraction correlation includes the effects of pressure, flow direction, mass velocity, quality, and subcooled boiling.

4.4.2.5.3 Heat Transfer Correlation

The Jens-Lottes heat transfer correlation (see GESTAR II[1]) is used in GE fuel design to determine the cladding-to-coolant heat transfer coefficients for nucleate boiling.

[End of HISTORICAL INFORMATION]

4.4.2.6 Thermal Effects of Anticipated Operational Occurrences

The evaluation of the core's capability to withstand the thermal effects resulting from AOOs is covered in Chapter 15 (Accident Analysis). [4.4-14]

4.4.2.7 Uncertainties in Estimates

Uncertainties in thermal-hydraulic parameters are considered in the statistical analysis which is performed to establish the fuel cladding integrity safety limit. The statistical analysis is discussed in Section 4.4.4.1.1. [4.4-15]

4.4.2.8 Flux Tilt Considerations

The inherent design characteristics of the BWR are well suited to limit gross power tilting. The stabilizing nature of the large moderator void coefficient effectively reduces the effect of perturbations on the power distribution. In addition, the incore instrumentation system, together with the online computer, provides the operator with information on the power distribution so that he can readily use control rods or other means to limit the undesirable effects of power tilting. Because of these design characteristics, it is not necessary to allocate a specific margin in the peaking factor to account for power tilt. If, for some reason, the power distribution could not be maintained within normal limits using control rods and flow, then the total core power would have to be reduced. [4.4-16]

4.4.3 Description of the Thermal and Hydraulic Design of the Reactor Coolant System

The Quad Cities reactor design employs variable recirculation flow control, which provides some degree of load following capability. The operating range is limited, however, by certain restrictions due to recirculation pump net positive suction head (NPSH), overall plant control characteristics, core thermal power limits, etc., as discussed in the following subsections. [4.4-17]

4.4.3.1 Quad Cities Units 1 and 2 Operating Map

The normal operating range for Quad Cities Units 1 and 2 is shown on a typical power-flow map in Figure 4.4-1. The equipment, nuclear instrumentation, and the reactor protection system, in conjunction with the operating procedures, maintain operations within the allowable area of this map for normal operating conditions. The boundaries, flow control lines (FCLs), and regions on this map are discussed in the following paragraphs.

4.4.3.1.1 Natural Circulation Line

The operating state for the reactor moves along this line in the absence of recirculation pump operation. [4.4-19]

4.4.3.1.2 Minimum Pump Speed Line

Startup of the unit is normally carried out beginning with the recirculation pumps operating at minimum speed, or approximately 32% speed. The operating state for the reactor follows the minimum pump speed (32%) line for normal control rod withdrawal with the recirculation pump held at minimum speed, up to approximately 40% core thermal power. [4.4-20]

4.4.3.1.3 100% Flow Control Line

This is the design FCL that passes through 100% power at 100% flow. The operating state for the reactor follows this line for rapid flow changes, at a fixed control rod pattern that corresponds to 100% power at 100% flow. The line is based on constant xenon concentration. [4.4-21]

4.4.3.1.4 Other Flow Control Lines

These lines represent the family of FCLs with control rod patterns corresponding to less than 100% power at 100% flow. During plant startup, a FCL is followed as the recirculation pump speed is increased above the minimum speed with a fixed control rod pattern. [4.4-22]

4.4.3.1.5 APRM Rod Block Line and APRM Scram Line

The APRM rod block line is established to limit the power increase, due to the inadvertent withdrawal of a selected control rod permitted by the rod worth minimizer (RWM), to values which avoid fuel damage. The APRM scram line represents power/flow conditions when a reactor scram is initiated. The APRM system is further described in Section 7.6. [4.4-23]

4.4.3.1.6 Recirculation Pump Constant Speed Line

This line shows the change in flow associated with power reduction from the 100% power, 100% flow condition while maintaining constant recirculation pump speed. [4.4-24]

4.4.3.1.7 Minimum Feedwater Flow Permissive Line

This line results from the recirculation pump NPSH requirements. Equipment automatically reduces the recirculation pumps to the minimum speed when the feedwater flow drops below a preset level. [4.4-25]

4.4.3.1.8 Maximum Extended Load Line Limit Analysis Region

The maximum extended load line limit analysis (MELLLA) is shown on power-flow map Figure 4.4-1. MELLLA provides a basis to support plant normal operation in the region of the power-flow map above the 100% FCL and bounded by the MELLLA Upper Boundary and a rated power of 2957 MWt. The MELLLA region provides operating flexibility to permit flow compensation for xenon buildup following startups and for fuel depletion later in the cycle while improving the efficiency of achieving and maintaining 100% power.

4.4.3.1.9 Increased Core Flow Operation Region

This is the area greater than 100% rated flow up to 100% rated core thermal power. This region allows for increasing core flow above the 100% rated value to increase reactor power, thereby increasing plant capacity. [4.4-27]

4.4.3.1.10 Region of Potential Thermal-Hydraulic Instability

The region of potential thermal-hydraulic instability is a region associated with high power and low flow conditions, the boundaries of which are determined each fuel cycle as part of the reload analysis. Continuous operation in this region is not permitted because of the possibility of neutron flux oscillations unless the OPRM system is fully functional. However, operation in this region should be avoided and immediate exit is desirable.[16] [4.4-28]

The OPRM is enabled at greater than 25% reactor power and less than 60% recirculation flow, which envelopes the region of potential instability. The OPRM function is to detect neutron flux oscillations, alert the operator and generate an RPS trip signal to scram the reactor if the oscillations are large enough to threaten the fuel integrity.

4.4.3.1.11 Section Deleted

4.4.3.2 Application of Thermal-Hydraulic Design to Plant Operation

The following is a simple description of BWR operation with recirculation flow control which summarizes the principal modes of normal unit operation. [4.4-29]

Assuming the unit is already critical and heatup is complete, full power operation is approached by withdrawing control rods, moving along the minimum recirculation pump speed line to approximately 40% core thermal power. At that point, power is increased to 100% by both withdrawing control rods and increasing recirculation pump speed. Care is taken to avoid the region of potential thermal-hydraulic instability depicted on the power-flow map. The stability analysis is performed on a cycle-specific basis. For the power/flow map with applicable stability regions, see the cycle-specific stability analysis as referenced in the supplemental reload licensing report.

For a normal unit startup, the recirculation pumps are at the minimum speed, and power is increased by control rod withdrawal until the feedwater flow has reached approximately 20%. An interlock prevents combinations of low power (low feedwater flow) and high recirculation flow which could create recirculation pump NPSH problems. Once this interlock is cleared, the operator is free to increase power by increasing pump speed as well as by further withdrawing control rods.

When recirculation pump speed is increased without rod movement, power and flow follow one of a family of FCLs. These lines are not followed perfectly during typical power ramps because of xenon buildup in the core, which causes actual power to fall to a lower FCL. Operation above the 100% FCL line is permitted in the MELLLA region. Operation in these regions allows operators to anticipate and compensate for the effects of xenon buildup during power increases. Care is taken to avoid the region of potential thermal-hydraulic instability.

Operation at greater than 100% rated flow can increase plant capacity. Analyses are performed to confirm the safety of operation at core flow rates up to 108% of rated flow condition.

For SVEA-96 Optima2 reloads, the basis for coastdown operation is provided in the cycle-specific Reload Licensing Report. The approval of Westinghouse BWR reload analysis methodology described in Reference 36 treats cycle extension strategies such as end-of-cycle coastdown operation as an operating flexibility option. For Quad Cities Unit 1 AREVA licensed reload cores, the basis for coastdown operation is provided in the cycle-specific reload licensing analysis report.

The limiting abnormal operational transients that are analyzed at rated flow as part of the reload licensing analysis are reanalyzed for increased core flow operation. The loss-of-coolant accident (LOCA) is evaluated at increased core flow operation as described in Section 6.3.3.2.2.4.3. The fuel loading error, rod drop accident (RDA), and rod withdrawal error are performed as part of the cycle-specific reload licensing analysis. Performance of the RDA evaluation is performed during initial startup conditions prior to core voiding conditions. For Westinghouse reload cores, the fuel loading error is evaluated for increased core flow operation and the rod withdrawal error event is performed at rated flow operation.

For AREVA reload cores, the mislocated assembly fuel loading error is evaluated at flow conditions representative of expected operation including the MELLA region and the Increased Core flow region. The rod withdrawal error analysis utilizes a flow condition consistent with the power level being analyzed.

The effects of the increased pressure difference on the reactor internal components, fuel channels, and fuel bundles as a result of the increased flow are also analyzed in order to ensure that the design limits are not exceeded.

4.4.4 Evaluation

The following describes the methods for evaluating the design limits and operating limits. Section 4.4.4.1 describes the evaluation of the Fuel Cladding Integrity Safety Limit MCPR and the Operating Limit MCPR. Section 4.4.4.2 describes the evaluations for other Technical Specification limits.

4.4.4.1 Fuel Cladding Integrity Safety Limit MCPR and Operating Limit MCPR Calculation Procedure

The statistical model used to evaluate the Fuel Cladding Integrity Safety Limit MCPR is described in Section 4.4.4.1.1. The Operating Limit MCPR calculations are addressed in Section 4.4.4.1.2.

4.4.4.1.1 Statistical Model

For reload cores licensed using Westinghouse methods (Unit 2), Section 5.3.2 of Reference 36 provides the Westinghouse methodology for establishing the MCPR safety limit. The methodology is general and acceptable for design and licensing for all BWR cores containing Westinghouse fuel as well as for mixed cores containing both Westinghouse fuel and non-Westinghouse fuel assemblies. The safety limit MCPR (SLMCPR) is established to provide that at least 99.9% of the fuel rods avoid boiling transition. The SLMCPR for Westinghouse fuel, which is evaluated on a cycle-specific basis, is determined by statistically convoluting the uncertainties associated with the calculation of CPR. These uncertainties include core power, system pressure, inlet temperature, core flow, manufacturing tolerances of the fuel assembly components, the calculation of the core flow distribution and the CPR correlation itself.

For Quad Cities Unit 1 AREVA licensed reload cores, the SLMCPR approved licensing methodology is described in Reference 27. The methodology is general and acceptable for design and licensing for all BWR cores containing AREVA fuel as well as for mixed cores containing both AREVA fuel and non-AREVA fuel assemblies. The SLMCPR methodology is used to determine the Technical Specification SLMCPR value that ensures that 99.9% of the fuel rods are expected to avoid boiling transition during normal reactor operation and anticipated operational occurrences. The SLMCPR is determined by statistically combining calculation uncertainties and plant measurement uncertainties that are associated with the calculation of CPR. AREVA calculates the SLMCPR on a cycle-specific basis to protect all allowed reactor operating conditions. The analysis is performed on a cycle-specific basis explicitly accounting for changes in fuel and core designs.

The results of the reload analysis show that at least 99.9% of the fuel rods in the core are expected to avoid the onset of transition boiling if the MCPR is equal to or greater than the cycle-specific safety limit MCPR. The value of the cycle-specific safety limit MCPR can be found in the cycle-specific Reload Licensing Report.

4.4.4.1.2 Operating Limit MCPR Calculation Procedure

A cycle-specific Operating Limit MCPR is established to provide adequate assurance that the Fuel Cladding Integrity Safety Limit MCPR for that plant is not exceeded for any AOO. This operating requirement is obtained by the addition of the maximum incremental CPR value (DELTA-CPR) for the most limiting AOO (including any imposed adjustment factors) to the Fuel Cladding Integrity Safety Limit MCPR. [4.4-35]

Some of the models used for analyzing transients are described in Sections 4.4.4.1.2.1 and 4.4.4.1.2.2. Calculation of Operating Limit MCPR is addressed in Section 4.4.4.1.2.3.

4.4.4.1.2.1 Core-Wide Rapid Pressurization Transients

For reload cores, the results of the fast transient analysis are provided in the cycle-specific Reload Licensing Report. For Westinghouse licensed reload cores, the basis for the methods employed in obtaining these results is described in Reference 36. [4.4-36] For Quad Cities Unit 1 AREVA licensed reload cores, the bases for the approved licensing analysis methods are described in References 31 and 34 and supported by the cycle-specific reload safety analysis report.

The significant nuclear input parameters for the pressurization transients are as follows.

A. Scram Reactivity

Scram reactivity is the reactivity worth of control rods as a function of time or position following the scram signal. The scram reactivity insertion is normally lowest at the end of cycle (all-rods-out condition), because there are no partially inserted rods to insert negative reactivity more quickly than the fully-withdrawn rods. Moreover, core power tends to be top-peaked due to plutonium, which also reduces the initial reactivity effect of rod insertion.

B. Moderator Void Reactivity Coefficient

The moderator void reactivity coefficient is an important parameter not only in transient analysis, but also in core stability (see Sections 4.3.2.2.1.3 and 4.3.2.3). The core average moderator void coefficient must be negative; however, it must not be so negative as to yield such a strong positive reactivity feedback during moderator void collapse events that core and vessel limits are threatened. Conversely, events with moderator void increase must produce sufficient negative feedback to maintain operation within safety limits. A transient index which is used to assess the moderator void reactivity characteristics is the dynamic moderator void coefficient. This parameter is defined as the core physics moderator void coefficient multiplied by the average full power moderator void fraction, divided by the delayed neutron fraction.

C. Doppler Coefficient

The presence of U-238 and Pu-240 results in a strong negative Doppler coefficient. This coefficient provides instantaneous negative reactivity feedback to any fuel temperature rise, either gross or local. The magnitude of the Doppler coefficient is not dependent on gadolinium position or concentration in any bundle, because gadolinium has very little effect on the resonance group flux or on the U-238 content of the core.

More information on reactivity coefficients is provided in Section 4.3.2.2.1.

4.4.4.1.2.2 Slow Core-Wide Transients

For reload cores, the results of the slow transient analysis are provided in the cycle-specific Reload Licensing Report. For Westinghouse licensed reload cores (Unit 2), the basis for the methods employed in obtaining these results is described in Reference 36. [4.4-37] For Quad Cities Unit 1 AREVA licensed reload cores, the bases for the approved licensing analysis methods are described in References 31 and 34 and supported by the cycle-specific reload safety analysis report.

4.4.4.1.2.3 Operating Limit MCPR

The Operating Limit MCPR (OLMCPR) is a function of core power and core flow. For any given power/flow condition, the OLMCPR is the higher of the power-dependent limit and the flow-dependent limit appropriate for the plant operating conditions. For AREVA licensed reload cores (starting with Quad Cities Unit 1), the bases for the approved licensing analysis methods are described in References 31 and 34 and supported by the cycle-specific reload safety analysis report.

A. Flow Dependent Limits:

The OLMCPR must be increased for low flow conditions because in the BWR, power increases as core flow increases which results in a corresponding lower MCPR. If the MCPR at a low flow condition were at the 100% power and flow Operating Limit MCPR, a sufficiently large inadvertent flow increase could cause the MCPR to decrease below the Fuel Cladding Integrity Safety Limit MCPR. Therefore, the required Operating Limit is increased at reduced core flow rates.

For Reload Cores Licensed Using Westinghouse Methods (Units 2):

The flow-dependent MCPR operating limit is defined by the rated MCPR limit and a flow-dependent MCPR limit function. The MCPR limit at any operating state is the higher of the rated MCPR limit and the appropriate value from the flow-dependent MCPR limit function. The flow-dependent MCPR limit function is defined using steady-state thermal-hydraulic methods.

Additional power-flow statepoints are selected for analysis along a constant xenon flow control line passing through this endpoint. The MCPR values resulting from the same relative power distribution assumed at each of these statepoints define a flow-dependent limit function which protects the MCPR Safety Limit during an arbitrary flow increase to the maximum attainable by the equipment. The analysis approximates a slow flow increase transient; however, no transient effects are modeled.

For AREVA licensed reload cores (starting with Quad Cities Unit 1):

The flow-dependent MCPR operating limit ($MCPR_f$) at a given core flow operating state is a function of the maximum core flow capability at rated power, the Safety Limit MCPR (SLMCPR), and the core flow operating state. The flow-dependent MCPR limit is determined using steady-state thermal-hydraulic methods. The analysis approximates a slow flow increase transient.

The slow flow excursion event assumes a failure of the recirculation flow control system such that the core flow slowly increases to the maximum flow physically attainable by the equipment. An uncontrolled increase in flow creates the potential for a significant increase in core power and heat flux. A conservatively steep flow runup path is used in the analyses. $MCPR_f$ limits are determined for all fuel types in the core. The $MCPR_f$ limit is set such that the increase in core power and flow conditions resulting from the maximum increase in core flow does not result in violation of the safety limit MCPR.

B. Power Dependent Limits:

For Reload Cores Licensed Using Westinghouse Methods (Unit 2):

Events such as loss of feedwater heating and turbine trip without bypass become less severe when initiated from power levels less than rated power. Reduced power corresponds to reduced total steam flow. For the loss of feedwater heating event, reduced steam flow results in a decrease in both the feedwater flow and the maximum temperature rise across a given heater. The core subcooling change associated with the loss of feedwater heating will be lower for less than rated steam flow, as will the associated positive reactivity insertion. Therefore, the event is less severe than that initiated from rated steam flow.

For a turbine trip without bypass event initiated from a reduced power state, the pressure rise and the resulting moderator void collapse caused by closure of the turbine stop valve would be reduced because of the reduced steam flow, and the event would be less severe than that initiated from rated steam flow. Results for other BWR plants confirm this observation.

At lower power levels, events such as inadvertent startup of an idle recirculation pump, recirculation flow controller failure (increasing flow), feedwater flow controller failure (to maximum demand), and rod withdrawal error can become more severe than transients which are limiting at design conditions. However, the decrease in MCPR for these events is more than compensated for by the increase in the reduced flow MCPR limit at lower power and flow.

Transient analyses are performed at the full power, EOC, all-rods-out condition. Once the unit reaches this condition, it may be shut down for refueling or it may be placed in coastdown operation which involves coasting to a lower power level. Cycle extension techniques, as identified in the Core Operating Limits Report or applicable cycle specific reload document, may be used during coastdown. [4.4-41]

Final Feedwater Temperature Reduction (FFWTR) at the end of cycle is used to extend full power operation. By reducing the feedwater temperature, more moderation of the neutrons will increase the power capability and extend the operating cycle. In practice, the application of FFWTR is coupled with coastdown operation. As a result, operating limits are established to support combined FFWTR/Coastdown operation with a decrease in feedwater temperature of up to 120^{0}F. Operating limits for combined FFWTR/Coastdown operation are applicable to FFWTR operation.

If rods are inserted during coastdown, the maximum obtainable core power is reduced and scram worth increases, which results in less severe transients relative to all rods out.

For reload cores licensed using Westinghouse methods, Section 7.3.3 of Reference 36 provides the Westinghouse methodology for establishing the operating limit MCPR (OLMCPR). The cycle-specific OLMCPR is established to ensure that the SLMCPR is not exceeded for any moderate frequency event (anticipated operational occurrence). This limit is established by adding the maximum ΔCPR value for the most limiting transient postulated to occur from rated conditions to the SLMCPR.

For SVEA-96 Optima2 reloads, the basis for coastdown operation is provided in the cycle-specific Reload Licensing Report. The approval of Westinghouse BWR reload analysis methodology described in Reference 36 treats cycle extension strategies such as end-of-cycle coastdown operation as an operating flexibility option.

<u>For AREVA licensed reload cores (starting with Quad Cities Unit 1):</u>

Postulated plant transients categorized as an anticipated operational occurrence (AOO) are considered in determining the power dependent MCPR limits. Typically limiting or near-limiting transients are the generator load rejection with no bypass (LRNB), turbine trip with no bypass (TTNB), feedwater controller failure (FWCF), loss of feedwater heating (LFWH), inadvertent high pressure coolant injection startup (IHPCIS), loss of stator cooling (LOSC), and control rod withdrawal error (CRWE).

These transients are analyzed at the licensed core power and appropriate lower power intervals as needed to determine the limiting AOO transient for setting fuel thermal limits as a function of core power. Below 38.5% core power (Pbypass), the direct control rod scram on turbine stop valve position or turbine control valve fast closure is bypassed, and the scram occurs on either high neutron flux or high pressure. The thermal limits Pbypass condition may be established for two core flow ranges (e.g., above and below 60% flow) to provide improved operating limits. Final Feedwater Temperature Reduction (FFWTR), coastdown operation, and Increased Core Flow to extend the fuel cycle are discussed in the reload safety analysis report.

The cycle-specific OLMCPR limits are established per the Reference 31 approved licensing methodology and documented in the cycle-specific reload safety analysis report. These limits ensure that the SLMCPR is not violated for any moderate frequency event (AOO) as a function of power, flow and cycle exposure. These limits are established by adding the maximum ΔCPR value for the most limiting AOO transient postulated to occur to the SLMCPR. In a similar manner, cycle-specific, power-dependent LHGR multiplier (LHGRFACp) limits are determined for each fuel type.

4.4.4.2 Additional Evaluation of Technical Specification Limits

The information in this section is intended to support the limits set forth in the Technical Specifications. An important element in this discussion is the demonstration of margins existing between levels of potential safety concern and those to which the unit is limited, either by nature or by protective device action. To do this, data for abnormal core conditions will be presented. These sets of data are for illustration only and do not represent expected operating situations. [4.4-44]

4.4.4.2.1 Core Power Safety Limit

The Technical Specifications contain a safety limit, related to core thermal power, to assure reasonable protection of the clad barrier. Operation at or below this limit is required to maintain the integrity of the fuel cladding. [4.4-45]

4.4.4.2.2 Core Thermal Power Limit (Reactor Pressure below 685 psig or Core Flow below 10% of Rated)

At low powers and flows a pressure drop is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low powers and flows will always be greater than 4.5 psi. Analyses show that with a fuel assembly flow of 28 x 10^3 lb/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than 28 x 10^3 lb/hr. Full scale critical power test data taken at pressures from 14.7 to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a thermal power of > 50% RTP. Thus, a core thermal power limit of 25% for reactor pressures below 685 psig is conservative. [4.4-46]

Limitations are enforced by the reactor protection system (RPS) discussed in Section 7.2.

4.4.4.2.3 Limiting Safety System Settings

The Technical Specifications also include limiting safety system settings. Those related to the core thermal safety limit include the reactor high pressure scram and the high neutron flux (APRM) scram. [4.4-47]

In the analyses for these limiting safety system settings, the assigned trip values are intended for use in normal operation. They are not set to the maximum levels that would be allowed by the assumed safety limit.

The APRM scram trip setting is selected to provide adequate margin to the Fuel Cladding Integrity Safety Limit MCPR and yet allow operating margin to reduce the possibility of unnecessary scrams. Analyses demonstrate that at the Technical Specifications APRM scram trip setting, none of the abnormal operational transients analyzed violate the safety limit, and there is substantial margin from fuel damage. The high pressure scram is available as a backup protection to the high flux scram. These limiting safety system settings are addressed in Section 7.2 and in Technical Specifications. [4.4-48]

The APRM system provides a control rod block to prevent rod withdrawal at any recirculation flow rate to provide gross protection against exceeding the Fuel Cladding Integrity Safety Limit MCPR. This rod block trip setting, which is automatically varied with recirculation loop flow rate, prevents an increase in the reactor power level to excessive values due to control rod withdrawal. The flow variable trip setting provides substantial margin from fuel damage, assuming a steady-state operation at the trip setting, over the entire recirculation flow range. The APRM rod block maximum setting is addressed in Section 7.6 and in the Technical Requirements Manual.

The turbine stop valve closure scram trip anticipates the pressure, neutron flux, and the heat flux increase that could result from rapid closure of the turbine stop valves. With the Technical Specification scram trip setting, the resultant increase in surface heat flux is limited such that MCPR remains above the Fuel Cladding Integrity Safety Limit MCPR even during the worst-case transient that assumes the turbine bypass is closed. This

Both the reactor protection system and the reactor's inherent physical characteristics prevent exceeding the safety limit conditions. Normal steady-state operation is limited by the Operating Limit MCPR, the MLHGR, and the MAPLHGR. Slow increases in power would be terminated by the action of pressure scram, APRM scram, or APRM rod block (in case of rod withdrawal). [4.4-49]

4.4.4.2.4 Power Transient

During an operating transient the heat flux would lag behind the neutron flux due to the inherent heat transfer time constant of the fuel, which varies by fuel type up to approximately 8 to 9 seconds. Also, the limiting safety system settings are at values which will not allow the reactor to be operated above the Safety Limit during normal operation or during analyzed abnormal operating situations. For normal operating transients, the neutron flux transient is terminated before a significant increase in surface heat flux occurs. Control rod scram times are checked as required by the Technical Specifications. [4.4-50]

As a result of the combination of rapid flux increase, negative reactivity feedback from fuel temperature and moderator voids, and a scram, the safety limit will not be exceeded for normal turbine or generator trips, which are the most severe normal operating transients expected. These analyses show that even if the bypass system fails to operate, the design limit of MCPR equal to the Fuel Cladding Integrity Safety Limit MCPR would not be exceeded.

During periods when the reactor is shut down, consideration must also be given to water level requirements due to the effect of decay heat. If the reactor water level should drop below the top of the active fuel during this time, the ability to cool the core is reduced, which could lead to high cladding temperatures and cladding perforation. The core would be cooled sufficiently to prevent clad melting should the water level be reduced to two-thirds the core height. Establishment of the safety limit above the top of the active irradiated fuel provides adequate margin (where the top of active fuel is 360 inches above vessel zero). This level is continuously monitored whenever the recirculation pumps are not operating.

In rapid power increases, the neutron flux may exceed the scram level by a considerable amount, but the resulting scram would be sufficiently fast that the heat flux would either decrease or exceed the initial value by only a small amount. [4.4-51]

Transients which would be limiting because of MCPR would primarily involve significant changes in power. The transients most likely to limit operation because of MCPR considerations are: [4.4-52]

A. Turbine trips or generator load rejection without bypass (Section 15.2),

B. Loss of feedwater heating (Section 15.1),

C. Feedwater controller failure (to maximum demand) (Section 15.1),

D. Control rod withdrawal error (Section 15.4),

E. Deleted

F. Inadvertent startup of high pressure coolant injection during power operation (Section 15.5.1).

G. Loss of Stator Cooling (Section 15.2)

The analyses discussed in the sections referenced above demonstrate that given a reasonable safety limit: [4.4-53]

A. The reactor conditions corresponding to the safety limit are extreme and would probably never be encountered in the life of the plant; and

B. With the protective devices provided, the reactor cannot approach conditions corresponding to the safety limit.

4.4.4.2.5 Single Loop Operation

For reactor operation with only one recirculation pump running (single loop operation), both the Safety Limit MCPR and the Operating Limit MCPR are increased, as specified in the Technical Specifications and the Core Operating Limits Report (COLR) respectively. This is to account for increased uncertainty in core flow and transversing incore probe (TIP) instrumentation readings for single loop operations. In addition, as with two-loop operation, the Operating Limit MCPR is flow-dependent and power-dependent as defined in the COLR. This prevents an inadvertent flow increase from causing the MCPR to drop below the Safety Limit MCPR.

For Quad Cities Unit 1, AREVA has concluded that for most events used to establish operating limits, the consequences in single loop operation are bound by those in two loop operation. While pump seizure is classified as an accident, it is treated as an AOO in establishing the single loop operating limits. As a result, the pump seizure event is a potentially limiting event in single loop operation. Therefore, the results of the single loop pump seizure event are considered when establishing operating limits for single loop operation. (Reference 33)

The flow-biased APRM scram and rod block setpoints are also reduced for single loop operation by amounts specified in the Technical Specifications, Core Operating Limits Report, or Technical Requirements Manual, as applicable.

For single loop operation, the APLHGR limit corresponds to the product of the two-loop limit and a reduction factor as specified in the COLR. This calculated APLHGR limit is conservatively used to ensure that the PCT for single loop operation is bounded by the PCT for two-loop operation

4.4.5 Testing and Verification

Testing and verification were performed initially and are performed following each reload to confirm that the thermal and hydraulic characteristics of the core and the reactor coolant system are in accordance with design values and will remain within required limits throughout core lifetime. The initial test program is described in Chapter 14, and test results are contained in the Startup Test Reports for individual cycles.

4.4.6 Instrumentation Requirements

The functional requirements for the instrumentation employed in monitoring and measuring those thermal-hydraulic parameters important to safety are addressed in Sections 7.2, 7.5, and 7.6.

Quad Cities Units 1 and 2 do not have a loose-parts monitoring system.

4.4.7 References

1. "General Electric Standard Application for Reactor Fuel, " (GESTAR II), June 2000, General Electric Company, NEDE-24011-P-A-14, and U.S. Supplement, June 2000, NEDE-24011-P-A-14-US.

2. "General Electric Fuel Bundle Designs," (Revision 5), General Electric Company, NEDE-31152-P, June 1996.

3. "10x10 SVEA Fuel Critical Power Experiments and CPR Correlation: SVEA-96 Optima2," Westinghouse Report WCAP-16081-P-A (proprietary), March 2005.

4. "SVEA-96 Optima2 CPR Correlation (D4): High and Low Flow Applications," Westinghouse Report WCAP-16081-P-A, Addendum 1-A (proprietary), March 2009.

5. "SVEA-96 Optima2 CPR Correlation (D4): Modified R-factors for Part-Length Rods," Westinghouse Report WCAP-16081-P-A, Addendum 2-A (proprietary), February 2009.

6. Deleted.

7. Deleted.

8. Deleted.

9. Deleted.

10. Deleted.

11. Deleted.

12. Deleted.

13. Deleted.

14. Deleted.

15. Deleted.

16. NRC Generic Letter 94-02, Long-Term Solutions and Upgrade of Interim Operating Recommendations for Thermal-Hydraulic Instabilities in Boiling Water Reactors, July 1994.

17. Deleted.

18. Deleted.

19. Deleted.

20. Deleted.

21. Deleted.

22. Deleted.

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24. Deleted.

25. Deleted.

26. Deleted.

27. ANP-10307PA Revision 0, “AREVA MCPR Safety Limit Methodology for Boiling Water Reactors,” AREVA NP, June 2011.

28. ANP-10298P-A Revision 1, “ACE/ATRIUM 10XM Critical Power Correlation,” AREVA, March 2014.

29. EMF-2209(P)(A) Revision 3, “SPCB Critical Power Correlation,” AREVA NP, September 2009.

30. EMF-2245(P)(A) Revision 0, “Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel,” Siemens Power Corporation, August 2000.

31. XN-NF-80-19(P)(A) Volume 3 Revision 2, Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description, Exxon Nuclear Company, January 1987.

32. BAW-10247PA Revision 0, Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors, AREVA NP, February 2008.

33. ANP-3565P Rev. 0, “Quad Cities Unit 1 Cycle 25 Reload Safety Analysis,” AREVA Inc., February 2017.

34. XN-NF-80-19(P)(A) Volume 4 Revision 1, Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads, Exxon Nuclear Company, June 1986.

35. Deleted.

36. CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel," July 1996.

Table 4.4-1

UNCERTAINTIES USED IN STATISTICAL ANALYSIS
(For P/BP8x8R and GE8x8EB Fuel)

Table has been intentionally deleted

Table 4.4-1A

UNCERTAINTIES USED IN STATISTICAL ANALYSIS
(SPC Reload Safety Limit MCPR)

Table has been intentionally deleted.

Table 4.4-2

NOMINAL VALUES OF STATISTICAL ANALYSIS PARAMETERS
(For P/BP8x8R and GE8xEB Fuel)

This table has been intentionally deleted

Table 4.4-2A

PLANT MEASUREMENT UNCERTAINTIES
(For ATRIUM-9B Fuel)

This Table has been intentionally deleted

4.5 REACTOR MATERIALS

4.5.1 Control Rod Drive System Materials

Control Rod Drive System materials are discussed in Section 4.6.

4.5.2 Reactor Internals Materials

The major internal components of the reactor include the fuel assemblies, control rods, control rod drives, incore monitors, shroud and other internal support structures, steam separators, steam dryers, jet pumps, and the feedwater, core spray, and standby liquid control spargers and nozzles. This section does not include the fuel assemblies, control rods or incore monitors; these components are discussed in Sections 4.2, 4.6, and 7.6, respectively. A description of the reactor internals, including steam separators and steam dryers, is included in Section 3.9.5.

A. Structural components: [4.5-1]

1. Shroud — 304 stainless steel,
   Shroud Tie Rod with Spring Stabilizers - 316/316L, XM-19 stainless steel; nickel base alloy X-750,

2. Baffle plate — Inconel,

3. Baffle supports — Inconel,

4. Core top grid — stainless steel,

5. Core bottom grid — stainless steel,

6. Fuel support piece — 304 stainless steel,

7. Control rod guide tubes — stainless steel,

8. Incore instrument tubes — 304 stainless steel. [4.5-2]

B. Jet Pumps — the jet pump assemblies are made entirely from corrosion-resistant Type-304 austenitic stainless steel except as noted: [4.5-3]

1. To accommodate the clamping loads generated in holding the inlet-throat subassembly in place, the beam is fabricated from Inconel X-750. As discussed in Section 5.4, jet pump parameters are monitored in order to detect integrity or operability problems which could indicate possible jet pump beam cracking. Visual inspection of the jet pump beam bolts is also performed during each refueling outage.

2. The jet pump assembly contains one slip joint which permits removal of the inlet-throat subassembly. Both contacting surfaces utilize Stellite-6 with a minimum Rockwell hardness of RC-30 to prevent wear in the mating parts.

3. The spring used in the restrainer wedge device is Inconel X-750 having the higher strength properties necessary for desirable spring forces.

4. Some stainless steel members which are threaded, or otherwise bear against other stainless steel parts, are surface hardened by nitriding to prevent galling. Nitriding, a process whereby nitrogen is diffused into the base material, provides a thin, wear resistant layer (up to 0.007 inch thick) with a minimum hardness of 90 on the Rockwell 15 N scale.

5. Replacement restrainer gates – 316 stainless steel with Inconel X-750 hardware.

6. Slip Joint Clamps – XM-19 Stainless steel with Inconel X-750 hardware.

C. Spargers and spray nozzles:

1. Core spray sparger/spray nozzles — 304 stainless steel. [4.5-4]

2. Feedwater spargers — the sparger pipe, reducing tee, adaptor, elbow and orifice are all 304 stainless steel. The end plates and the extension on the reducing tee are 316L and the sparger brackets are CF-3 cast stainless steel. The thermal sleeve is made of Inconel 600 and 316L stainless steel. The upstream portion which contacts the safe end is made of Inconel 600. [4.5-5]

3. Standby liquid control sparger — stainless steel. [4.5-6]

D. Missing Piece of Dryer Outer Hood Plate Evaluation:

The steam dryer outer bank hood was found to be missing a triangular ½ inch thick 304 stainless steel plate measuring approximately 6.5 inches by 9.0 inches during Q1F51. It could not be determined if the missing material was lost as one piece or more than one piece. Extensive searches were performed without finding any of the lost material. The consequences of the missing piece(s) were evaluated in EC 345951, Missing Piece of Dryer Outer Hood Plate Evaluation. The possible consequences of impact damage, fretting, flow obstructions and valve closure interference were evaluated and it was concluded that there was no undue risk operating Unit 1 with the parts remaining in the reactor coolant system.

4.6 FUNCTIONAL DESIGN OF REACTIVITY CONTROL SYSTEMS

4.6.1 Design Bases

The reactivity control system is designed so that during conditions of normal operation sufficient reactivity compensation is always available to make the reactor adequately subcritical from its most reactive condition, and means are provided for continuous regulation of the core excess reactivity (see Section 4.3). This system is also designed to compensate for positive and negative reactivity changes resulting from changes in nuclear coefficients, fuel depletion, and fission product transients and fission product buildup.

The reactivity control system will respond to signals from the reactor protection system (RPS) to prevent fuel damage from gross or local reactor power disturbances caused by operator error or equipment malfunctions.

The system is designed to limit control rod worths and the rate at which reactivity can be added, to assure that reactivity accidents cannot cause a transient capable of damaging the reactor coolant pressure boundary or disrupting the reactor core, its support structures, or other vessel internals sufficiently to impair the core standby cooling system effectiveness. Fuel damage limits will not be exceeded for any reactivity transient resulting from a single equipment malfunction or operator error.

The inherent safety features of the reactor core design as described in Section 4.3, in combination with the control rod velocity limiter, control rod drive housing supports and the reactivity control system described in this section, are such that the consequences of a potential nuclear excursion accident, caused by any single component failure within the reactivity control system, by itself, cannot result in damage (either by motion or rupture) to the reactor primary coolant system. [4.6-1]

4.6.2 Reactivity Control Methods

The plant design contains two independent reactivity control systems. Operational control of reactivity is provided by a combination of moveable control rods, a variable reactor coolant recirculation system flow, and burnable neutron absorbers contained in the fuel. These systems accommodate load changes, fuel burnup, and long-term reactivity changes. Reactor shutdown by the control rod drive system is sufficiently rapid to prevent violation of fuel damage limits for all operating transients. A standby liquid control (SBLC) system is provided as a redundant, independent shutdown system to cover emergencies in the operational reactivity control system.

The control rod velocity limiter is an integral part of the bottom assembly of each control rod. This engineered safety feature protects against a high reactivity insertion rate by limiting the control rod velocity in the event of a rod drop accident. It is a one-way device in that the control rod scram velocity is not significantly affected but the control rod dropout velocity is reduced to a permissible limit. The control rod velocity limiter is discussed in Section 4.6.2.2. [4.6-2]

4.6.2.1 Control Rods

One hundred seventy-seven (177) moveable cruciform-shaped control blades are used in the reactivity control system. The control blades are capable of being positioned in 6-inch increments, by the control rod drive (CRD) system (see Section 4.6.3), to control the neutron flux distribution, the reactor power level, and to shutdown and maintain the reactor in a shutdown condition. Several different control blade designs have been placed in service.

Boron carbide (B_4C) is the preferred neutron absorbing material because boron-10 (B-10) has a very large neutron absorption cross section, approximately 3800 barns (as compared to Iron for example, which as an absorption cross section of approximately 160 barns), in the thermal energy range. Boron carbide is also able to withstand high temperatures. The boron carbide powder is compacted to approximately 70% of theoretical density. The free volume of approximately 30% provides a plenum for helium from the B(n,α)Li reaction. The boron carbide pins of the CR-99 control blade have a density greater than 70%. The free volume of the CR-99 control blade ensures stresses due to pressure differences across the walls remain below design values.

Hafnium in metal form is used as an alternate absorber material because it has excellent nuclear and material characteristics. Although lower in initial reactivity worth as compared to B_4C, hafnium has a slower depletion rate. The five principle isotopes of hafnium have absorption cross-sections for thermal neutrons ranging from 13.04 to approximately 370 barns. Since hafnium and its daughter products have these large absorption cross sections and several large resonance cross sections at slightly higher energies, the worth of hafnium remains constant for a long time. Placing hafnium in high neutron flux portions of the control blade can significantly extend the lifetime of the control blade. Hafnium is also less susceptible to swelling than B_4C due to its low thermal expansion coefficient and lack of internal helium pressure buildup. When exposed to high temperatures, steam, and radiation, hafnium is not readily susceptible to corrosion, thus reducing the susceptibility of the blade wings to stress corrosion cracking. However, hafnium is heavier than B_4C. The increased weight due to the use of hafnium is offset by the design improvements, which lower the weight of other components of the control blade.

A cruciform-shaped top assembly with a single plane handle aligns the absorber sections and provides structural rigidity at the top of the control blade. Rollers or pads attached to the top assembly maintain the spacing between the control blade and the adjacent fuel assembly channels. A similar connector assembly, which incorporates the velocity limiter, is located at the bottom of the control blade and contains rollers to position the lower part of the control blade in the control rod guide tube located below the core. These bottom rollers always remain in the guide tube during operation. A coupling at the bottom of the control blade is connected and locked to the control rod drive index tube by an expandable ball and socket joint.

The velocity limiter is a device, which is an integral part of the control rod and mitigates the impact of the low probability event of a rod drop accident. It is designed to limit the free fall velocity to less than 3.11 ft/s. This limits the reactivity insertion rate of a control blade so that minimal, if any, fuel damage would occur. It is a one-way device, in that the control rod scram time is not significantly affected.

The control blades are cooled by the core bypass flow. The core bypass flow is made up of core flow from several leakage flow paths.

The following subsections identify the key design features of the various control blade designs, which have been approved for use at this site.

4.6.2.1.1 Original Equipment Control Blades

The original equipment control blade, also known as DuraLife 100 (D-100), is shown in Figure 4.6-1. The control blade contains 84 (76) vertical stainless steel absorber tubes filled with B_4C powder evenly divided among the four wings. The B_4C powder is compacted to approximately 70% of solid density. A free volume of 30% is provided in each tube as a plenum for helium from the $B(n,\alpha)Li$ reaction. Plugs are welded into the ends of the tubes to seal them. In the tubes containing B_4C powder, stainless steel balls are used to separate the tubes into individual compartments. The stainless steel balls are held in position by a slight crimp in the tube. The tubes are held in the cruciform array by a stainless steel sheath extending the full length of the tubes. The OEM control blade weighs approximately 218 (186) pounds. This design has since been replaced with updated designs and is no longer available.

4.6.2.1.2 Advanced GE Control Blade Designs

DuraLife 120 (D-120)

The GE Improved control blade, also known as DuraLife 120, is similar in design and appearance to the OEM control blade, the only difference being that the absorber tubes are made with high-purity stainless steel rather than commercial 304 stainless steel. The purpose of the high-purity stainless steel is to improve resistance to intergranular stress corrosion cracking (IGSCC).

DuraLife 160 (D-160)

The GE Hybrid-I control rod (HICR) is also known as DuraLife 160. The HICR design configuration (e.g., upper handle, velocity limiter, and coupling socket) is similar to the OEM GE control blade assembly. However, the HICR control blade has three solid unclad Hafnium neutron absorber rods in the outer 3-locations of each wing. Based on statistical analysis, the negative reactivity worth of the HICR control blade is equivalent to the OEM GE all-B_4C control blade design. Since Hafnium is a heavier material than B_4C, the sheath wall thickness is reduced to maintain the same weight as the OEM control blade. The result is a control blade that weighs approximately 218 pounds. Also, the overall blade thickness is reduced slightly and, in turn, results in an increase in clearance to the fuel assembly channel. These factors ensure similar SCRAM insertion times as the OEM control blade. In addition, the roller and pin assemblies at the top and bottom end of the blades are made of non-cobalt alloys. Therefore, the roller-and-pin assembly is less of an activation and radioactive waste concern than the OEM control blade. More information on the DuraLife-160 control blade is provided in NEDE-22290.

DuraLife 190 (D-190)

The Advanced Longer Life Control Rod (ALLCR) design, also known as the Duralife 190, is an extension of the HICR design. It is designed to increase control blade assembly life and to further eliminate cracking of absorber tubes containing B_4C. This design is also compatible with reactor internals and existing site equipment. The ALLCR design differs from the HICR only in the following areas:

- There is a 6-inch Hafnium plate in the top portion of the absorber section of each wing. The Hafnium plate increases blade lifetime since the blade tip generally experiences the highest neutron flux.

- There is a modified velocity limiter, which weighs approximately 20 pounds less than the original design to compensate for the additional weight of the Hafnium absorber plates. SCRAM and normal movement performance is maintained.

- The overall weight (198 pounds) is slightly less than the weight of either the standard blade or the HICR design due to the lighter velocity limiter. The new design results in average rod drop velocities of less than 2.78 ft/s at operating conditions which is bounded by the design basis value of 3.11 ft/s, which is the assumed velocity in the rod drop accident analysis (see Section 15.4.10).

All other features of the HICR are retained, such as the high-purity, Type 304 stainless steel B_4C absorber tubing, the full-length Hafnium rods in the outer 3-locations of each wing, and the non-cobalt pin-and-roller material. More information on the DuraLife–190 control blade is provided in NEDE-22290, Supplement 2.

DuraLife 230 (D-230)

The DuraLife 230 control blade is a ‘D’ lattice design and has a basic design similar to the DuraLife 190. However, to increase the control blade lifetime beyond the D-190, the volume of absorber material in the absorber section was increased. This was accomplished by increasing the diameter of the absorber tubes and decreasing the tube thickness. In addition, the outer three-hafnium absorber rods were replaced with a hafnium strip of equivalent width. These changes resulted in an overall weight (217 pounds) slightly less than the weight of the OEM control blade. More information on the DuraLife 230 control blade is provided in NEDE-22290, Supplement 3.

Marathon

Another advancement in GE control blade designs is the Marathon (Figures 4.6-10 and 4.6-11). The primary difference between the Marathon control blade and the previous GE designs is the use of externally-square but internally-circular tubes that are welded full length to each other to form the absorber section of the four wings, which eliminates the need for a sheath. Each wing is comprised of 14 tubes with each tube acting as an individual chamber to hold the encapsulated B_4C and/or hafnium rods. Some of the square absorber tubes that contain B_4C capsules are also loaded with empty capsules to accommodate the helium release from the B_4C. The four wings are then welded to several short length tie rod sections to form the cruciform-shaped member of the control rod. The square absorber tube material is high purity RAD RESIST 304S stainless-steel that is similar to the material used in previous GE designs except that an alloying component has been added to provide additional resistance to irradiation-assisted corrosion cracking. These design features allow the Marathon to have an enhanced lifetime compared to the previous GE control blade designs.

The Marathon design is considered a match in reactivity worth to the OEM control blade design, which means that the reactivity is within $\pm$5% of the original equipment control blades design and thus no additional considerations need to be applied in core design or analysis.

The U.S. NRC SER for NEDE-31758P-A requires surveillance of the Marathon blades which addresses lead exposure control blades. Examinations to satisfy this requirement have been conducted, and no problems have been found during Marathon inspections to date through a peak segment fluence of 6.3 snvts. The weight of the Marathon control blade with the FabriCast velocity limiter has been maintained slightly less than the OEM control blade. This configuration maintains critical parameters to ensure compliance with the drop speed limit of 3.11 ft/s assumed in the rod drop accident analysis (Section 15.4.10). More information on the Marathon control blade is provided in the GE Control Rod Specification, "Technical Specification – Marathon," 24A5352 Rev. 2.

Marathon Ultra HD

The Marathon-Ultra (Ultra HD) design is derivative of the Marathon design. The absorber section load pattern of the Marathon Ultra HD has full length hafnium rods placed along the outer-edge, high depletion areas of the blades and boron carbide capsules at other locations. The Marathon Ultra HD utilizes the thin-walled boron carbide capsules used in the original Marathon design. This maximizes the neutron absorber mass.

4.6.2.1.3 Westinghouse ATOM AB Control Blades

The Westinghouse ATOM ABB control blades (Figure 4.6-2) nominally weigh between 213 and 222 pounds as compared to the OEM control blade, which weighs about 218 pounds. These control blades yield equal to or better scram insertion times than the OEM blade design. These control blades are mechanically compatible with the reactor and rod drive components in that the coupling and velocity limiter section is identical to the OEM control blades. Blade span, blade thickness, and handle features are also similar to those for the OEM blades. The dimensional tolerance envelope of the Westinghouse ATOM AB control blade is well within that of the OEM control blade.

CR-70

The Westinghouse ATOM AB CR-70 design wings are solid 304 stainless steel with horizontally drilled holes containing B_4C as the neutron absorber material. The B_4C is loaded into the horizontal holes and compacted to 70% of theoretical density. The absorber holes are closed at the outer blade edge by a stainless steel cover bar but are connected through a narrow slot. This slot allows Helium gas pressure equalization between the absorber holes without any significant displacement of the B_4C powder. Inserting a bar over the absorber holes seals the wing edges, which are then rolled together and welded. Horizontal holes are used to minimize the effect of any further B_4C densification after initial filling. Instead of a pin-and-roller arrangement as with the OEM control blades, the CR-70 utilizes a button at the top of the assembly. However, the rollers are still used on the velocity limiter. This blade design weighs approximately 213 pound (96.6 kg). The CR-70 design was used only in Dresden Unit 3 and is no longer in use or available.

CR-82

The CR-82 design improves upon the CR-70 design by incorporating Hafnium metal rods in the upper 6-inches of the absorber section. B_4C remains the neutron absorber material in the remainder of the absorption section. This blade design weighs approximately 218 (198) pounds (99 (90) kg).

CR-82B

The CR-82B design was a slight variation upon the CR-82 design by replacing the stainless steel hole closure bar in the hafnium zone (upper 6-inches) with a hafnium bar. This blade design weighs approximately 218 (198) pounds (99/90 kg).

CR-82M-1

The CR-82M-1 design incorporates improved stainless steel material (type 316 vs. original type 304) for the blade wings. The 316L SS wing material has been demonstrated to be more resistant to stress corrosion cracking as compared to 304 SS. This blade design weighs approximately 218 (198) pounds (99/90 kg). Westinghouse ABB is continuing to collect in reactor data on the performance of the CR82M-1 design.

CR-99

The Westinghouse CR-99 design is similar to the CR-82M-1 design with a change in the use of hot isostatic pressed B_4C pins as absorber material.

4.6.2.1.4 Mechanical Design of Control Rods

For the GE original equipment control rods, design stress intensity limits for control rod neutron absorber tubes are given in Table 4.6-1. For information on the mechanical design of the advanced control rod designs, refer to the topical report for that design.[1,2,3] [4.6-9]

A stress analysis was performed on a control rod similar to that used in this reactor. It was assumed that all control rod neutron absorptions were in Boron-10 (B-10). Based on experimental data , a value of 18% was used for the fraction of He gas, generated by the B (n,α) Li reaction, which is released from the B_4C to cause internal pressure within the neutron absorber tubes. When the nuclear life due to depletion of B-10 was reached, the internal pressure in the most highly exposed neutron absorber tube was 13,000 psi, and the resultant maximum general primary membrane stress was less than 50,000 psi compared to a design limit of 51,500 psi ($1/2S_u$) for irradiated material.

Operating experience has shown that the materials used in the control blades are not susceptible to significant dimensional distortion in service.

Control rod tubing is internally supported and can withstand external pressures far in excess of that experienced under accident conditions.

4.6.2.1.5 Control Blade Life

The end of control blade life is defined as the exposure when any quarter segment of control blade reaches a 10% reduction in relative worth. The reduction in control blade worth is due to a combination of B-10 depletion, transmutation of hafnium isotopes and, in original equipment control rods, the B_4C loss resulting from the cracking of the absorber tubes. [4.6-10]

Depletion occurs when a B-10 atom captures a neutron to form helium and lithium (i.e., $^{10}B + {}^{1}n \rightarrow {}^{7}Li + {}^{4}He$). Thus, for each B-10 neutron capture, the number of B-10 atoms is reduced. In control blades utilizing hafnium, hafnium transmutation is expressed in terms of B-10 equivalent depletion. Washout has been observed in original equipment control blades irradiated to exposures greater than 50% of the B-10 depletion lifetime criteria. The B_4C in original equipment control blades is in a powder form packed in stainless steel tubes. At high exposures in the reactor, these tubes have been observed to crack thereby allowing the reactor coolant to wash-out the local B_4C. Therefore, an acceleration term is used in determining the recommended lifetime for original equipment control blades. The end of control blade life (i.e., 10% reduction in relative worth) for the different control rod designs is provided in the appropriate lifetime documents.[4, 5, 10]

The on-line core monitoring code, POWERPLEX, calculates control blade exposure in snvts (nvt * 10^{21}) on both nodal and quarter segment basis. These control blade exposures are then compared to appropriate lifetime limits for the various control blade types in the core.

The original equipment control rods are controlled with lifetime and reshuffling criteria to account for blade cracking and B_4C leaching. [4.6-11]

4.6.2.2 Control Rod Velocity Limiters

4.6.2.2.1 Design Bases

The purpose of the control rod velocity limiter is to reduce the consequences in the event a high-worth control rod became detached from its drive mechanism and dropped out of the reactor core. To accomplish this purpose the velocity limiter was designed using the following bases: [4.6-12]

A. The control rod free fall velocity shall be less than 3.11 ft/s.

B. The velocity limiter effect on control rod scram time or positioning ability will be minimized.

C. The velocity limiter will be integrally attached to the control rod structure.

Refer to Section 15.4 for the definition and safety analysis of the control rod drop accident.

4.6.2.2.2 Original Velocity Limiter

A description and evaluation of the original control rod velocity limiter was submitted separately to the Atomic Energy Commission under APED-5446, Control Rod Velocity Limiter,[9] and is incorporated herein by reference.

4.6.2.2.2.1 System Design

The velocity limiter assembly consists of a single Type 304 stainless steel casting in the shape of two nearly-mated conical elements. These elements are separated from one another by four radial spacers. The separated surfaces of the upper and lower conical elements differ by 15 degrees, with the peripheral separation less than the central separation. [4.6-13]

The original velocity limiter assembly, shown in Figure 4.6-3 with its associated components, acts within a cylindrical guide tube. The annulus between the guide tube and the velocity limiter assembly permits the free passage of water over the smooth surface of the cone when the control rod is scrammed in the upward direction. In the opposite direction, however, water is trapped by the lower cone and discharged through the interface between the two conical sections. Because this water is jetted in a partially reversed direction into water flowing upward in the annulus, a severe turbulence is created, thereby slowing the decent of the control rod and limiter assembly.

The guide tubes are 10-inch, schedule 10, Type 304 stainless steel pipe. Each guide tube has a backseat on the lower end which rests on the control rod drive housing. This seat restricts water flow out of the tube during a velocity limiter free-fall; the seat also restricts water flow into the interior of the guide tube during normal reactor operation to prevent coolant bypass of the fuel elements.

4.6.2.2.2.2 Design Evaluation

During the development of the original velocity limiter, sensitivity tests were performed to assess the effect of manufacturing tolerances in the following items on the velocity limiter performance: Limiter and guide tube diametral tolerance; Nozzle (interfacial gap between cones) gap; Top cone thickness; Limiter/guide tube eccentricity; and Surface finish. These tests and the optimization of the velocity limiter design are described in detail in APED-5446, Control Rod Velocity Limiter.[9] The results of these tests are summarized as follows: [4.6-14]

Dropout Velocities	
Cold reactor	2.46 ft/s.
Hot reactor	2.86 ft/s.

Scram Times	
10% of full insertion	0.33 seconds.
90% of full insertion	3.05 seconds.

4.6.2.2.3 Replacement Velocity Limiters

The replacement of B_4C absorber material with hafnium metal results in an increase in control rod weight. To compensate for the additional weight, GE developed a lightweight velocity limiter for incorporation into the design of the Duralife-190 and Duralife-230 control rods. The lightweight velocity limiter uses principles similar to the original velocity limiter. The design of lightweight velocity limiter is considered proprietary to GE. A description and evaluation of the lightweight velocity limiter can be found in NEDE-22290 Supplement 2.[1] The velocity limiter of the ABB-ATOM control rod is the same design as the original GE equipment. [4.6-15]

The Marathon and Marathon HD control rods include the FabriCast Velocity Limiter (VL), which was designed to replace those used in the DuraLife – 190 and DuraLife – 230 Control Blades. The FabriCast VL consists of a cast vane section, a SS316 bar “transition piece” body section, and four fins made from a SS316 plate material. These individual sections are welded together to produce the FabriCast VL. The FabriCast VL retains the same critical dimensions and weight as the velocity limiter used on the original equipment design, and therefore does not impact the necessary control rod drop velocity of 3.11 ft/s.

4.6.2.3 Burnable Neutron Absorbers

See section 4.2 for information on the use of gadolinium in the fuel.

4.6.2.4 Recirculation Flow Control

The recirculation flow control system is discussed in Section 7.7.

4.6.2.5 Standby Liquid Control System

The standby liquid control system is discussed in Section 9.3.5.

4.6.3 Information for Control Rod Drive Systems

4.6.3.1 Control Rod Drive System Design

The control rod drive (CRD) system controls changes in core reactivity by incrementally positioning neutron absorbing control rods within the reactor core in response to manual control signals. It is also required to quickly shutdown the reactor (scram) by rapidly inserting control rods into the core in response to a manual or automatic signal. The control rod drive system includes the CRD mechanisms and the CRD hydraulic system. [4.6-16]

4.6.3.2 Control Rod Drive Mechanism

The CRDs are of the locking piston type. An assembly drawing of the drive mechanism is shown in Figure 4.6-4. The drive mechanisms are mounted vertically in housings which are welded into the reactor bottom head penetrations. The low end of each housing terminates in a special flange which contains ports for attaching the hydraulic system lines, and a machined face which mates with a corresponding flange at the lower end of the drive. [4.6-17]

At the top end of the stainless steel index tube (the moveable element), a multi-fingered coupling spud is provided which engages and locks into a socket at the base of the control rod (Figure 4.6-5). The fingers are made of Inconel-750, aged to produce maximum physical strength, and protected by a thin vapor – deposited chromium plating for increased wear life. The weight of the control rod alone will engage and lock this coupling. Once locked, the drive and rod form an integral unit which must be manually unlocked before a drive or rod can be removed from the reactor. This mechanism is established to prevent accidental separation of control rod from control rod drive.

The drives position the control rods in 6-inch increments and hold them in a discrete latch position until actuated for movement by the hydraulic system to a new position. The normal rate of withdrawal or insertion (other than scram) of the control rods is set at 3 in/s $\pm$ 20%. Therefore, the nominal rate of positive or negative reactivity insertion due to control rod positioning is about 0.001 $\Delta k/s$ (except under scram conditions). The maximum control rod drive withdrawal speed is 5.14 inches/second when the Operating Limit MCPR established in the Core Operating Limits Report (COLR) is set greater than or equal to the value corresponding to a RWE – at Power analysis for an "unblocked" condition. See subsection 15.4.2. Visible indication of the position of each drive is displayed in the control room by means of illuminated numerals which correspond with the respective latched positions. In addition, indication is provided that shows when insert and withdraw overtravel limits on the drive have been reached. Control rod seating at the low end of the stroke prevents the withdraw overtravel limit from being reached unless the control rod is uncoupled from the drive. This allows the coupling to be checked. These indicators and the rod position indicators are arranged on the control panel to mimic the rod radial positions in the core. [4.6-18]

4.6.3.3 Control Rod Drive Hydraulic System

The CRD hydraulic system controls the pressure and flow to and from the drives through the hydraulic control units (HCUs) to provide for cooling and control rod motion. Under normal operation, the CRD hydraulic system as shown in FSAR Figure 4.6-6 and P&ID M-41 uses unheated condensate supplied by one of two drive system pumps as the working fluid to accomplish hydraulic positioning of the control rod drives and their attached control rods. Pressure and flow for scram of the control rods is supplied by stored energy in the scram accumulators and by reactor vessel pressure. Scram capability of these sources is shown in P&ID M-41. The water discharges from a CRD discharge header to the corresponding scram discharge instrument volume. [4.6-19]

4.6.3.3.1 Design Bases

The CRD hydraulic system is designed to achieve the following objectives: [4.6-20]

A. Provide a water source at a pressure of approximately 1500 psig for charging the scram accumulators.

B. Provide a water source at a constant pressure nominally 260 psig above reactor pressure to move the CRDs.

C. Provide a water source at a pressure approximately 5—15 psi above reactor pressure to supply cooling water for each CRD mechanism at a constant flow rate.

D. Provide a water source for the recirculation pump seal purge system.

4.6.3.3.2 System Description

The CRD hydraulic system, shown in FSAR Figure 4.6-6 is made up of supply pumps, filters, strainers, control valves, and associated instrumentation and controllers. The system draws water from a line common to the contaminated condensate storage tank (CCST), condensate reject, and fuel pool reject. The water is filtered, pressurized, passed through another filter, then regulated through a flow control valve (FVC) and a pressure control valve (PCV). This flow path develops the pressures required for accumulator charging, CRD operation and CRD cooling. [4.6-21]

The CRD pump supplies pressure nominally at 1500 psig to the scram subsystem for charging the scram accumulators while the system flow control valve maintains a constant flow rate to the drive water pressure reducing valve. The pressure control valve develops a constant pressure drop due to the constant flow. The drive water pressure control station and cooling water station are connected in series. The drive pressure control station is adjusted to maintain pressure of nominally 260 psi above reactor pressure, and supplies water for normal drive operation. Cooling water flow to the drive mechanisms is maintained at approximately 60 gpm by adjustment of the cooling water FCV. This flow typically yields a cooling pressure approximately 5-15 psi above reactor pressure. Cooling water pressure is not adjusted by the cooling water station because of the removal of the CRD return line. Cooling water pressure is the result of the flow rate and the flow resistance of the CRD cooling water orifices plus the CRD seal leakages. [4.6-22]

A directional control valve and scram accumulator are provided in each CRD hydraulic control unit. The directional control valve consists of a four-way valve system which directs CRD hydraulic water to the area above the CRD piston and vents the area below for CRD withdrawal. For CRD insertion, water is admitted to the area below the piston, while the area above is vented. When a reactor scram is initiated, the scram inlet valve is opened to admit pressurized water from the accumulator to the area below the CRD piston, and the scram outlet valve is opened to vent the area above the CRD piston to the scram discharge volume, thereby causing rapid control rod insertion.

4.6.3.3.2.1 Supply Pump

One CRD supply pump pressurizes the system. A spare pump is provided as a 100% capacity standby unit. Changeover from one pump to the other can be performed by the operator in the control room. Each pump is equipped with a suction strainer, filter, and appropriate isolation valves to permit pump maintenance. The CRD pump discharge lines are crosstied to enable one CRD pump to supply both units or one pump to supply the opposite unit. [4.6-23]

The CRD pumps take suction from the CCST; however, condensate reject also taps into this line. Since condensate is normally rejected at 60 gal/min, the CRD pumps actually pump rejected condensate during normal operation. The system can also be aligned to take suction from the fuel pool reject. [4.6-24]

Minimum-flow bypass connections between the discharge of the pump and the suction line connected to the CCST prevents the pump from overheating if the pump discharge valve is inadvertently closed. The pump discharge pressure is indicated by a local pressure gauge. [4.6-25]

Two parallel filters are provided to remove foreign material larger than 50 microns (25 microns nominal) from the CRD hydraulic system water. Normally one filter is valved in while the other is in standby; however, occasional operation with flow through both filters is allowed. Either filter can be vented, drained and cleaned for reuse while the other is in service. A differential pressure indicator and an alarm monitor the filter element as it collects foreign material. Strainers in the filter discharge lines protect the CRD hydraulic system in the event of a filter element failure.

4.6.3.3.2.2 Accumulator Charging Pressure

The accumulator charging pressure is established by the throttling of the CRD pump discharge valve and is independent of reactor pressure. The accumulator charging header taps off the outlet of the drive water filters downstream of the flow-sensing controller but before the flow control valve. On a scram, the scram inlet valves open causing the accumulators to discharge to the area below the drive pistons. The scram outlet valves allow the water above the pistons to discharge to the scram discharge headers. The accumulators will not recharge until the scram is reset. The pressure in the header is monitored in the control room with a pressure indicator and low pressure alarm. [4.6-26]

4.6.3.3.2.3 Flow Control Station

The flow control station maintains a nearly constant flow rate through the downstream drive water pressure control station and cooling water header. The flow is maintained automatically by a flow-sensing controller and by an air-operated flow control valve. A parallel spare valve is provided to permit valve maintenance.

4.6.3.3.2.4 Drive Water Pressure Control Station

The drive water pressure is maintained at approximately 260 psi above the reactor vessel pressure by the combined operation of the drive pressure control valve and the stabilizing valves. The motor-operated pressure control valve is remotely adjusted from the main control room and is used to adjust the upstream pressure. A manual bypass valve and isolation valves are provided to permit maintenance of the motor-operated pressure control valve. [4.6-27]

4.6.3.3.2.4.1 Drive Water Header

The drive water header taps off after the flow control valve. It supplies drive water to each CRD HCU for inserting and withdrawing. The drive water header flow element and indicator are used to measure flow to the drives for adjustment and testing. A differential pressure indicator in the main control room and a local indicator show the differential pressure between the reactor vessel and the drive water header. This pressure indicator is used when adjusting the drive water pressure with the motor-operated drive water pressure control valve. [4.6-28]

The CRD drivewater header also provides the source of water to the Reactor Vessel Level Indication System (RVLIS) Backfill Subsystem. This water is clean and deaerated to prevent the buildup of non-condensable gases in the instrument reference legs.

4.6.3.3.2.4.2 Stabilizing Valves

Solenoid-operated stabilizing valves are used to divert flow for rod insertion or withdrawal. A stabilizing valve closes when the "insert" valves for any drive are actuated; a second valve is closed when the "withdraw" valves for any drive are actuated. In this manner, the flow through these valves always balances the flow to the drives through the 1-inch drive water header, the flow through the motor-operated valve is virtually constant, and the required pressure is maintained in the 1-inch drive header. The variation in flow requirements between drives is small enough that the corresponding pressure variation is within acceptable limits. A standby set of stabilizing valves is provided as a backup in the event of a failure of the operating set. [4.6-29]

Filters are installed before the stabilizing valves to prevent fouling of the valves. Isolation valves are provided for the maintenance of the stabilizing valves. A flow element and an indicator are installed for measuring the flow through the stabilizing valves so that the valves can be adjusted to provide the required flow for normal drive operation.

4.6.3.3.2.5 Cooling Water Header

The cooling water header taps off after the drive water pressure control valve. The cooling water is maintained at approximately 5—15 psi above reactor pressure as a result of the constant flow through the flow control valve and adjustment of the drive pressure control valve. With no rod motion, drive pressure changes are normally required only to compensate for changes in drive seal and pump flow characteristics over time. Cooling water pressure is not adjusted independently. Rather, it is a function of drive water pressure and cooling water FCV setting. [4.6-30]

The cooling water header supplies cooling water to the 177 HCUs and their associated drives. With no rod motion in progress all flow through the flow control station passes through the cooling water header. A check valve opens to admit cooling water to the underside of the drive piston when the drive is stationary. Scram performance is also affected per SIL 173 and SIL 173 Supplement 1. The seal life is also shortened by any foreign material in the system that could cause abrasion. When a drive is in motion, the pressure under the piston is higher than the cooling water pressure, and the check valve is closed.

The cooling water is monitored by a flow indicator. A differential pressure indicator shows the difference between reactor pressure and cooling water pressure.

4.6.3.3.2.6 Exhaust Header

The exhaust header receives water discharged by the drives during normal control rod movement (not a scram). The return flowpath to the reactor vessel is through the exhaust header and then through the insert exhaust directional control valves of nonactuating CRD HCUs. [4.6-31]

4.6.3.3.2.7 Hydraulic Control Unit

4.6.3.3.2.7.1 Accumulator

The accumulator for each drive is an independent source of stored energy to scram that drive. [4.6-32]

The accumulator is a piston type accumulator. The lower side is connected to a N_2 cylinder and the upper side is aligned to the charging water header. Under normal conditions the accumulator is charged and the piston is in the full down position. The piston serves as a barrier between the high-pressure N_2 and the water used to initiate a control rod scram. The accumulator pressure is maintained $\geq$ 940 psig to ensure that adequate scram insertion capability exists over the entire range of reactor pressure.

To assure that it is always capable of producing a scram, the accumulator is continuously monitored for water leakage and for nitrogen pressure. A float-type level switch will actuate an alarm if water leaks past the piston and collects in the N_2 side of the accumulator. A pressure indicator and a pressure switch are connected to the accumulator to monitor nitrogen pressure. A decrease in nitrogen pressure will actuate the pressure switch and sound an alarm. An isolation valve allows each of the accumulator instruments to be isolated and serviced.

4.6.3.3.2.7.2 Scram Pilot Valves

Each control rod is equipped with two scram valves controlled by either two individual scram solenoid pilot valves (SSPVs) or one SSPV with two solenoid coils. During normal operation, each of the two parallel Channels of the reactor protection system (RPS) logic energizes one of the SSPV solenoid coils. During normal operation, both SSPV solenoids are energized and the instrument air is ported to the two scram valves, holding them closed. During a full scram, both of the RPS Channels de-energize, de-energizing both SSPV coils which vents both of the scram valves' operators, thus allowing the scram valves to open. To protect against spurious scrams, both SSPV coils must be de-energized to vent the scram valve operators. Failure of electric power to both solenoids, or failure of the Instrument Air (IA) supply to the hydraulic control unit (HCU), will cause the affected control rod to insert. The SSPVs are selected based on simplicity of design, a minimum of moving parts, fast opening time (approximately 0.050 seconds) and satisfactory statistical operating history on similar units. [4.6-33]

For added protection, the instrument air header to the scram dump valves and all the scram pilot valves have a pair of backup scram valves. Upon a scram signal, these valves close off the air supply and vent the section of the instrument air system header between the pilot valves and the backup scram valves. This will scram all drives should any of the scram pilot valves fail to vent. [4.6-34]

4.6.3.3.2.7.3 Inlet/Outlet Scram Valves

A pair of scram valves isolate the control rod drive from the HCU accumulator and SDV during normal operation. The inlet scram valve is a globe valve which is opened by the force of an internal spring and closes when air pressure is applied on top of the diaphragm operator. The opening force of the spring is approximately 700 pounds. Each valve has a position indicator switch which energizes a light in the control room when the inlet and outlet valves are open. The scram valve is selected based on high operating force, fast opening time (approximately 0.1 second) and satisfactory operating history on similar units. [4.6-35]

The outlet scram valve is identical in construction to the inlet scram valve, but smaller. The internal spring preload in the outlet scram valve is slightly greater than the inlet scram valve to produce a faster opening time.

4.6.3.3.2.7.4 Directional Control Valves

Four solenoid-directional control valves (DCVs) are used for switching the drive water header and the exhaust header to the two drive ports. By energizing and opening two valves at a time, the drive water header can be connected either under or over the CRD piston while the exhaust header is connected to the opposite side. Two directional control valves, which include speed control valves, are connected so that they always pass the flow to or from the underside of the CRD piston. Proper speed for control rod insertion and withdrawal is obtained by timing individual rod motion and adjusting the respective speed control valve so as to obtain a normal drive speed equivalent to 3 in/s +/- 20%. Drive flow to a rod moving at normal speed is approximately 4 gal/min. The balance of forces in the drive mechanism is such that the pressure under the piston is approximately 90 psi whenever the drive is either inserting or withdrawing. Thus, for control rod insertion, a pressure drop of 170 psi exists across DCV (from 260 psi in the drive-water header to 90 psi under the piston). Similarly, for control rod withdrawal, normal flow (normal rod speed) creates an 85 psi drop across DCV (from 90 psi under the piston to 5 psi in the exhaust header). The DCVs are protected from dirt by filters. [4.6-36]

The two directional control valves connected to the drive water header can be forced open by scram pressure on their outlet ports. A check valve in the drive water riser prevents significant reverse flow into the drive water header during scram. A check valve in the cooling water riser similarly prevents significant reverse flow into the cooling water header during scram. [4.6-37]

4.6.3.3.2.8 Scram Discharge Volume

The scram discharge volume (SDV) is used to limit the loss of, and contain the reactor vessel water from all the drives during a scram. The SDV consists of the header piping leading from each HCU scram outlet valve to the instrument volumes. There are two scram discharge headers, one for each bank of HCUs and two instrument volumes, one for each scram discharge header. The header piping is sized to receive and contain all the water discharged by the drives during a scram independent of the instrument volumes. [4.6-38]

During normal operation, the SDV is empty, with both its double isolation drain and vent valves open. These drain and vent valves operate very much like the scram valves. With a scram signal, the RPS is de-energized and the two scram dump (solenoid pilot) valves open venting the drain and vent valves operators, causing them to close. Local position indicator switches on the main valves indicate the position of the vent and drain valves. A test pilot valve allows the SDV vent and drain valves to be tested without disturbing the RPS.

During a scram, the SDV partly fills with the water from the over piston area of the drives. While scrammed, the CRD seal leakage continues to flow to the SDV until the discharge volume pressure equals reactor vessel pressure or until the scram valves are reset. When the scram signal is removed from the RPS, the scram valves may be closed and the discharge volume may be drained. The two instrument volumes drain into the drain header of the reactor building equipment drain tank. A control system interlock will not allow the drives to be withdrawn until the instrument volume is emptied to below the rod block set point.

The scram discharge instrument volumes provide for the measurement of the volume of water discharge from the CRDs to the SDV for alarm and control purposes. There are a series of diverse and redundant level switches connected to each instrument volume. There are six level switches. Two of the level instruments are of the differential pressure type which measure the differential pressure between the top and bottom of the instrument volume. Two of the other four level instruments are of the thermal type and the remaining two are of the float-type. The alarm and control functions of these instruments are as follows:

A. High Level Alarm

   One of the thermal instruments is used to provide a high level alarm at approximately 10 gallons.

B. Rod Block

   A second thermal instrument provides a rod withdrawal block at approximately 25 gallons.

C. Reactor Scram

Should the SDV fill up with water to the point where not enough volume remains for the water displaced during a scram, control rod movement could be hindered in the event a scram were required to prevent this situation.

1. Differential pressure type instruments are used to provide an alarm and initiate a reactor scram at an instrument volume of approximately 40 gallons.

2. Two float-type switches are used to provide an alarm and initiate a reactor scram at an instrument volume of approximately 40 gallons.

4.6.3.3.2.9 Alternate Rod Insertion Valves

The alternate rod insertion (ARI) scram valves provide an alternate means of initiating control rod insertion during an anticipated transient without scram (ATWS) event. The ARI system and other ATWS related systems are discussed in Section 7.8. [4.6-39]

4.6.3.4 Control Rod Drive System Operation

4.6.3.4.1 Insertion/Withdrawal

Control rod drive insertion is accomplished by opening both insert valves in the directional control valve circuit. This applies drive pressure to the underside of the drive piston and permits water displaced from above the piston to exit to the exhaust header. The driving pressure acting on the drive piston area of 4.1 in.^{2} exerts a maximum upward force of up to 1000 pounds, which exceeds the force of friction plus the weight of the control rod coupled to the CRD index tube, so that the CRD inserts the control rod. [4.6-40]

The insert valves also open for approximately 1/2 second during a withdrawal operation in order to slightly insert the drive and unload the collet so it can be unlocked to allow withdrawal. After this, the pair of withdraw valves are opened to apply drive pressure above the drive piston (1.2 in.^^{2}^ area) and to open the area under the piston to the exhaust header. When the notch withdrawal mode of operation is selected by the operator, the proper pair of valves are electrically energized long enough to allow the drive to move to the next notch position at which time the valves are automatically de-energized even if the operator holds the switch closed.

If all four directional control valves are closed while the drive is in a position between notches, water displaced by the drive piston must leak past the drive seals in order for the drive to settle into the latched position. With only seal leakage this settling speed would be a fraction of normal withdrawal speed. To speed up the settling and latching of a drive following an insert or withdrawal movement, the closing of the under piston withdraw valve is delayed for several seconds to facilitate the displacement of water from the drive. This allows the drive to settle at about 1/2 normal speed to the next latch position.

Normal withdrawal speed is determined by differential pressures at the drive, and is set for a nominal value of 3 in/s ± 20%. The characteristics of the pressure regulating system are such that this speed is maintained independent of reactor pressure. Tests have determined that accidental opening of the speed control valve to the full open position will produce a velocity of approximately 6 in./s. Should this system fail, thus producing maximum available pump pressure (1750 psig) to the drive system with zero reactor pressure, the hydraulic resistances in the system would limit the withdrawal velocity to 2 ft/s. The maximum control rod drive withdrawal speed is 5.14 inches/second when the Operating Limit MCPR established in the Core Operating Limits Report (COLR) is set greater than or equal to the value corresponding to a RWE – at Power analysis for an "unblocked" condition. See subsection 15.4.2. [4.6-41]

The allowable operating limits on withdrawal and insertion speed are determined by requirements for the insert-before-withdraw motion and for jogging. These limits are lower than those which might be set by considering the maximum allowable reactivity variations. The jog withdrawal operation of the drive is an excellent test of the correctness of the speed setting; the drive generally will fail to withdraw if the speed is incorrectly adjusted. A pressure of approximately 60 psi higher than reactor pressure must be maintained above the main drive piston in order to keep the collet unlocked. Any malfunction which allows the pressure to drop below this value, a condition necessary for higher withdrawal speeds, results in collet locking.

During reactor shutdown, and with fuel loaded into the core, all control rods are normally inserted. Interlocks are provided which prevent the inadvertent withdrawal of more than one control rod with the mode switch in the refuel position. The refuel mode interlocks are addressed in Section 7.7.1.2. [4.6-42]

4.6.3.4.1.1 Operational Reliability

Each drive mechanism has its own complete set of electrically-operated directional control valves, which are closed when de-energized. The correct operation of all four valves in the correct sequence is required to cause the drive to withdraw. Consequently, the probability of multiple simultaneous independent valve failures that could cause accidental multiple rod withdrawals is extremely small. The electrical system which actuates the directional control valves is designed to prevent any credible failure from producing accidental movement of more than one control rod. [4.6-43]

High operational reliability contributes generally to overall safety by minimizing the occasions when abnormal operating conditions are encountered. High operational reliability is the objective of the following features of the control rod drive hydraulic system.

A. Components in the hydraulic system are picked based on established reliability. A spare pump and control valves are provided for reliability. Operating valves are accessible for maintenance while the reactor is in operation.

B. Provisions are made to operate with a reasonable amount of foreign material in the reactor water and in the water supplied to the hydraulic system. Filters and strainers are incorporated in the drive mechanism in passages through which water is drawn into the mechanism.

The operating pump and flow control valve are monitored for proper operation. When signs of decreased operational performance are identified, such as excessive seal leakage or abnormally high vibration, the standby equipment will be alternated into service, thereby improving overall performance of the CRD system.

Instrumentation and alarms monitor operation of flow and pressure regulation to assure availability of drive water and cooling water.

Operation of drive control, scram and scram pilot valves are observed during periodic testing of control rod drive operation and during scram tests.

4.6.3.4.2 Scram Operation

During a scram, the inlet and outlet scram valves associated with each CRD open, admitting accumulator pressure (approximately 1500 psi) to the underpiston area of the CRD and venting the overpiston area of the CRD to the scram discharge volume. The large differential pressure (initially about 1500 psi and always several hundred psi depending on reactor pressure) produces a large upward force on the index tube and control rod, to overcome any possible friction or binding. This initial scram force is a maximum of 6000 pounds under cold reactor conditions and at least 2800 pounds when the reactor is at operating pressure and greatly exceeds the normal insert resisting force of the control rod. The characteristics of the hydraulic system are such that after initial acceleration (less than 30 milliseconds after start of motion) the desired scram velocity of about 5 ft/s is achieved and the drive travels at a fairly constant velocity. This characteristic provides a high initial rod insertion rate and a high operating force margin. As the drive piston nears the top of its stroke, the piston seals progressively close off internal drive ports restricting exhaust flow and reducing the drive velocity. Each drive requires about 2.5 gallons of water during the scram stroke. [4.6-44]

There is adequate water capacity in each drive's accumulator to complete a scram in the required time at low reactor pressures. At higher reactor pressures, the accumulator is assisted by reactor pressure reaching the drive through a ball check valve located in the drive itself. As water is drawn from the accumulator, the accumulator discharge pressure falls below reactor pressure. This causes the ball check valve to reposition and admits reactor pressure to the underpiston area. Thus, reactor pressure furnishes the force needed to complete the scram stroke at higher reactor pressures, while the accumulator alone will accommodate the low-pressure scrams. When the reactor is up to full operating pressure, the accumulator is not needed to meet scram time requirements. Typical scram time characteristics are shown in FSAR Figure 4.6-7.

4.6.3.4.2.1 Rate of Scram Response

Under conditions of abnormal reactor system disturbances, the reactivity control system provides a sufficient rate of negative reactivity insertion, upon a signal from the reactor protection system, to prevent fuel damage. Abnormal reactivity disturbances and resulting power transients in the core originate from any of three sources. These are: [4.6-45]

A. Reactor system disturbances of core parameters such as coolant flow or pressure;

B. Single operator errors or procedural violations; or

C. Single equipment malfunctions.

The RPS described in Section 7.2.1 senses the disturbances and, under certain specified conditions, will initiate a scram signal. Upon receipt of a scram signal the reactivity control system is required to render the reactor subcritical at a rate sufficient to prevent the initiating disturbance from causing fuel damage.

The Technical Specifications place limitations on the number, as well as the relative location, of operable control rods which are considered "slow". The following scram time limits are utilized for the determination of a "slow" control rod (scram time limits based on de-energization of the scram pilot valve solenoids as time zero). [4.6-46]

A. 5% of maximum stroke, 7.2 inches, in 0.48 seconds maximum.
B. 20% of maximum stroke, 28.8 inches, in 0.89 seconds maximum.
C. 50% of maximum stroke, 72.0 inches, in 1.98 seconds maximum.
D. 90% of maximum stroke, 129.6 inches, in 3.44 seconds maximum.

4.6.3.4.2.2 Scram Reliability

High scram reliability is the object of a number of features in the system such as the following: [4.6-47]

A. There are two sources of scram energy (accumulator and reactor pressure) which complement each other for each drive whenever the reactor is operating.
B. Each drive mechanism has its own scram valves and pilots so that only one drive can be affected by a scram valve failure to open. A separate backup pilot valve is provided to vent the instrument air header scramming all drives (after some time delay) should this failure occur.

C. Under scram conditions the drive mechanism develops 6000 pounds (at zero reactor pressure) to 2800 pounds minimum (at rated pressure) of force, providing a large margin to overcome possible friction.

D. The scram system is designed so that the scram signal overrides all others.

E. The scram valves fail open on loss of either air or electrical power. Hence, failure of the valves' air system or electric system will produce, rather than prevent, a scram. All components used in the scram hydraulic system are selected either after an extensive testing program or after many millions of accumulated operating hours in service.

F. The ARI system provides an alternate path for reactor shutdown in the event that the normal scram path cannot be initiated by RPS. The ARI system is diverse and independent from RPS. [4.6-48]

4.6.3.5 Control Rod Drive Housing Support

4.6.3.5.1 Design Bases

Control rod drive housing support is an engineered safety feature provided to prevent ejection of a control rod from the reactor core in the event a control rod drive housing should fail. The reactivity addition associated with a sudden control rod ejection could exceed the threshold of fuel cladding rupture. To obtain a margin of safety, there must not be a failure of a control rod drive housing associated with the control rod drive mechanism which would permit a significant movement of the rod and its drive at high velocities. To achieve this margin, the housing support design was based upon permitting less than 3 inches of total control rod motion; this is less than a normal withdrawal increment. [4.6-49]

4.6.3.5.2 System Design

The support system consists of structural members (beams), hanger rods, grid plates, support bars and disc springs. [4.6-50]

Figure 4.6-8 presents a cutaway view of the support system to illustrate the various components. The beams are placed between the rows of housing, immediately below the bottom head of the reactor vessel. These beams are supported on the reinforced concrete pedestal which supports the reactor vessel.

The grid plates are located under the drive flanges. These grid plates are attached to the hanger rods which are supported from the beams. A stack of disc springs is provided on each rod. The support system, therefore, is an elastic structure which is capable of absorbing the energy resulting from the assumed failure. This system also limits the magnitude of the resulting dynamic forces on the supports. Turning moments on the rods are prevented by the support bars.

4.6.4 Evaluations of the Control Rod Drive System

4.6.4.1 Scram Effect

The rod insertion time profile for scram, combined with the reactivity worth profile of the control rods, provides the basis for evaluating the time dependence of negative reactivity insertion and its impact on terminating postulated power excursion events.

The rod position vs. time curve used in the transient analyses lies on the conservative side of the mean scram time data (discussed in Section 4.6.3.4.2.1) i.e., on the slow side of the data spread but faster than the specification given in Section 4.6.3.4.2.1. The analysis of the limiting power transient shows that the negative reactivity rate resulting from a scram, with an average response of all the drives, provides the required protection and the minimum critical power ratio remains greater than the fuel cladding integrity safety limit. The cycle-specific transient analyses are performed using NRC-approved methodology. [4.6-51]

Therefore, the response of the reactivity control system on signal from the RPS in combination with the size, heat transfer features, and inherent dynamic response characteristics of the core, prevent fuel damage resulting from a reactivity insertion accident due to any single equipment malfunction or single operator error.

4.6.4.2 Control Rod Drive Uncoupling and Control Rod Drop

The coupling mechanism connecting each control rod to its CRD, which allows for removal of either component for maintenance or replacement, also creates the potential for accidental decoupling of a control rod from its CRD and subsequent drop of a control rod from the active core region. [4.6-52]

The consequences of a postulated control rod drop accident (CRDA) are ultimately mitigated by the control rod velocity limiters, discussed in Section 4.6.2.2. The use of planned control rod withdrawal sequences, enforced by the rod worth minimizer (addressed in Section 7.7) or an independent verifier, further reduces the potential consequences of certain CRDA events. The analysis of CRDAs is addressed in Section 15.4.10.

4.6.4.3 Control Rod Drive Ejection

The CRD housing support system was installed with a gap of about 1 inch between the lower grid clamps and the contact surface on the control rod drive flanges. During system heatup this gap is reduced due to a net downward expansion of the housings with respect to the grid clamps. In the hot operating condition, the gap will be approximately 1/4 inch. [4.6-53]

Downward travel of the housing following an assumed housing failure will be the sum of the initial gap, plus the elastic deflection of the supporting structure under dynamic loading. The support system will limit the total downward movement of the drive and housing to 3 inches under the worst case assuming an initial gap of approximately 1 inch. Total deflection will normally be substantially less than 3 inches because an operating gap of 1/4 inch exists between the lower grid clamps and the contact surface on the CRD flange. Thus, the drive movement following a housing failure, will always be less than 1/2 one normal drive "notch" position.

A number of different arrangements regarding the size of the hanger rods, the number of springs and the stiffness of the support beams and grid clamps were investigated to determine the optimum design for this system. The design stresses for the control rod housing support structure components were limited to 90% or less of the yield strength of the materials. The stress criteria was selected to provide a system that is as elastic as possible and which can be considered adequate for the loading condition.

4.6.4.4 Collet Housing Failure

The collet assembly serves as the index tube locking mechanism of the CRD mechanism. It contains fingers which engage a groove in the index tube when the drive is locked in position. The collet housing surrounds the collet and spring assembly. The collet housing is a cylinder with an upper section of wall thickness 0.1 inches and a lower section with a wall thickness of about 0.3 inches. The collet housing is 304 stainless steel with the inner surface nitrided for wear resistance. Cracks have been found on the outer surface of the upper thin walled section near the change in wall thickness at several BWRs. The cause of the cracking appears to be a combination of thermal cycling and intergranular stress corrosion cracking. The thermal cycling results from insertion and scram movements. [4.6-54]

If a collet housing were to fail completely at the reported crack location, the index tube would lock into position so that the control rod could not be inserted or withdrawn. The chance that a large number of collet housings would fail completely at about the same time is very remote. This is primarily true because the distributions of failures by cracking mechanisms such as stress corrosion and fatigue are not linear functions. That is, failure is a function of log time or log cycles. Distribution of failures of similar specimens generally follow a log normal pattern, with one to two orders of magnitude in time or cycles between failures of the first and failures of the last specimen.

4.6.4.5 Scram Discharge Volume Pipe Break

The probability of a scram discharge volume pipe rupture resulting in a loss of coolant accident is of such a small magnitude that the event is beyond the range of a credible occurrence. This is based on the results of two independent studies that have been performed on scram discharge volume piping systems, and the low probability that all scram outlet valves will fail simultaneously with the pipe break. [4.6-55]

The first of these studies was performed by GE and documented in report NEDO-24342.[7] In this generic evaluation report regarding BWR scram system pipe breaks, the probability of a scram discharge volume pipe break per challenge was calculated to be less than 5 x 10^{-6} per reactor year.

In addition to the GE report, a plant specific pipe fracture analysis was performed on the scram discharge volume system piping installed at LaSalle Station, Unit 1.[8] As a result of this study, the probability of an scram discharge volume pipe break was conservatively calculated to be 7 x 10^{-6} per reactor year. The SDV system piping at LaSalle Station is similar to the SDV system piping found at Quad Cities.

The SDV pipe break scenario can only result in a loss-of-coolant accident when the postulated pipe break occurs simultaneously with a failure of all scram outlet valves to close. Conservatively assuming that the probability of all scram outlet valves failing to close is 1 x $10^{\wedge-1\wedge}$ per reactor year, the combined events result in a probability of 10^{-7} per reactor year or less. This places the frequency of occurrence beyond the range which need to be taken into account in the design of a nuclear facility.

The results of the two independent studies, combined with the positive results of the hydrostatic tests performed on the scram discharge volume piping, and the low probability of all scram outlet valves failing to close, provide sufficient evidence that the probability of an SDV pipe break resulting in fuel failure is so minuscule that it does not merit further review.

4.6.4.6 Scram Failure Modes

Studies have been made to determine the potential for common mode failure of the CRD system. The postulated events, and the results of these studies are discussed in the following. The postulated common mode failures that would inhibit reactor scram are also addressed in NEDC-20634.[6] [4.6-56]

The CRD hydraulic system is so arranged that the equipment common to each drive can be packaged in modular form (Figure 4.6-9), one module for each drive. Any failure of the scram system within a particular module would, therefore, affect only its associated drive. Areas which are necessary to the scram system and common to all modules include the accumulator charging header, scram discharge headers and scram discharge instrument volumes. Each of the two banks of HCUs has its own scram discharge volume consisting of a scram discharge header which connects to an instrument volume. [4.6-57]

Failure of a HCU to effect a scram can be attributed to one of the following four potential faults: [4.6-58]

1. Electric,
2. Pneumatic,
3. Hydraulic, or
4. Mechanical damage combined with the above is possible.

Electrical failure of both scram solenoid pilot valve (SSPV) solenoids will cause them to "fail safe," thus causing a CRD scram. An electrical "failure" which would interfere with scram is an inadvertent energization of a scram valve bus, which could result from failure of scram contactors to open. However, since the RPS bus is divided into four sections, only 1/4 of the CRDs (distributed) are delayed by the short time required for backup scram valves to vent the scram air supply header. [4.6-59]

Pneumatic failure of both scram valves results in a CRD scram since these valves are fail safe. It is unlikely that these scram solenoid valves could "freeze up" because of contaminated air. The air used comes from the instrument air supply which is filtered before entering the scram air header. Also, this air is used by much more sensitive instruments than the scram solenoid valves, so detection of contaminated air would appear in other instruments prior to its detection in the scram solenoid valves. [4.6-60]

A "freezing up" of the inlet scram valve to open at low reactor pressure or the outlet scram valve to open at high reactor pressure will prevent a single rod scram. Common mode failures which could affect more than one rod are:

- Over-tightening of the scram valve stem packing of the drives is prevented by performing valve maintenance, including packing adjustments, under control processes. The packing nuts are torqued to a specified value. The valve is then stroked to verify proper operation.

- Severe distortion of the HCU's by external forces. This fault is precluded since the HCU's are designed and installed to operate with maximum earthquake loadings.

The only common link between accumulators is the charging water supply line. Failure of the water supply does not result in a common mode failure since each accumulator will retain its charge by the action of its individual charging water check valve. The check valve closes holding the accumulator pressure for a limited time. In addition, each HCU has a main control room alarm which will annunciate low pressure and a main control room panel light which will identify the affected HCU. The charging water header also has a low charging water pressure indicator and alarm in the main control room. [4.6-61]

Gross contamination of the water supply could cause the accumulator pistons to stick; this is not considered possible as the water is the same water used in the reactor primary system. Furthermore, this water has been subject to further filtering in the CRD system, and these filters are monitored for indication of contamination by sensing high differential pressure across the filter. Abnormal conditions are annunciated by an alarm. [4.6-62]

There are two discharge instrument volumes which receive water ejected from the drives during scram. Each is equipped with six level switches; two in each channel of the RPS, plus one that provides a rod withdrawal block, and one with a high level alarm if the water level rises to a preset value. This arrangement assures that adequate volume is available in the system to receive the scram water. Diversity among sensor types also reduces the potential for common mode failures. [4.6-63]

The only common point in the system where an accident, such as a plugged line, could affect the scram time of more than one drive would be in the scram discharge header. As this header is much larger than the individual lines feeding into it, it is extremely unlikely that this line could become plugged. Further, action of the drive during a scram is such that it will develop a pressure in excess of 2000 psig if its discharge is restricted. This pressure should be capable of expelling any conceivable line restrictions. The system is designed to accommodate such pressures. [4.6-64]

Also, because of the unique design of the locking piston drive, an automatic scram occurs if both drive lines or only the outlet line is severed at any point with the reactor at pressure.

In conclusion, no known single failure, even all identified potential common mode failures, can render a significant portion of the CRD system inoperable. [4.6-65]

4.6.5 Testing and Verification of the Control Rod Drive System

4.6.5.1 Control Rods and Control Rod Drives

Testing and inspecting of the control rod velocity limiter are not required following installation on the control rod assembly. In addition to close surveillance during the fabrication of the rod velocity limiter and control rod assembly manufacture, random control rod assemblies were shop tested which included rod drop tests. Each velocity limiter was visually inspected and gauged prior to assembly. Preoperational tests confirm the operation of the individual control rod assemblies for normal operation and scram conditions. [4.6-66]

After installation, all rods and drive mechanisms undergo a full-stroke test. In those instances where work on the CRD System or a control rod could affect coupling or scram time, the Technical Specifications identify the activities necessary prior to declaring the control rod operable. [4.6-67]

During reactor operation individual CRD mechanisms can be actuated to demonstrate functional performance. Each time a control rod is withdrawn, the operator observes the incore instrumentation to verify that the control rod is following the drive mechanism. All control rods that are partially withdrawn from the core are tested at least once per 31 days by inserting the rod one notch and returning it to the original position.

When the operator fully withdraws a control rod out of the core during power operation, the mechanical coupling integrity is tested by trying to withdraw the rod drive mechanism to the overtravel position. Failure of the drive to overtravel demonstrates rod-to-drive coupling integrity.

The surveillance requirements for the CRD system are described in the Technical Specifications. [4.6-68]

4.6.5.2 Control Rod Drive Housing Supports

The CRD housing support will be in place any time the reactor is to be operated. Sections may be removed during outages to permit maintenance on control rod drives. Any time maintenance or other work on the system has been performed the support structure will be inspected to assure proper installation before the reactor is returned to operation. [4.6-69]

4.6.6 References

1. "Safety Evaluation of the General Electric Advanced Longer Life Control Rod Assembly," NEDE-22290-A, Supplement 2, January 1985.

2. "Safety Evaluation of the General Electric Duralife 230 Control Rod Assembly," NEDE-22290-A, Supplement 3, May 1988.

3. "Topical Report ASEA-ATOM BWR Control Blades for US BWRs," TR UR 85-225A, October 1985.

4. "GE BWR Control Rod Lifetime," NEDE-30931-2-P, Revision 2, May, 1988.

5. "ABB-ATOM Control Rods for BWR 2/3/4/5/6 Service Life Limits Recommendations," UR 87-102, Revision 1, April 15, 1987.

6. "Reactor Protection System Common Mode Failure Analysis." NEDC-20634,

7. "Fracture Mechanics Analysis of Scram Piping Reliability," SAI - San Jose, D.O. Harris, January 11, 1982.

8. "GE Evaluation in Response to NRC Request Regarding BWR Scram System Pipe Breaks," NEDO-24342, April 1981.

9. APED-5446, February 1967, Control Rod Velocity Limiter.

10. Determination of Boron Depletion Limits for Use by the Process Computer in Determining Control Blade Lifetime, BSSCN:95-007, January 29, 1996.

11. WCAP-16182-P-A, March 2005, Westinghouse BWR Control Rod CR 99 Licensing Report.

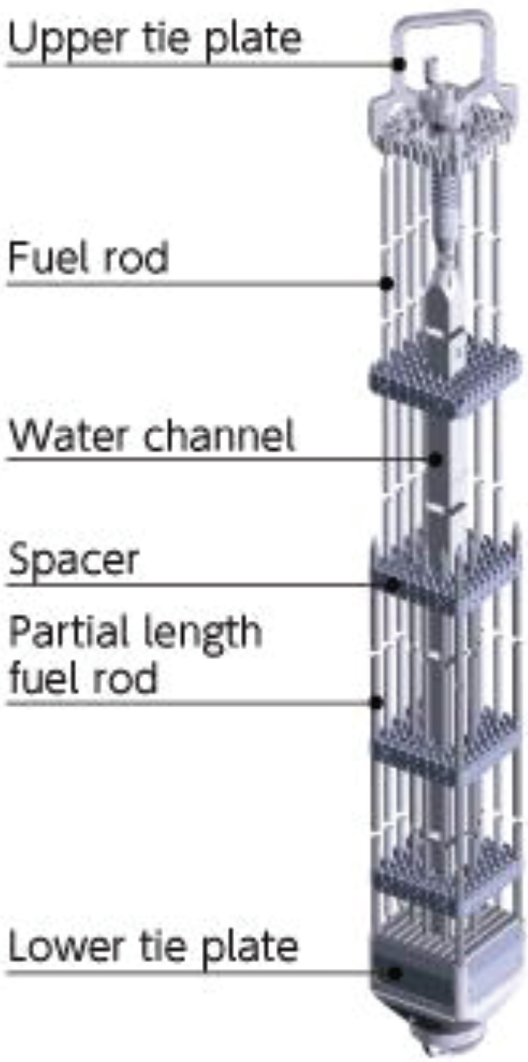
12. NEDE-33284 Supplement 1P-A, Revision 1, March 2012, Marathon-Ultra Control Rod Assembly Licensing Topical Report.

Table 4.6-1

DESIGN STRESS INTENSITY LIMITS
FOR ORIGINAL EQUIPMENT CONTROL ROD ABSORBER TUBES

Categories	Limits: Yield Strength (y)	Limits: Ultimate Strength (u)
General Primary Membrane	2/3 S_y	1/2 S_u
Local Primary Membrane	S_y	3/4 S_u
Primary Membrane plus Bending	S_y	3/4 S_u
Primary plus Secondary	2 S_y	1.5 S_u

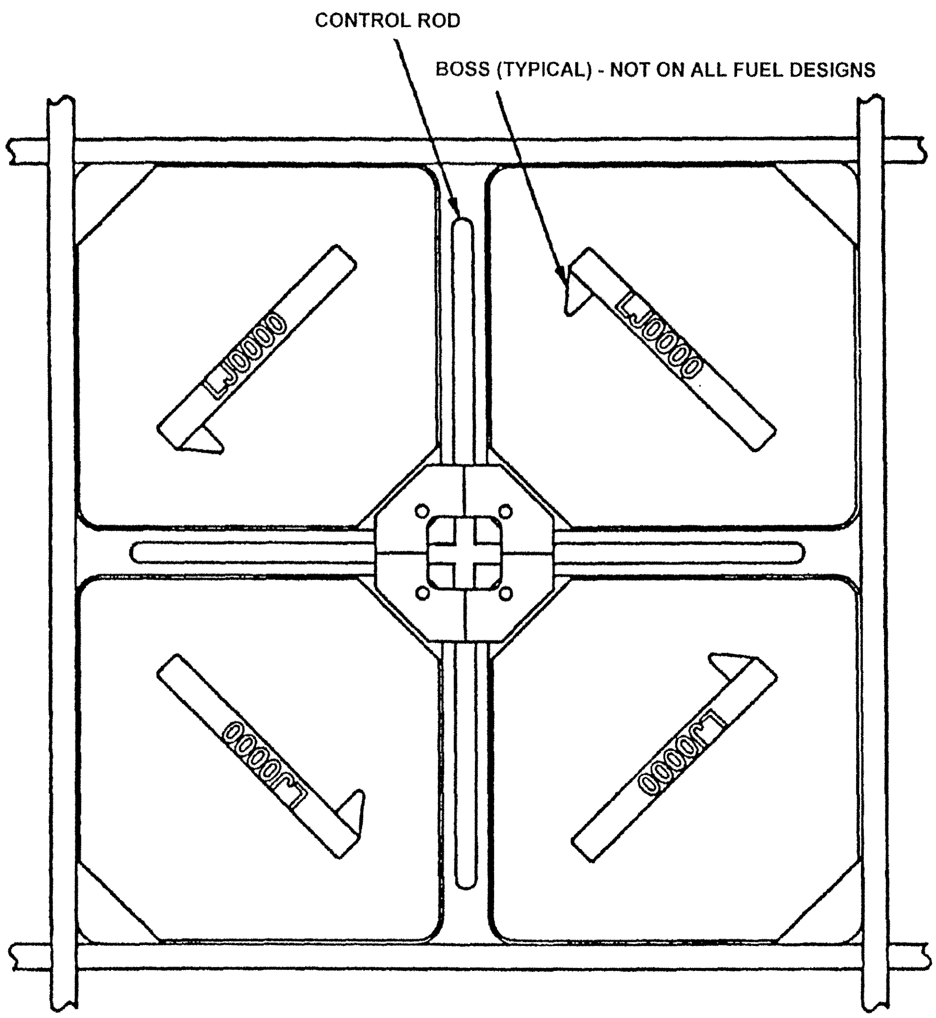
Figures 4.2-1 and 4.2-2 are deleted.



Quad Cities Station
Unit 1

Atrium 10XM Fuel Assembly

Figure 4.2.3
Revision 14, October 2017

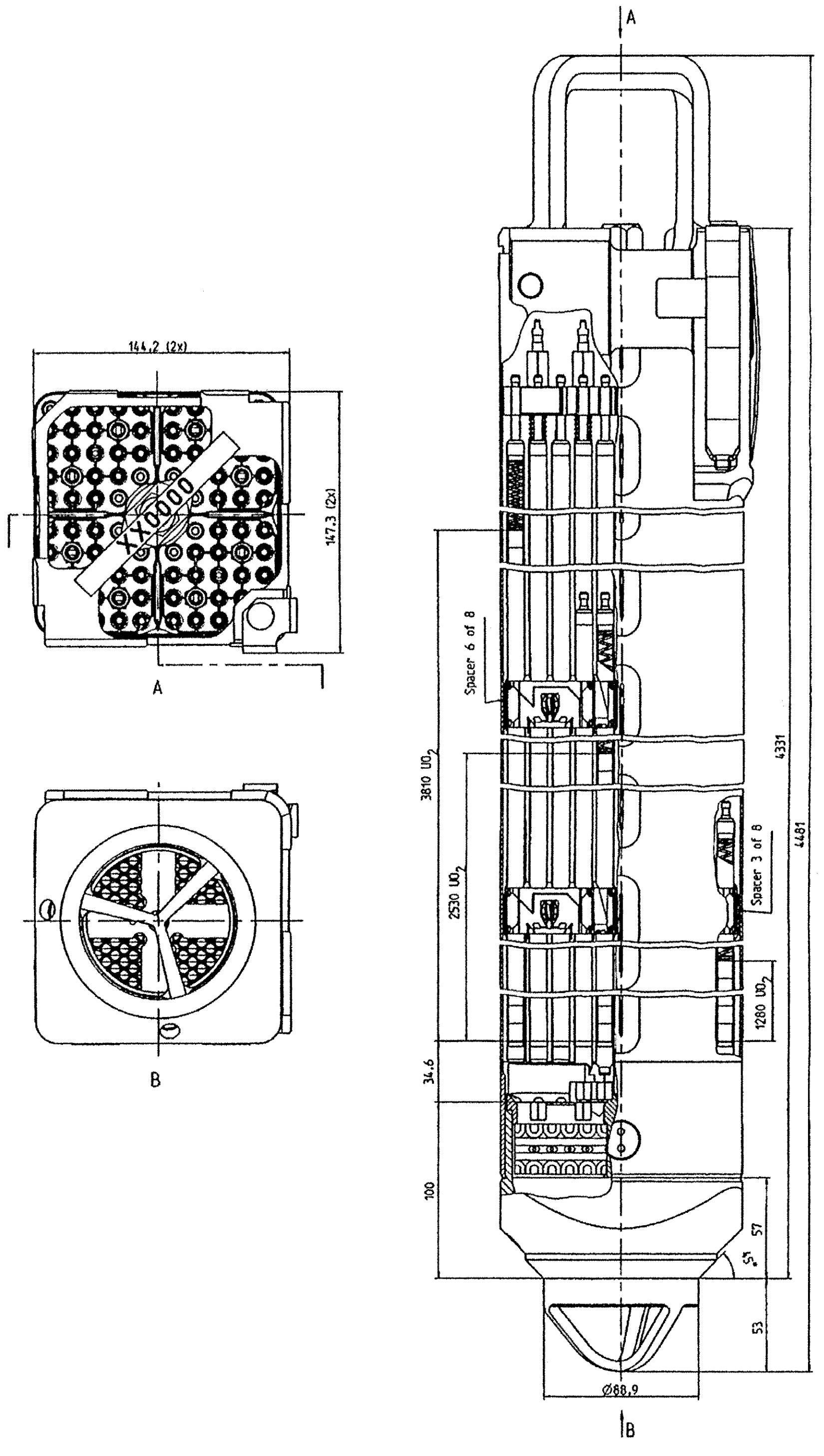


QUAD CITIES STATION
UNITS 1 & 2

TYPICAL CORE CELL TOP VIEW

FIGURE 4.2-4

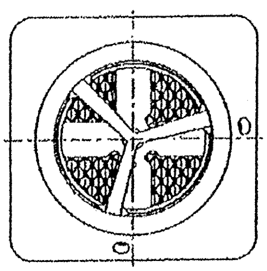
REVISION 9, OCTOBER 2007

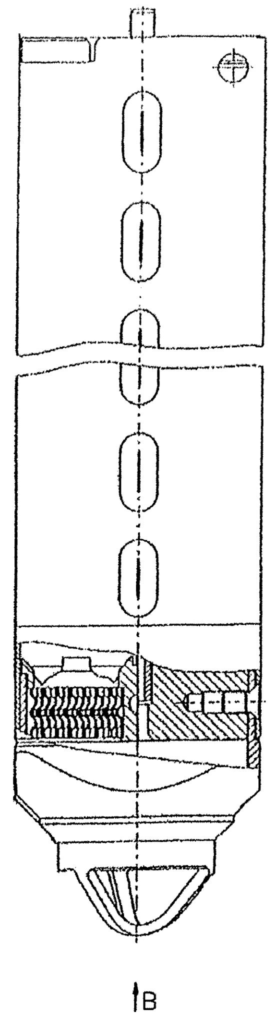


QUAD CITIES STATION
UNIT 2

SVEA-96 OPTIMA2 FUEL ASSEMBLY

FIGURE 4.2-5
REVISION 9, OCTOBER 2007

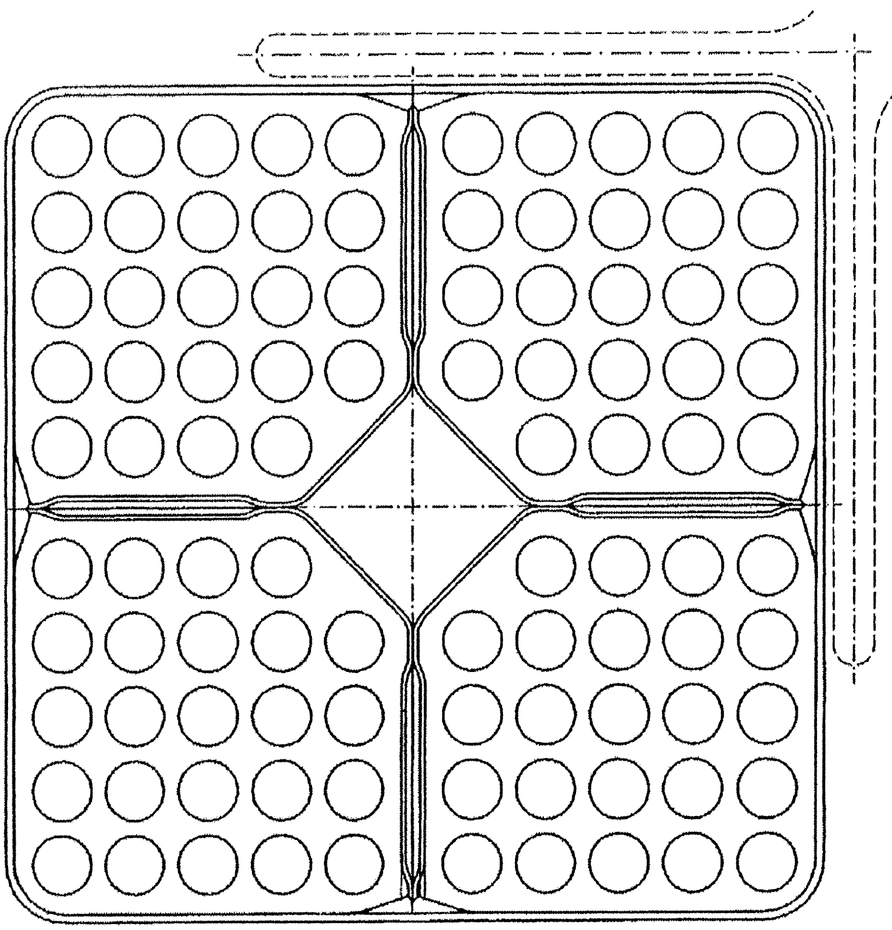




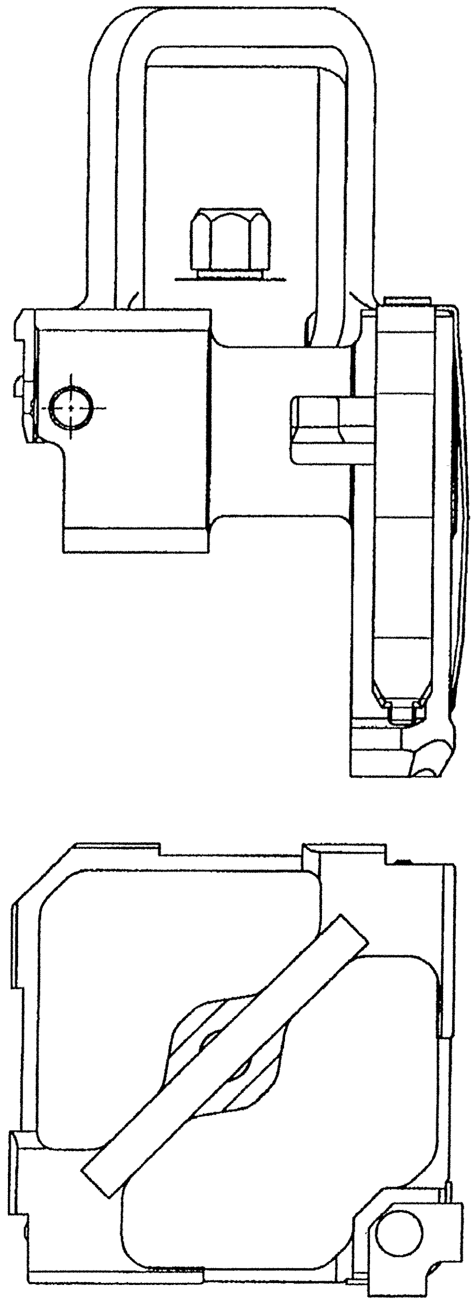
QUAD CITIES STATION
UNIT 2

SVEA-96 OPTIMA2 FUEL CHANNEL
(FULL VIEW)

FIGURE 4.2-6a
REVISION 9, OCTOBER 2007



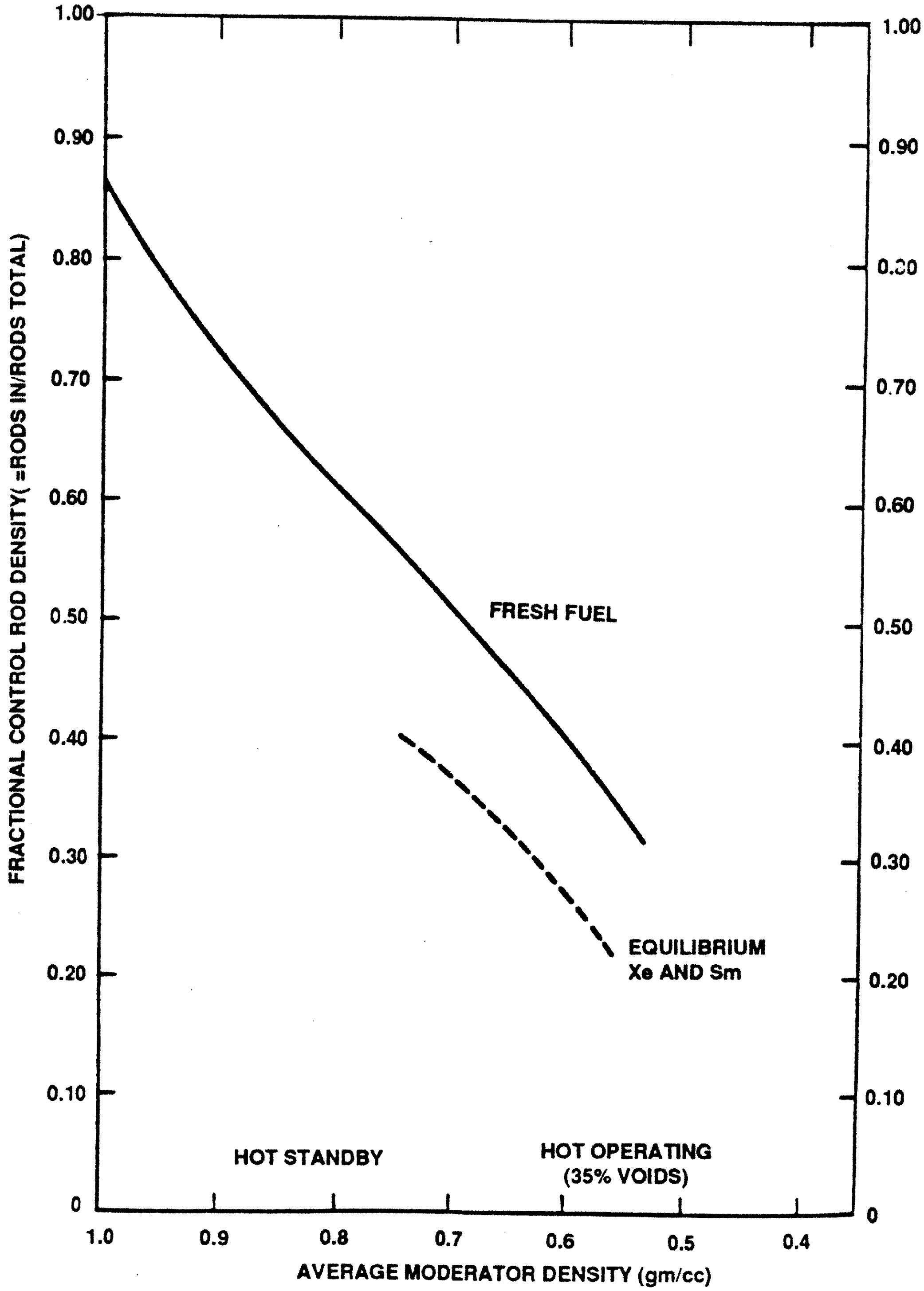
QUAD CITIES STATION UNIT 2
SVEA-96 OPTIMA2 FUEL CHANNEL (CROSS-SECTION)
FIGURE 4.2-6b REVISION 9, OCTOBER 2007



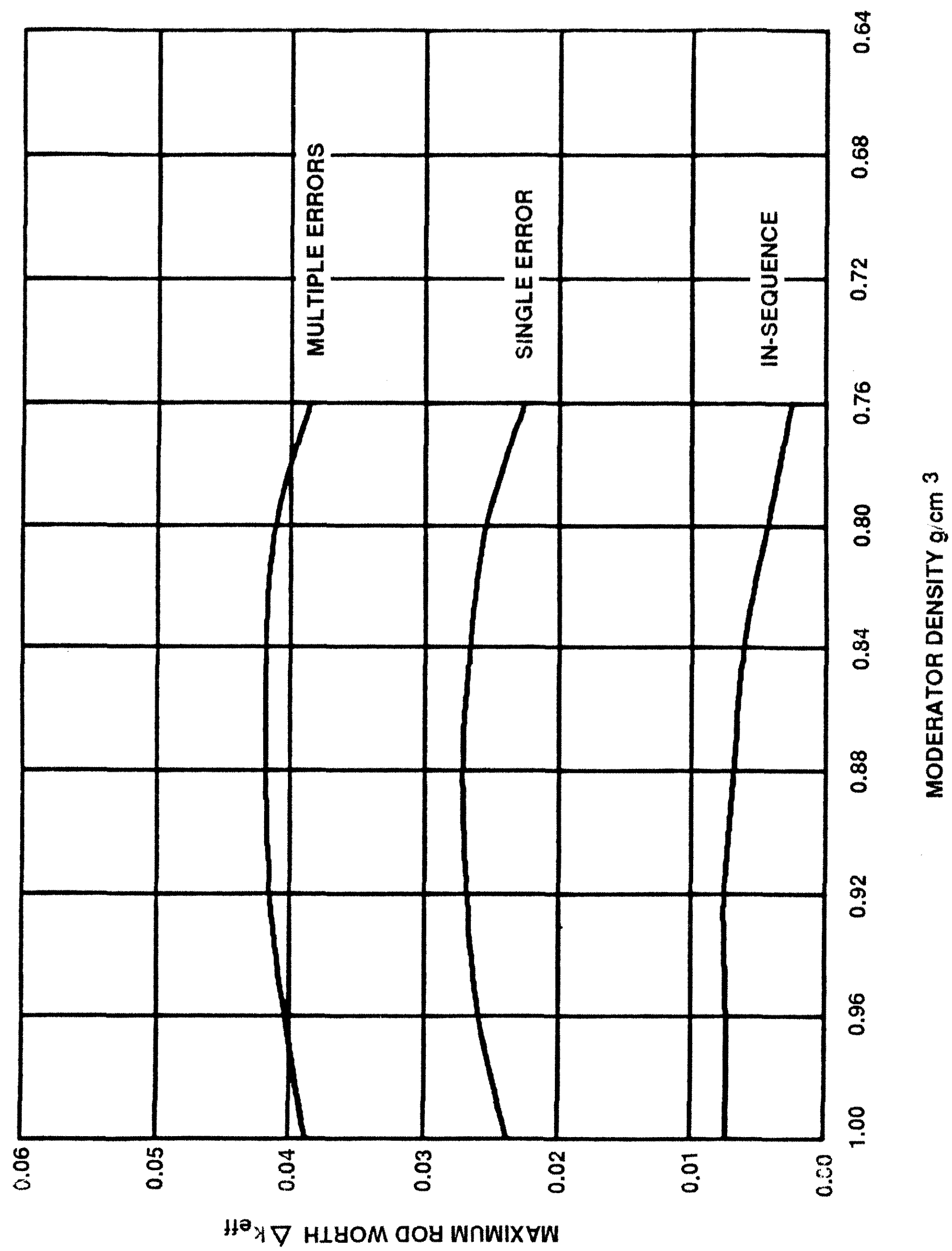
QUAD CITIES STATION
UNIT 2

SVEA-96 OPTIMA2 FUEL HANDLE

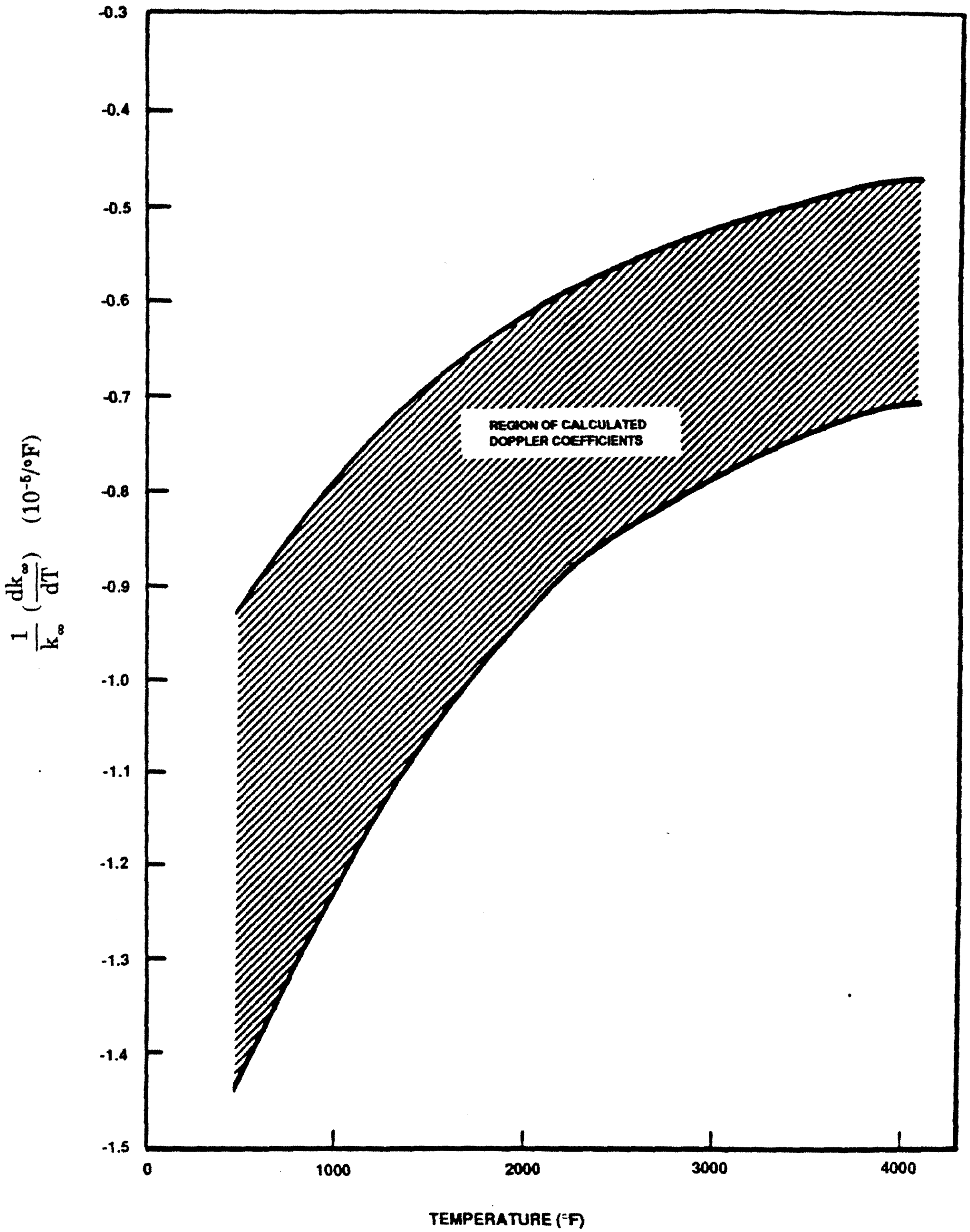
FIGURE 4.2-7
REVISION 9, OCTOBER 2007



QUAD CITIES STATION UNITS 1 & 2
(TYPICAL) FRACTIONAL CONTROL ROD DENSITY VS. AVERAGE MODERATOR DENSITY
FIGURE 4.3-1
REVISION 4, APRIL 1997



QUAD CITIES STATION UNITS 1 & 2
MAXIMUM ROD WORTH VS. AVERAGE MODERATOR DENSITY (REFERENCE NEDO - 10527)
FIGURE 4.3-2



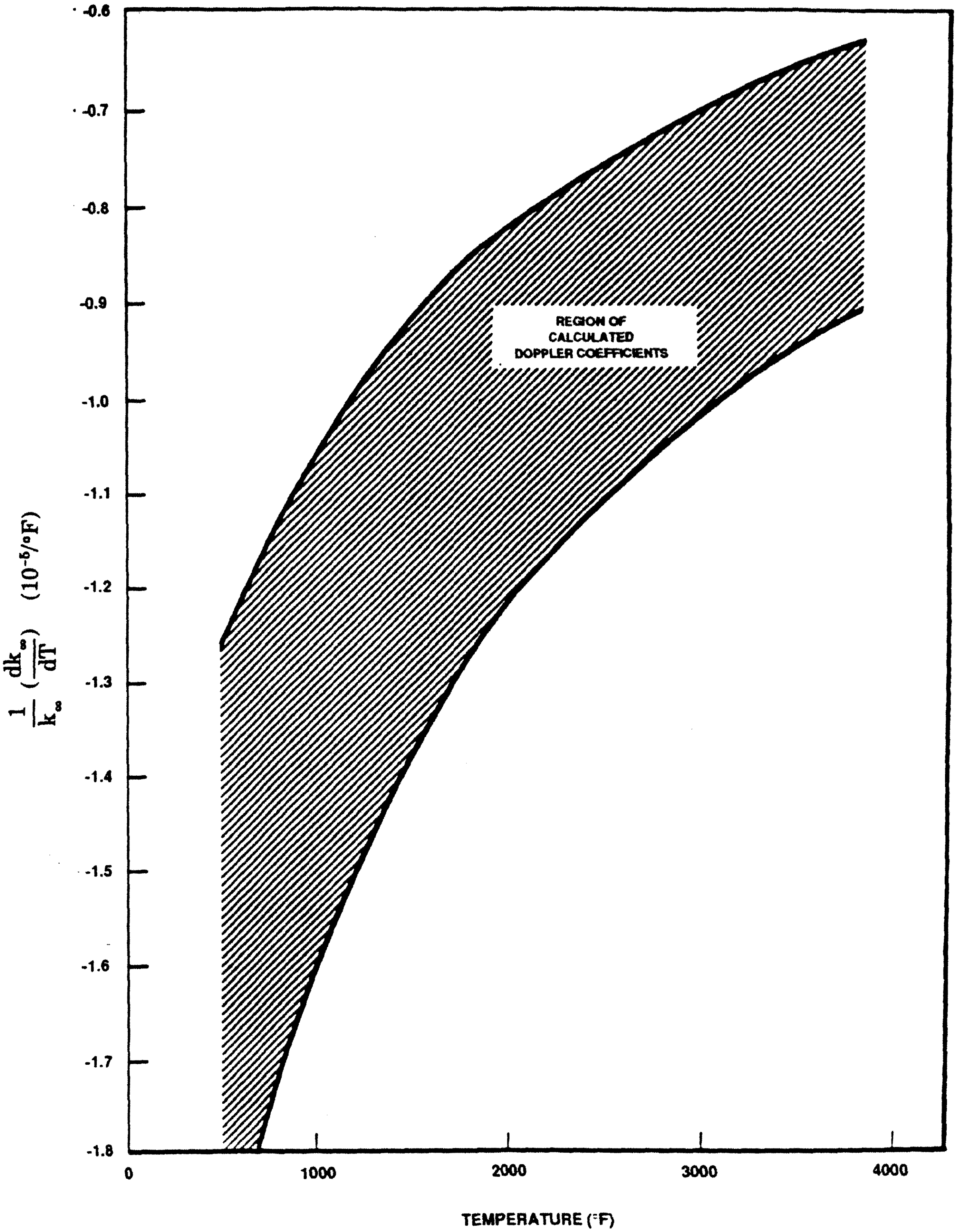
QUAD CITIES STATION

UNITS 1 & 2

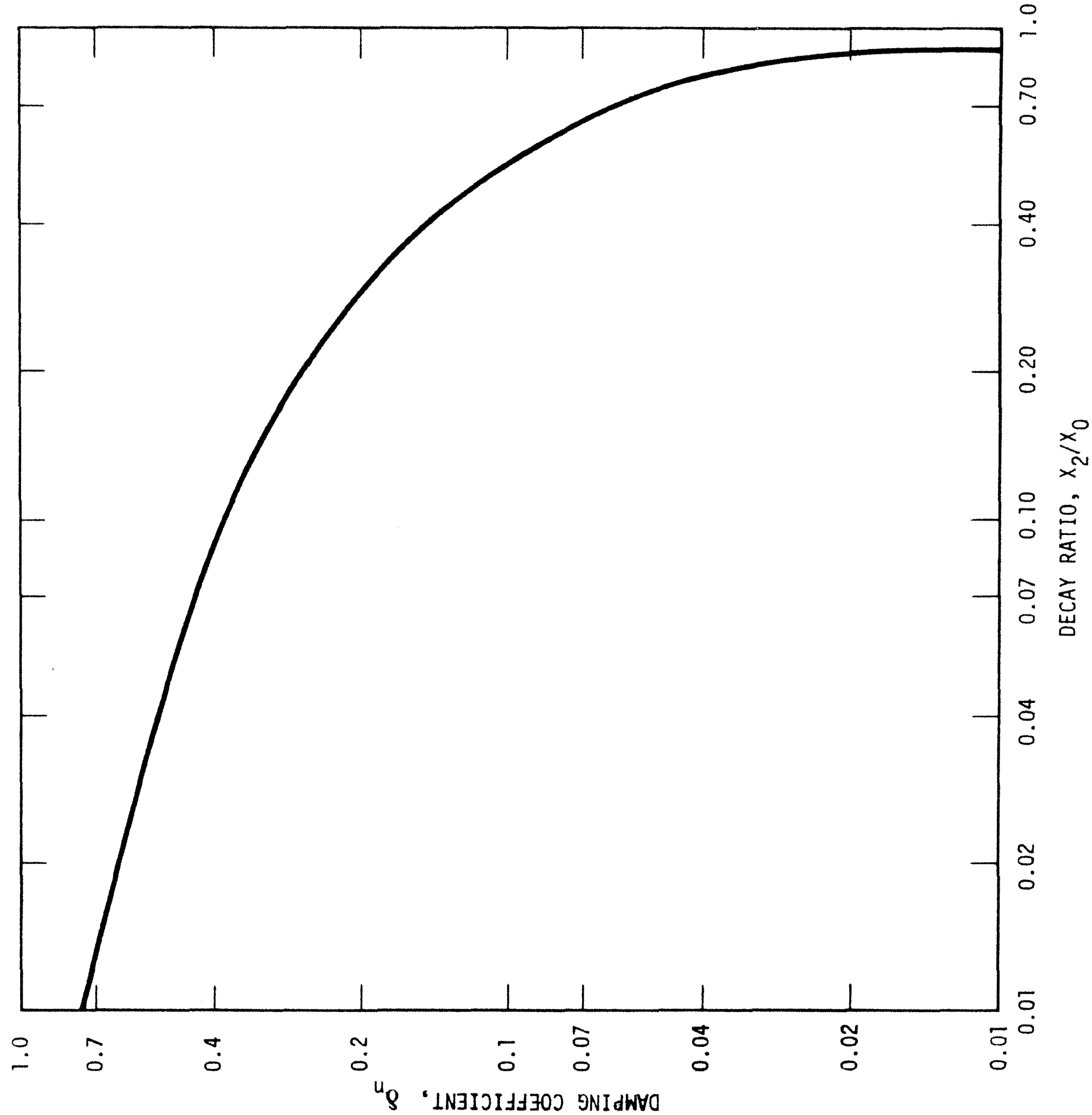
(TYPICAL) ENVELOPE OF DOPPLER COEFFICIENT VS. TEMPERATURE, E=200MWd/t

FIGURE 4.3-3

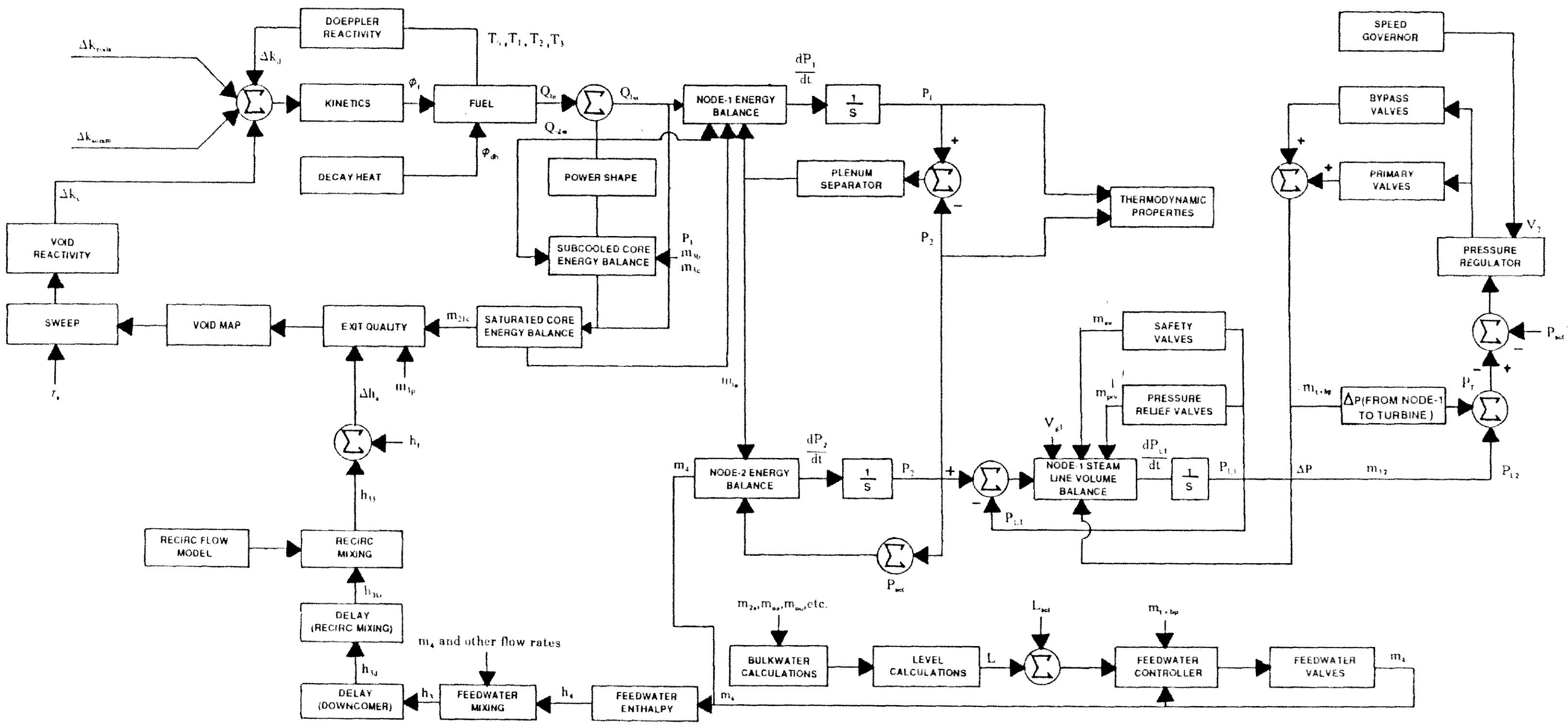
REVISION 4, APRIL 1997



QUAD CITIES STATION UNITS 1 & 2
(TYPICAL) ENVELOPE OF DOPPLER COEFFICIENT VS. TEMPERATURE, E=15,000MWd/t
FIGURE 4.3-4 REVISION 4, APRIL 1997



QUAD CITIES STATION UNITS 1 & 2
DAMPING COEFFICIENT VS. DECAY RATIO
FIGURE 4.3-5



h_f = Saturated fluid enthalpy
h_3 = Enthalpy of downcommer fluid
h_{3c} = Enthalpy of fluid in recirculation loop
h_4 = Feedwater enthalpy
h_{33} = Core inlet enthalpy
h_{34} = Delayed enthalpy — enthalpy of fluid at entrance to recirculation loops
L = Reactor vessel water level
L_{set} = Level setpoint
m_4 = Feedwater mass flow rate
m_{1s} = Core exit — steam mass flow rate
m_{2s} = Separator saturated water mass flow rate
m_{sc} = Mass flow rate from subcooled region of the core
m_{3c} = Core mass flow rate
m_{3p} = Core exit mass flow rate
m_{21s} = Steaming rate
m_{12} = Steaming flow from Node 1 to Node 2 of the steam line
m_{t+bp} = Steam flow to the turbine and steam line bypass

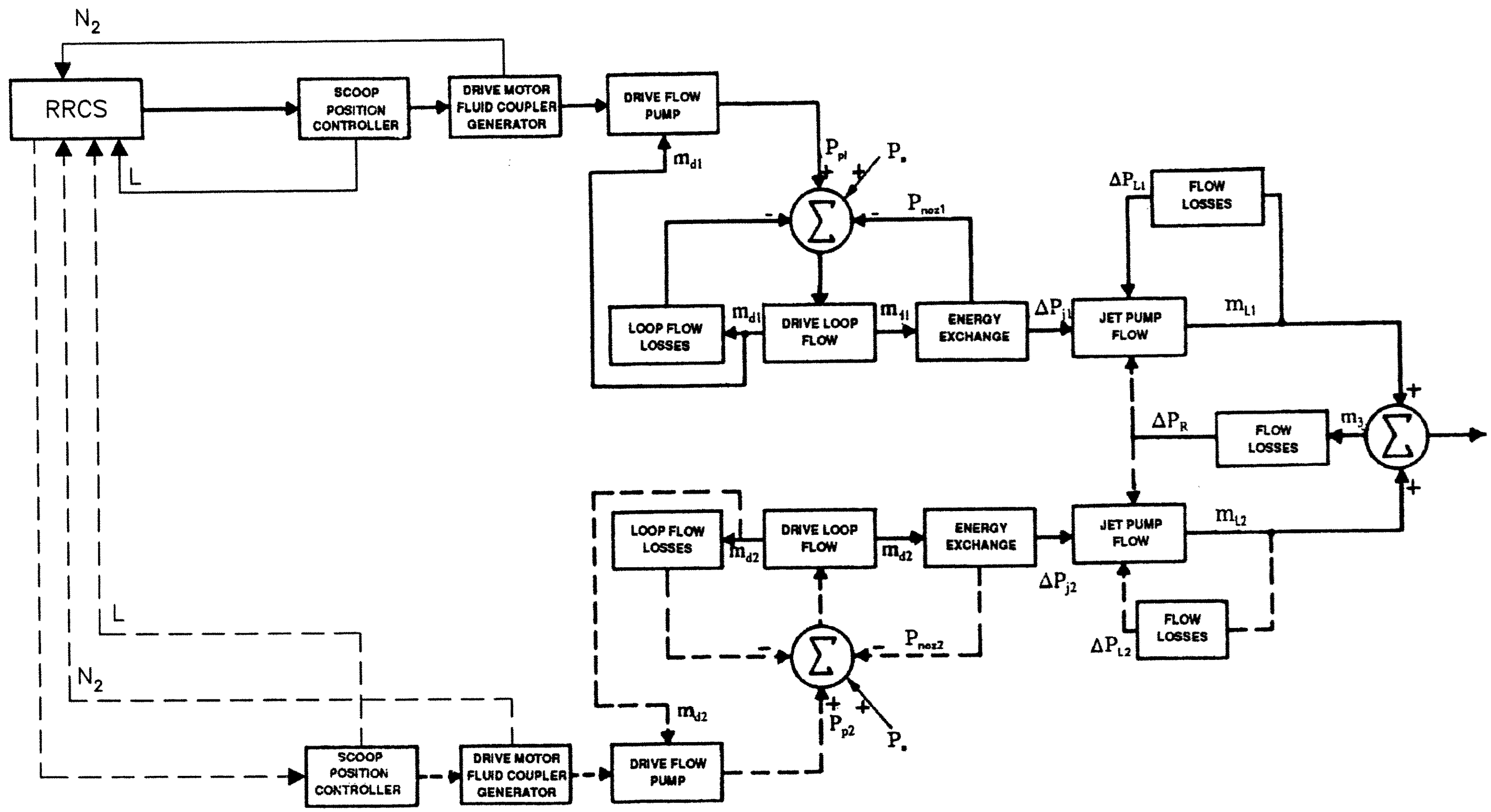
m_{ce} = Saturated vapor mass flow into bulk water (carryunder)
m_{cu} = Saturated vapor mass flow out of Node 3 (dome)
m_{prv} = Pressure relief valve mass flow rate
m_{sv} = Safety valve mass flow rate
P_1 = Pressure — Node 1 (core)
P_2 = Pressure — Node 2 (upper plenum)
P_{L1} = Pressure — Steam line Node 1
P_{L2} = Pressure — Steam (in Node 2)
P_T = Turbine pressure
P_{set} = Pressure setpoint
Q_{ba} = Boiling active heat (heat to saturated region of the core)
Q_{nba} = Nonboiling active heat (heat to subcooled region of the core)
Q_{ta} = Total active core heat
T_0 = Fuel temperature — Center of fuel rod
T_1 = Fuel temperature — Radius r_1 of fuel rod
T_2 = Fuel temperature — Raduis r_2 of fuel rod

T_3 = Fuel temperature — Radius r_3 of fuel rod
V_2 = Speed governor demand signal
V_{g1} = Specific volume — Node 1 steam line
Δp = Pressure difference from Node 1 to turbine
Δh_s = Reactor subcooling
Δk_d = Doppler reactivity
Δk_{rods} = Control rod reactivity
Δk_v = Void reactivity
τ_s = Sweep time constant
φ_{dh} = Decay heat
φ_f = Neutron flux which produces heat in the fluid

QUAD CITIES STATION
UNITS 1 & 2

SINGLE CYCLE, FORCED CIRCULATION BWR

FIGURE 4.3-8



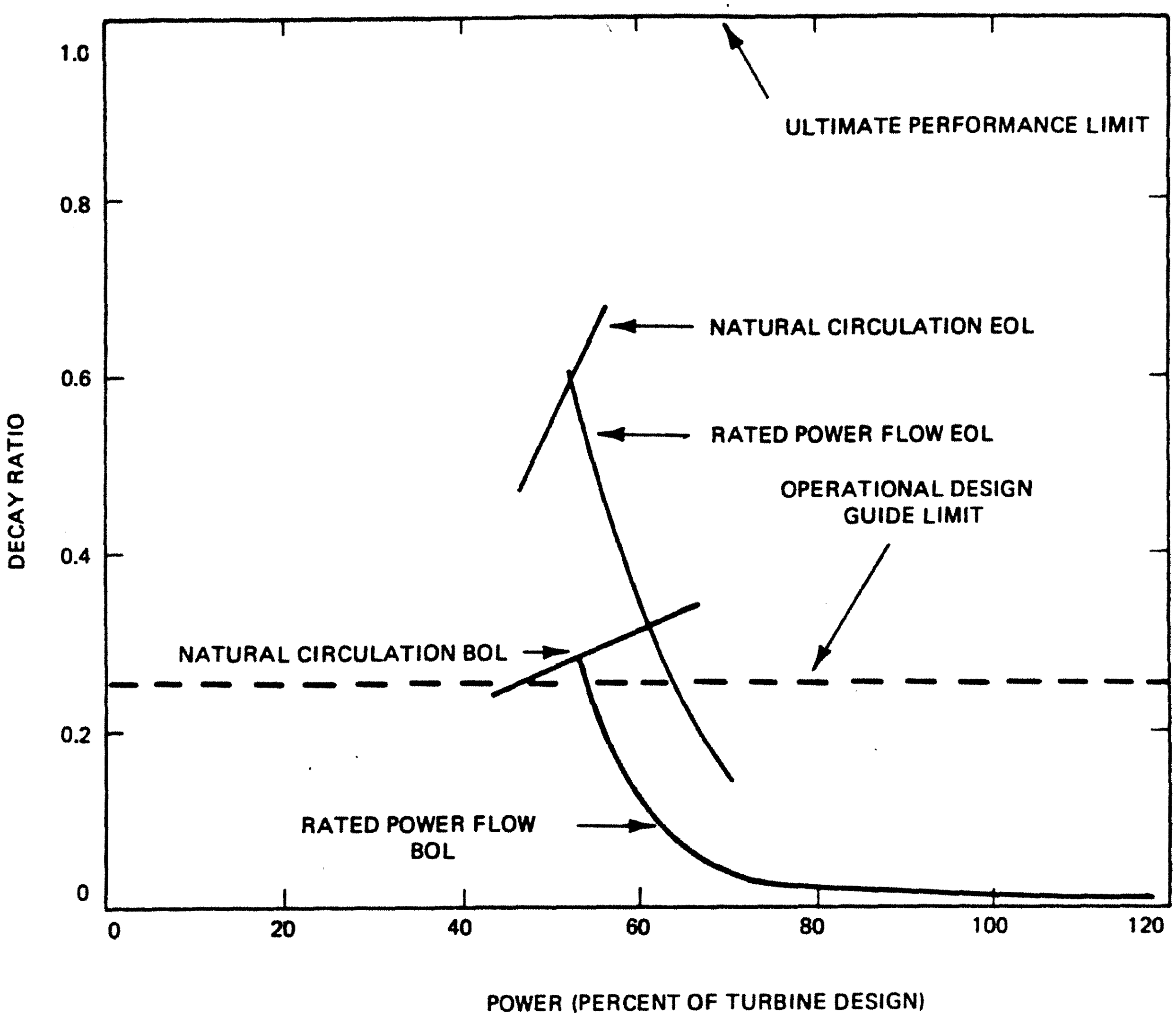
L = Scoop Tube Position (for misposition lockup)
m_{L1} = Jet diffuser flow — Loop 1
m_{L2} = Jet diffuser flow — Loop 2
m_3 = Recirculation mass flow rate
m_{d1} = Mass flow rate — jet pump driveline — Loop 1
m_{d2} = Mass flow rate — jet pump driveline — Loop 2
N_2 = Flow control generator speed
P_s = Jet pump suction pressure
P_{p1} = Drive pump pressure — Loop 1
P_{p2} = Drive pump pressure — Loop 2
ΔP_{L1} = External loop friction and local pressure losses — Loop 1
ΔP_{L2} = External loop friction and local pressure losses — Loop 2
ΔP_R = Core friction, local, and elevation pressure losses
P_{noz1} = Jet pump nozzle pressure — Loop 1
P_{noz2} = Jet pump nozzle pressure — Loop 2
ΔP_{j1} = Jet pump pressure head — Loop
ΔP_{j2} = Jet pump pressure head — Loop 2

RRCS = Reactor Recirculation Control System

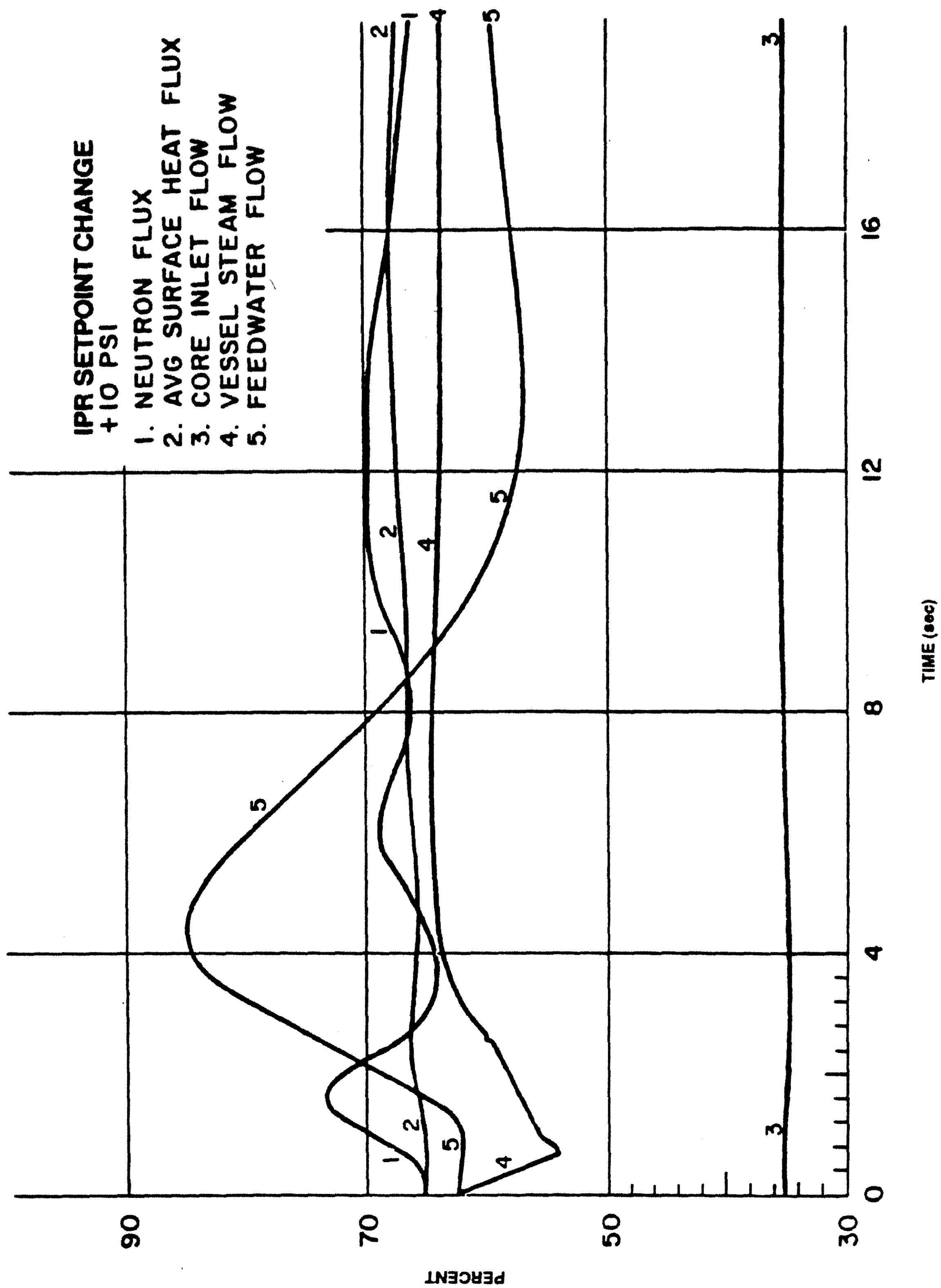
QUAD CITIES STATION
UNITS 1 & 2

RECIRCULATION FLOW MODEL – JET PUMPS

FIGURE 4.3–9
REVISION 8, OCTOBER 2005



QUAD CITIES STATION UNITS 1 & 2
(INITIAL CORE) DECAY RATIO VS. POWER
FIGURE 4.3-10 REVISION 4, APRIL 1997

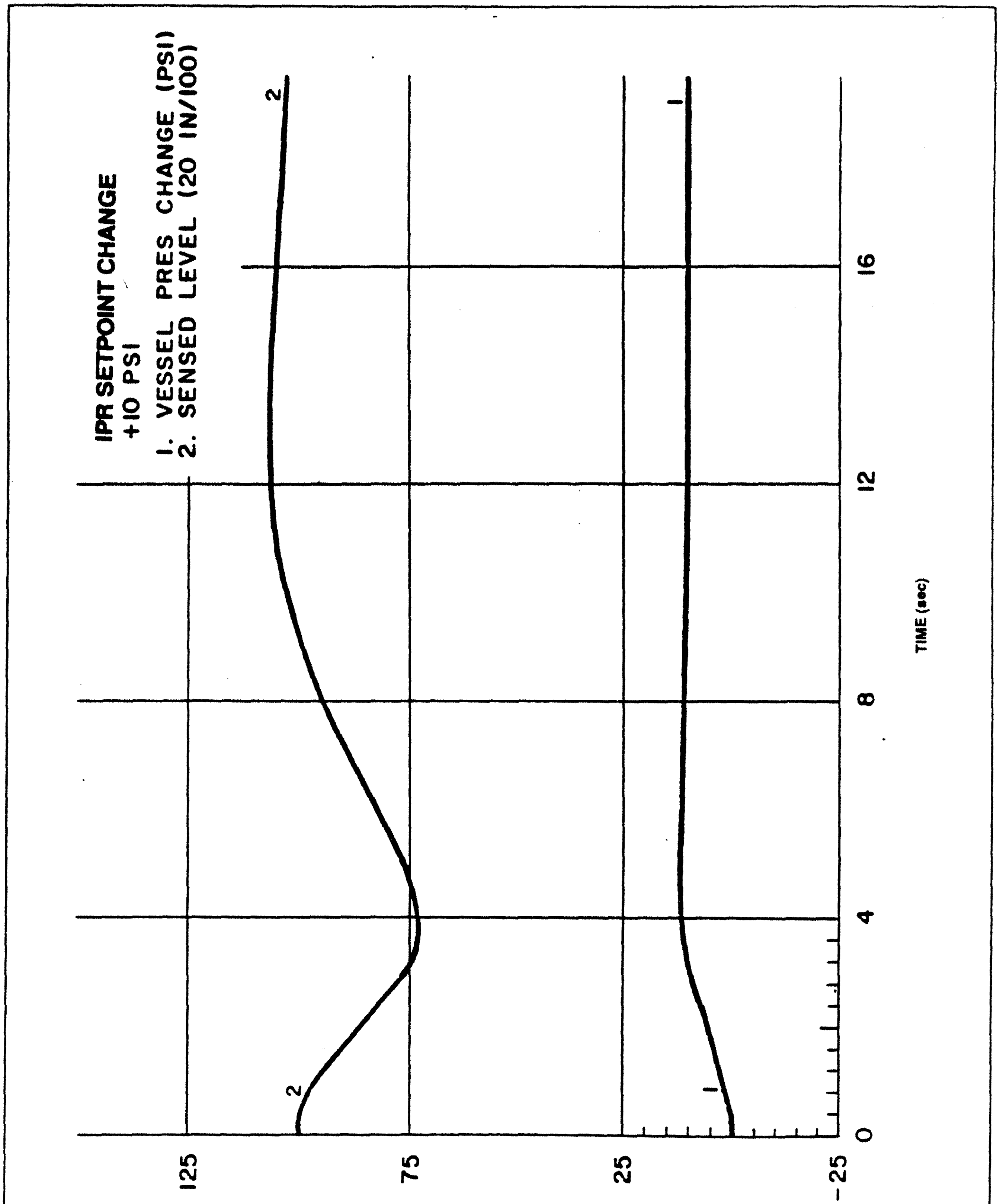


QUAD CITIES STATION
UNITS 1 & 2

10 PSI SETPOINT CHANGE
TRANSIENT ANALYSIS - 1642 MWt
(Results are based on initial core.)

FIGURE 4.3-11

REVISION 4, APRIL 1997

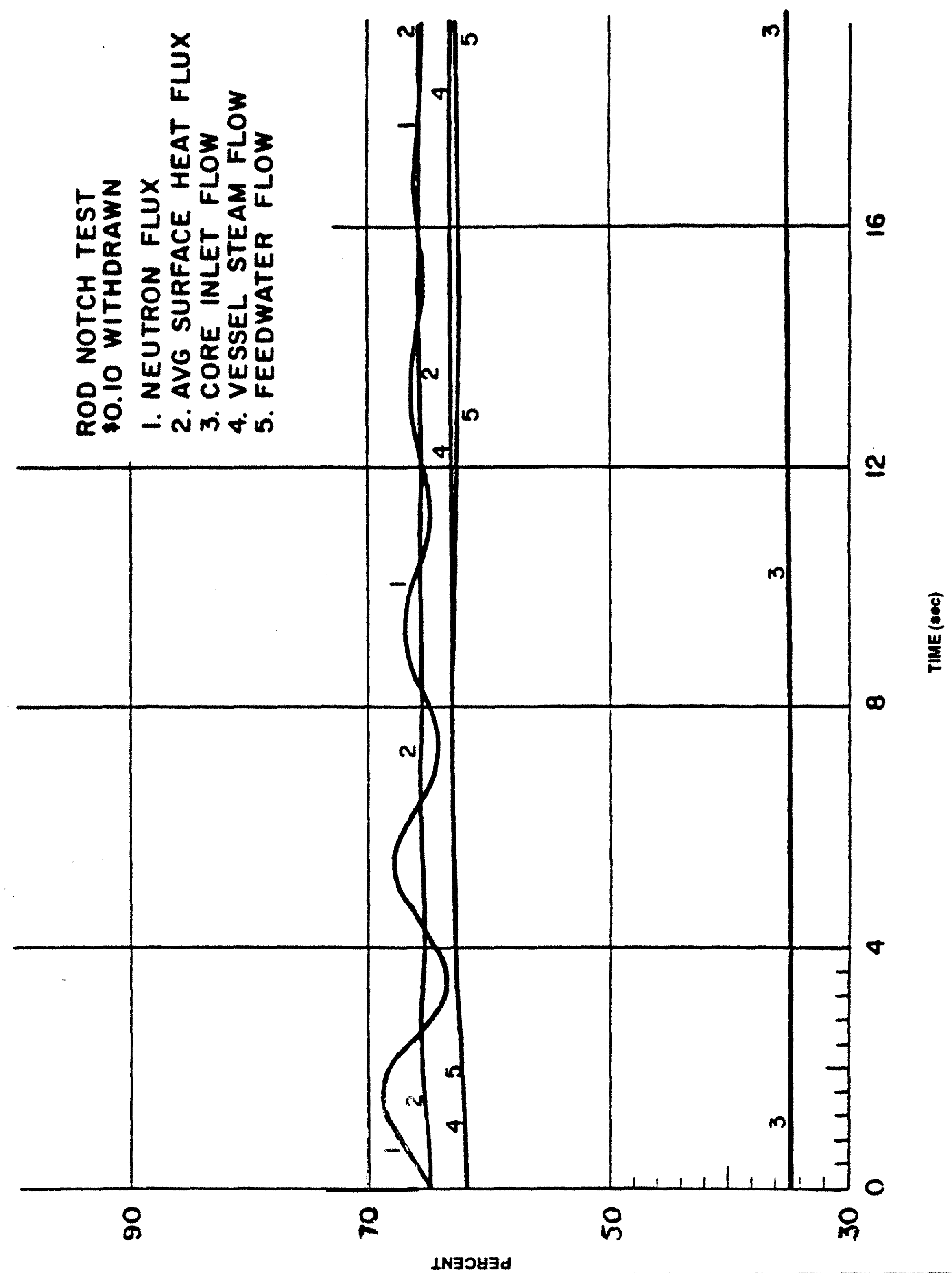


QUAD CITIES STATION
UNITS 1 & 2

10 PSI SETPOINT CHANGE
TRANSIENT ANALYSIS - 1642 MWt
(Results are based on initial core.)

FIGURE 4.3-12

REVISION 4, APRIL 1997

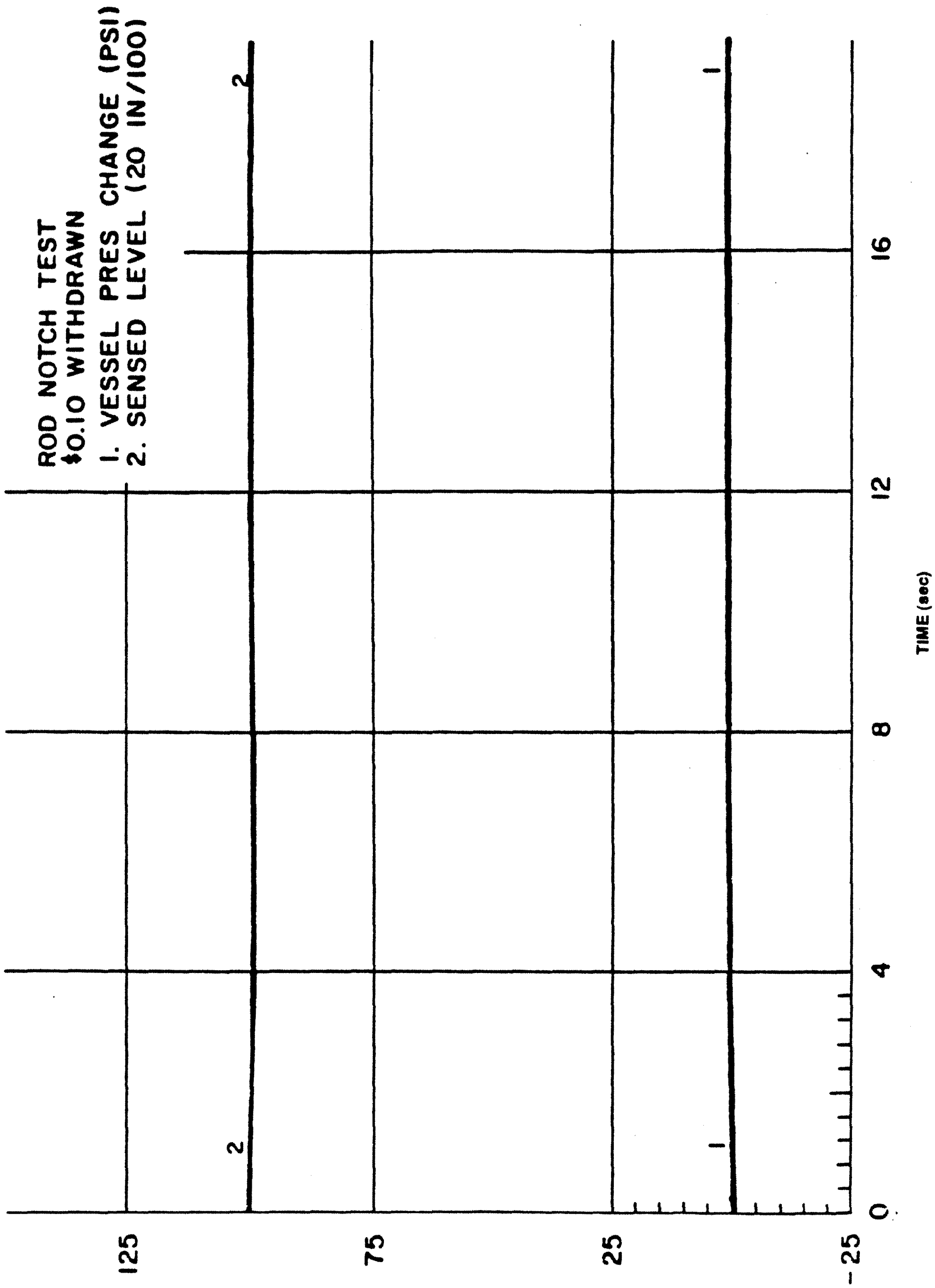


QUAD CITIES STATION
UNITS 1 & 2

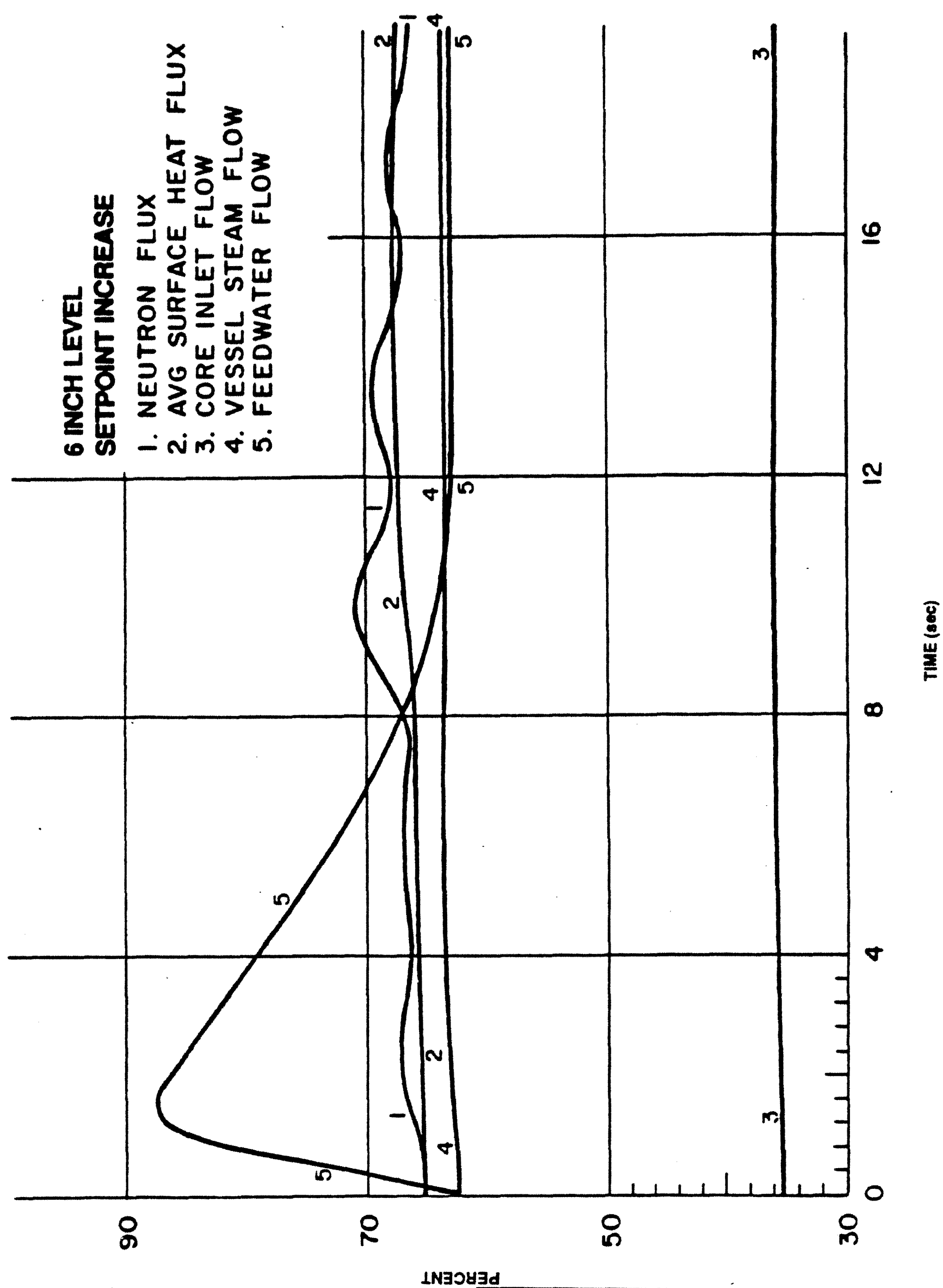
$0.10 ROD NOTCH REACTIVITY CHANGE
TRANSIENT ANALYSIS - 1642 MWt
(Results are based on initial core.)

FIGURE 4.3-13

REVISION 4, APRIL 1997



QUAD CITIES STATION UNITS 1 & 2
$0.10 ROD NOTCH REACTIVITY CHANGE TRANSIENT ANALYSIS - 1642 MWt (Results are based on initial core.)
FIGURE 4.3-14 REVISION 4, APRIL 1997

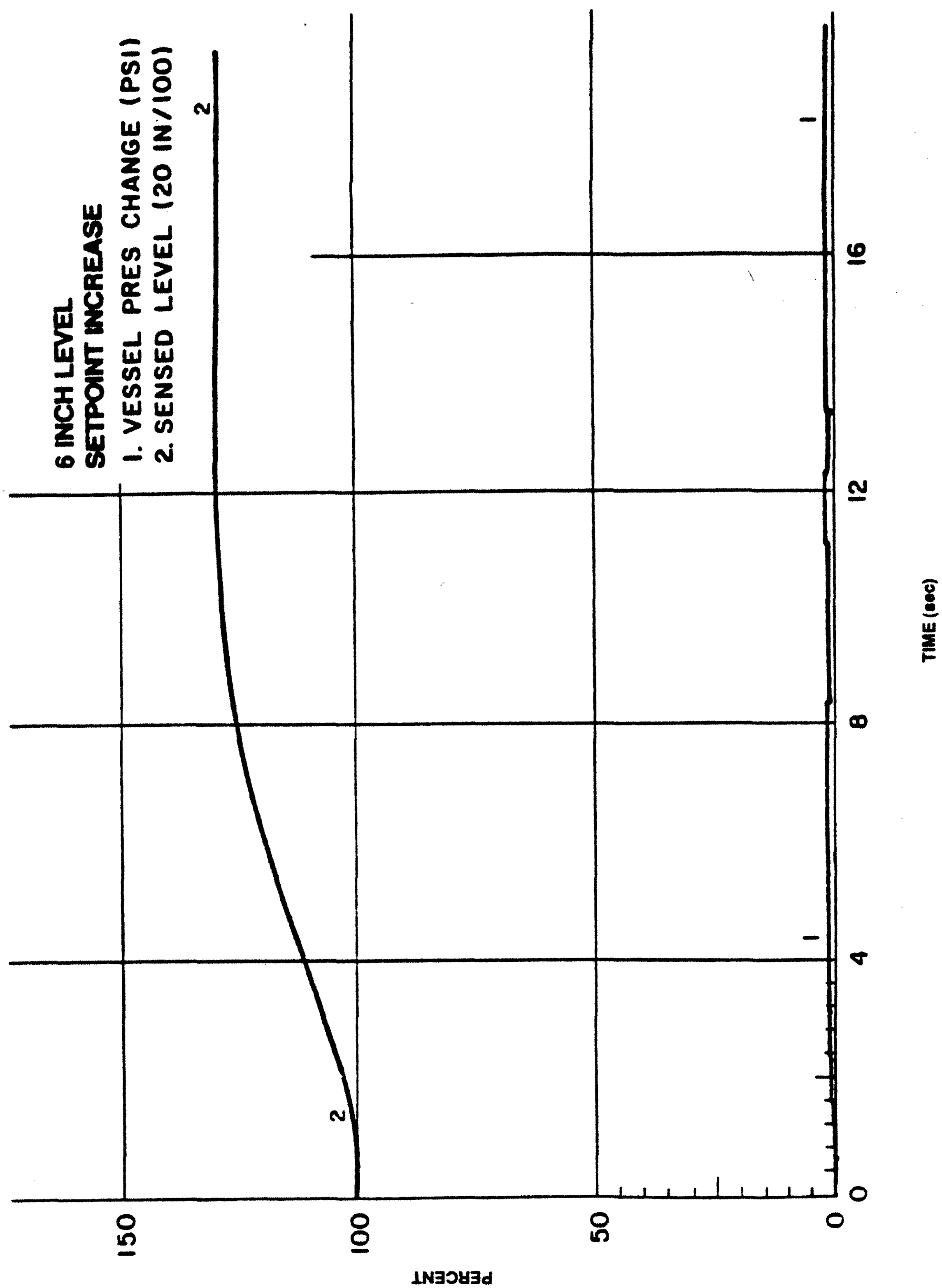


QUAD CITIES STATION
UNITS 1 & 2

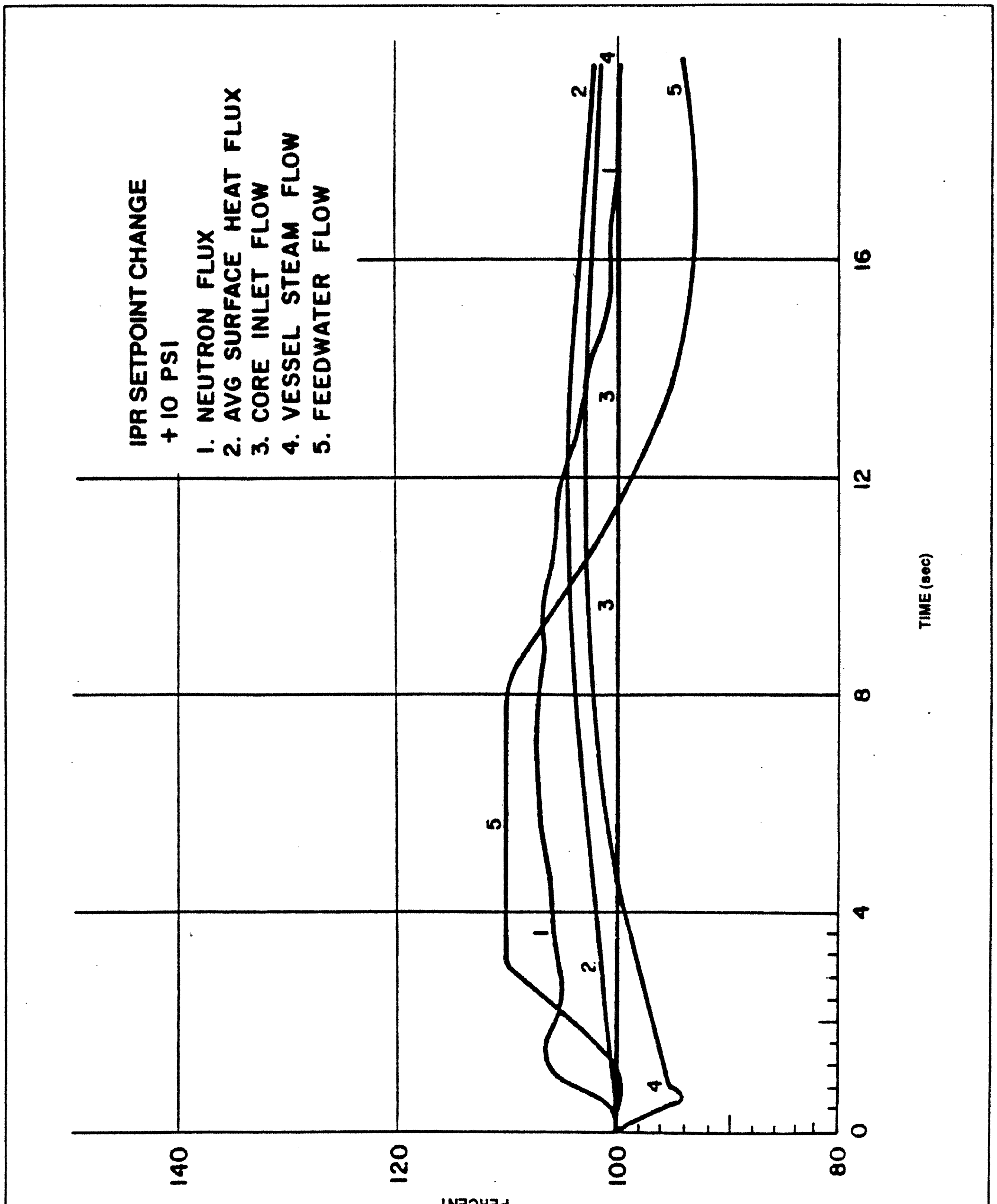
6-INCH WATER LEVEL SETPOINT CHANGE
TRANSIENT ANALYSIS - 1642 MWt
(Results are based on initial core.)

FIGURE 4.3-15

REVISION 4, APRIL 1997



QUAD CITIES STATION UNITS 1 & 2
6-INCH WATER LEVEL SETPOINT CHANGE TRANSIENT ANALYSIS - 1642 MWt (Results are based on initial core.)
FIGURE 4.3-16 REVISION 4, APRIL 1997

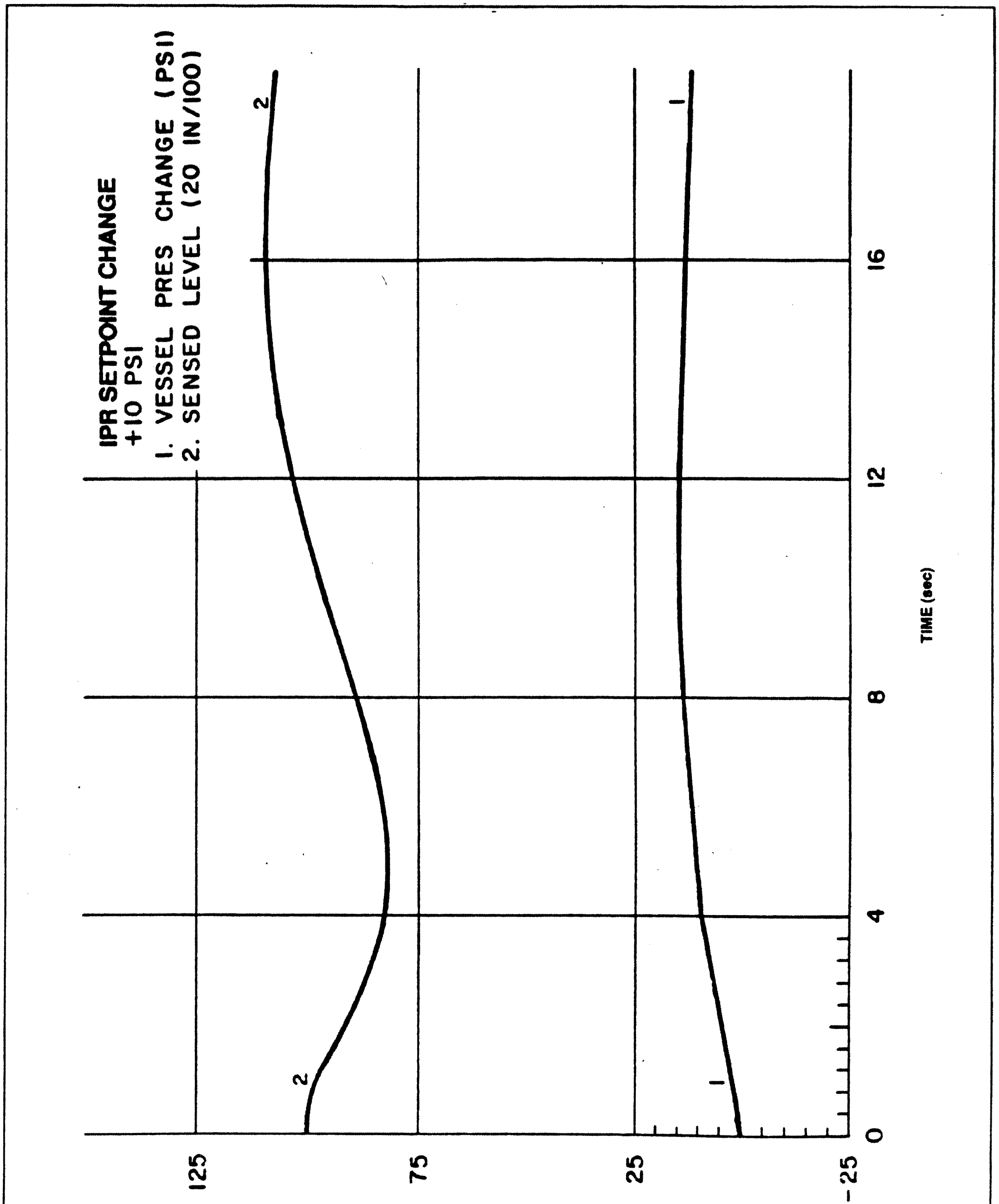


QUAD CITIES STATION
UNITS 1 & 2

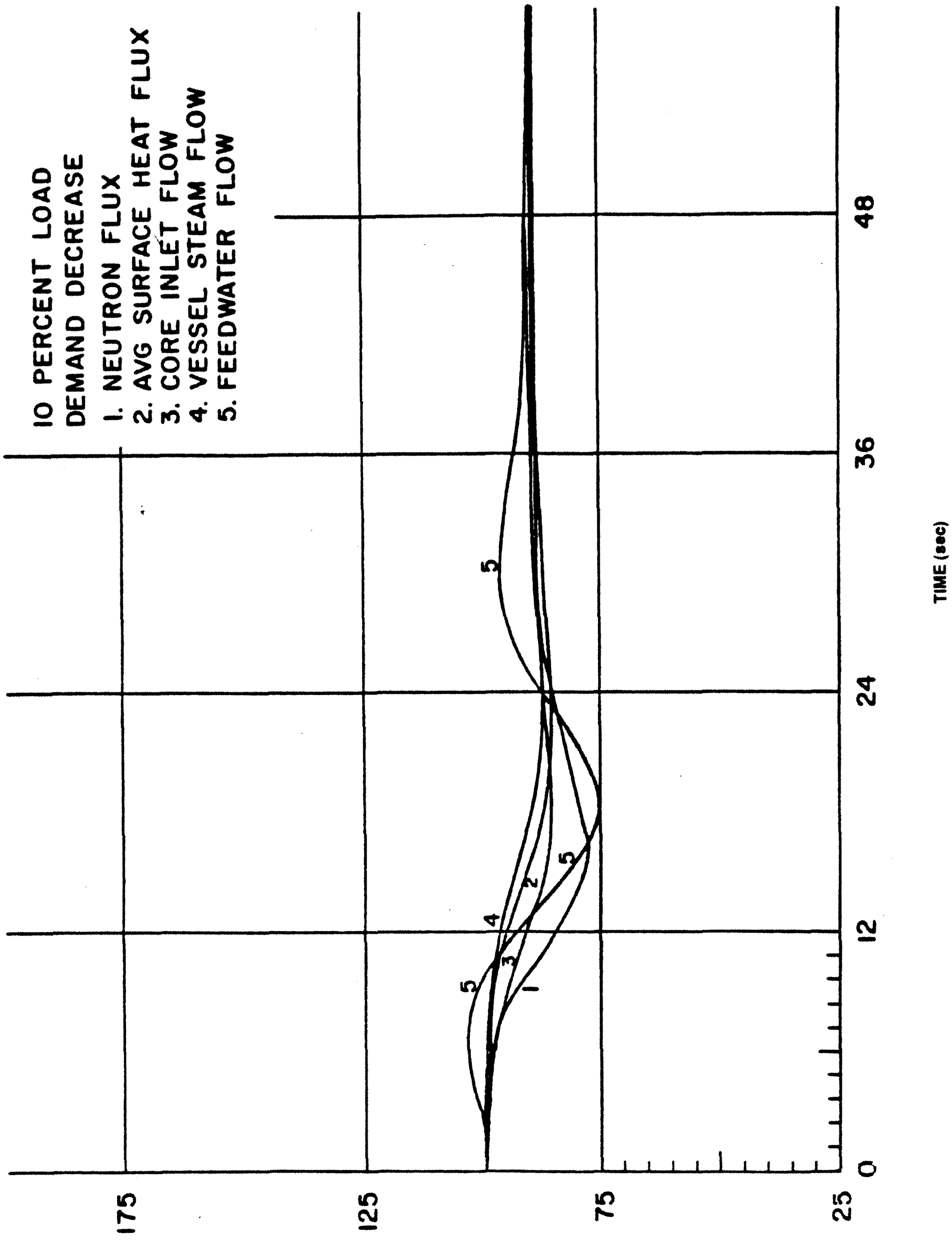
10 PSI SETPOINT CHANGE
TRANSIENT ANALYSIS - 2511 MWt
(Results are based on initial core.)

FIGURE 4.3-17

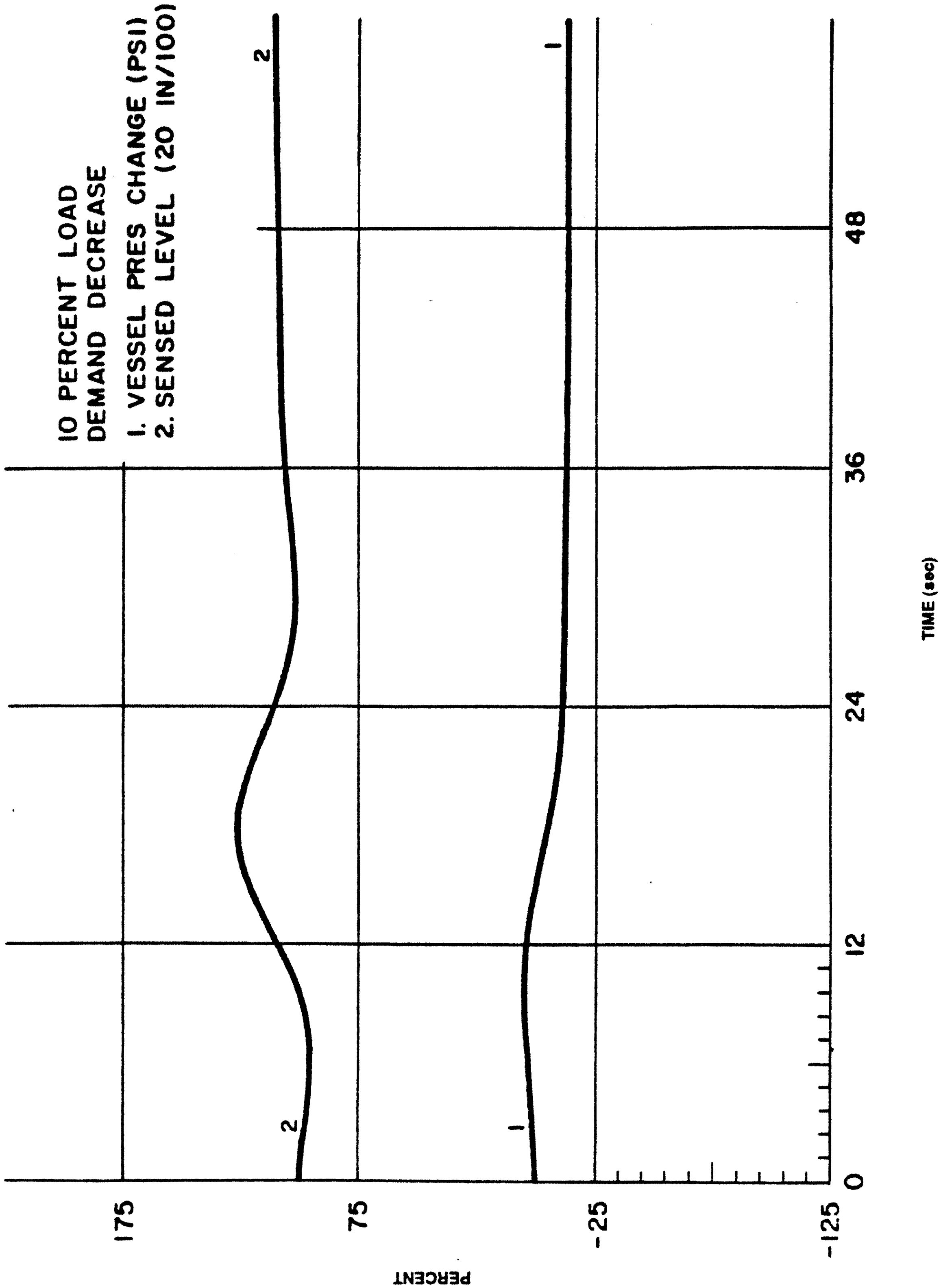
REVISION 4, APRIL 1997



QUAD CITIES STATION UNITS 1 & 2
10 PSI SETPOINT CHANGE TRANSIENT ANALYSIS - 2511 MWt (Results are based on initial core.)
FIGURE 4.3-18 REVISION 4, APRIL 1997



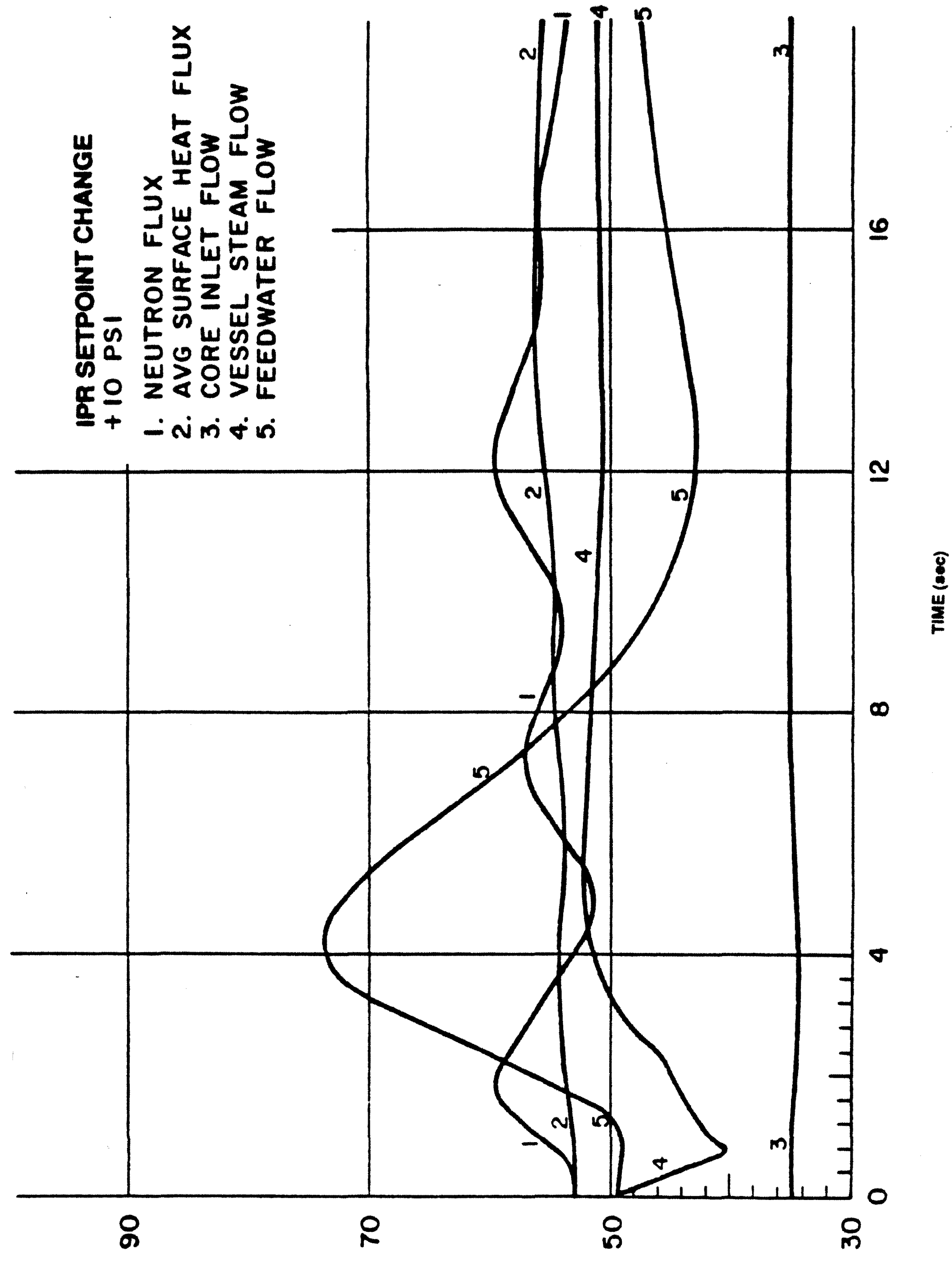
QUAD CITIES STATION UNITS 1 & 2
10% DEMAND LOAD CHANGE TRANSIENT ANALYSIS - 2511 MWt (Results are based on initial core.)
FIGURE 4.3-19 REVISION 4, APRIL 1997



QUAD CITIES STATION
UNITS 1 & 2

10% DEMAND LOAD CHANGE
TRANSIENT ANALYSIS - 2511 MWt
(Results are based on initial core.)

FIGURE 4.3-20
REVISION 4, APRIL 1997

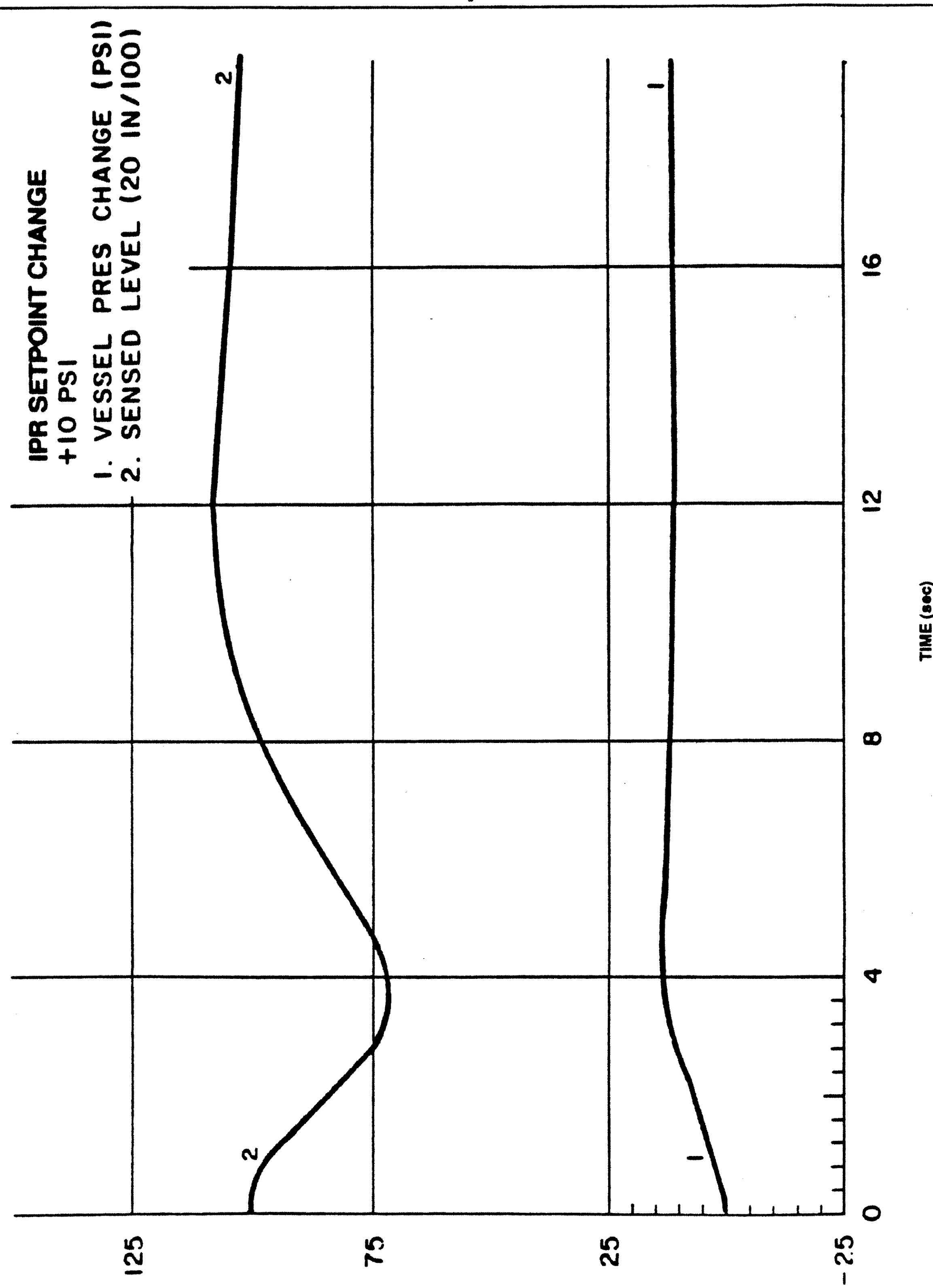


QUAD CITIES STATION
UNITS 1 & 2

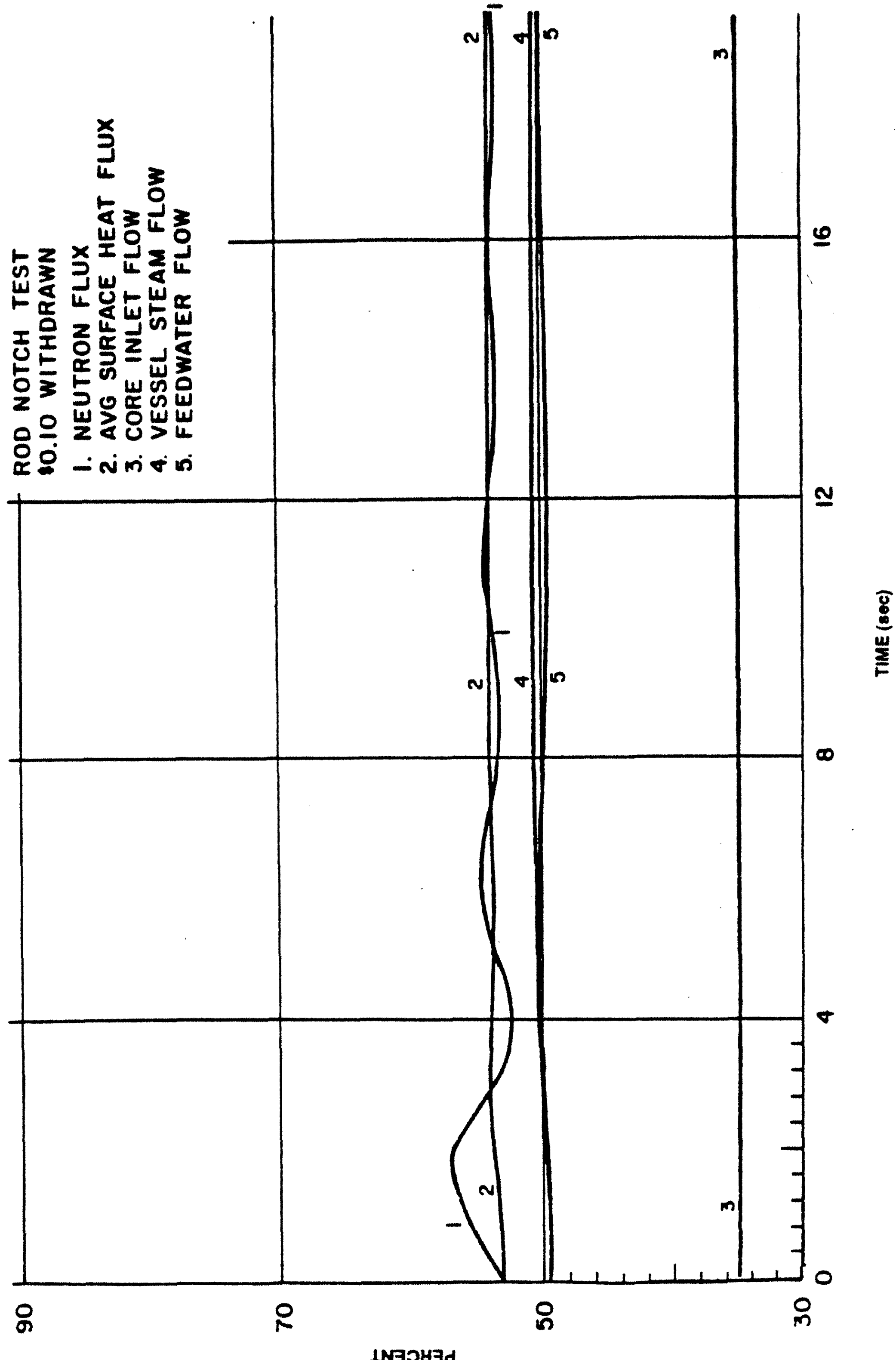
10 PSI SETPOINT CHANGE
TRANSIENT ANALYSIS - 1339 MWt
(Results are based on initial core.)

FIGURE 4.3-21

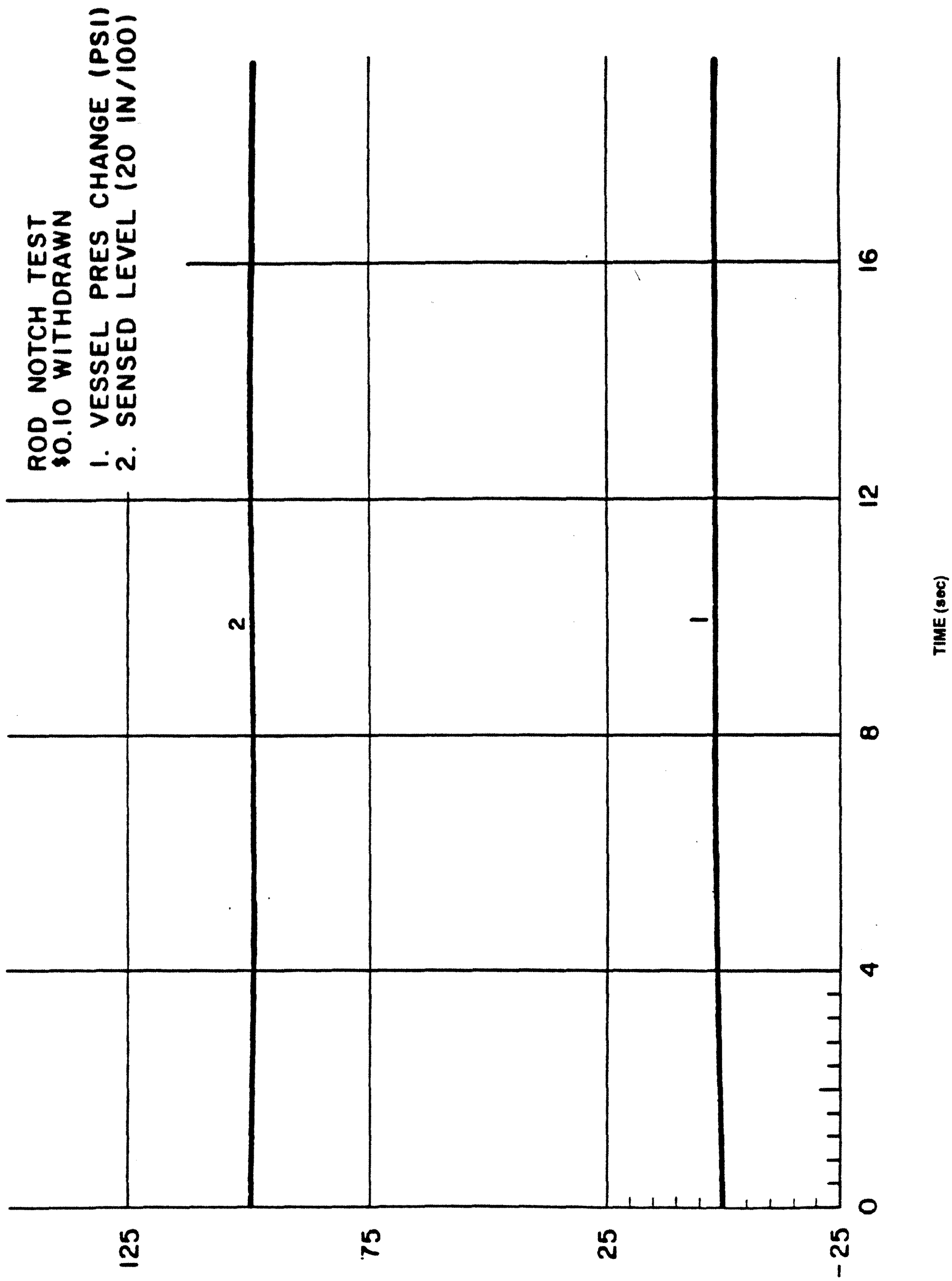
REVISION 4, APRIL 1997



QUAD CITIES STATION UNITS 1 & 2
10 PSI SETPOINT CHANGE TRANSIENT ANALYSIS - 1339 MWt (Results are based on initial core.)
FIGURE 4.3-22 REVISION 4, APRIL 1997



QUAD CITIES STATION UNITS 1 & 2
$0.10 ROD NOTCH REACTIVITY CHANGE TRANSIENT ANALYSIS - 1339 MWt (Results are based on initial core.)
FIGURE 4.3-23 REVISION 4, APRIL 1997

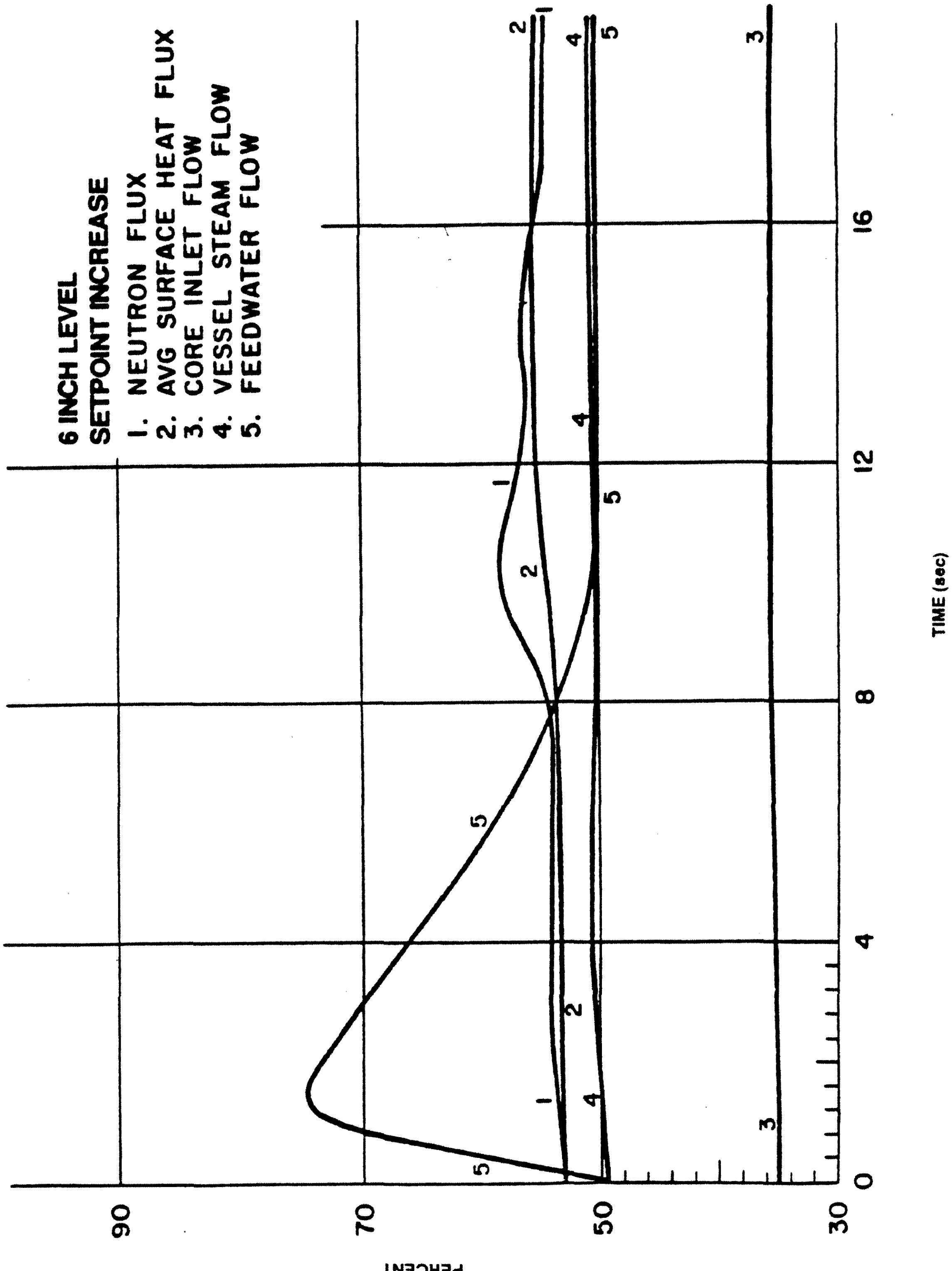


QUAD CITIES STATION
UNITS 1 & 2

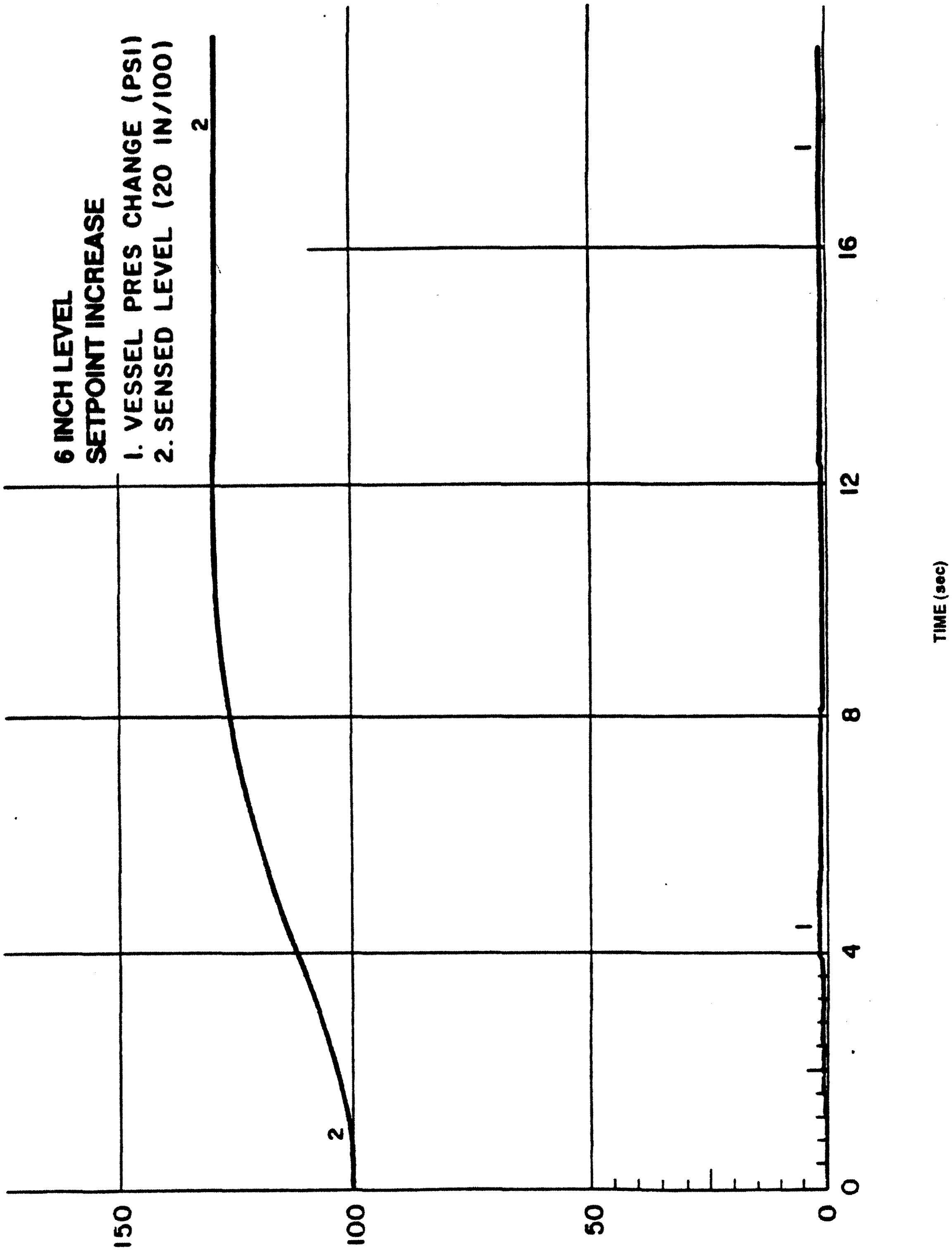
$0.10 ROD NOTCH REACTIVITY CHANGE
TRANSIENT ANALYSIS - 1339 MWt
(Results are based on initial core.)

FIGURE 4.3-24

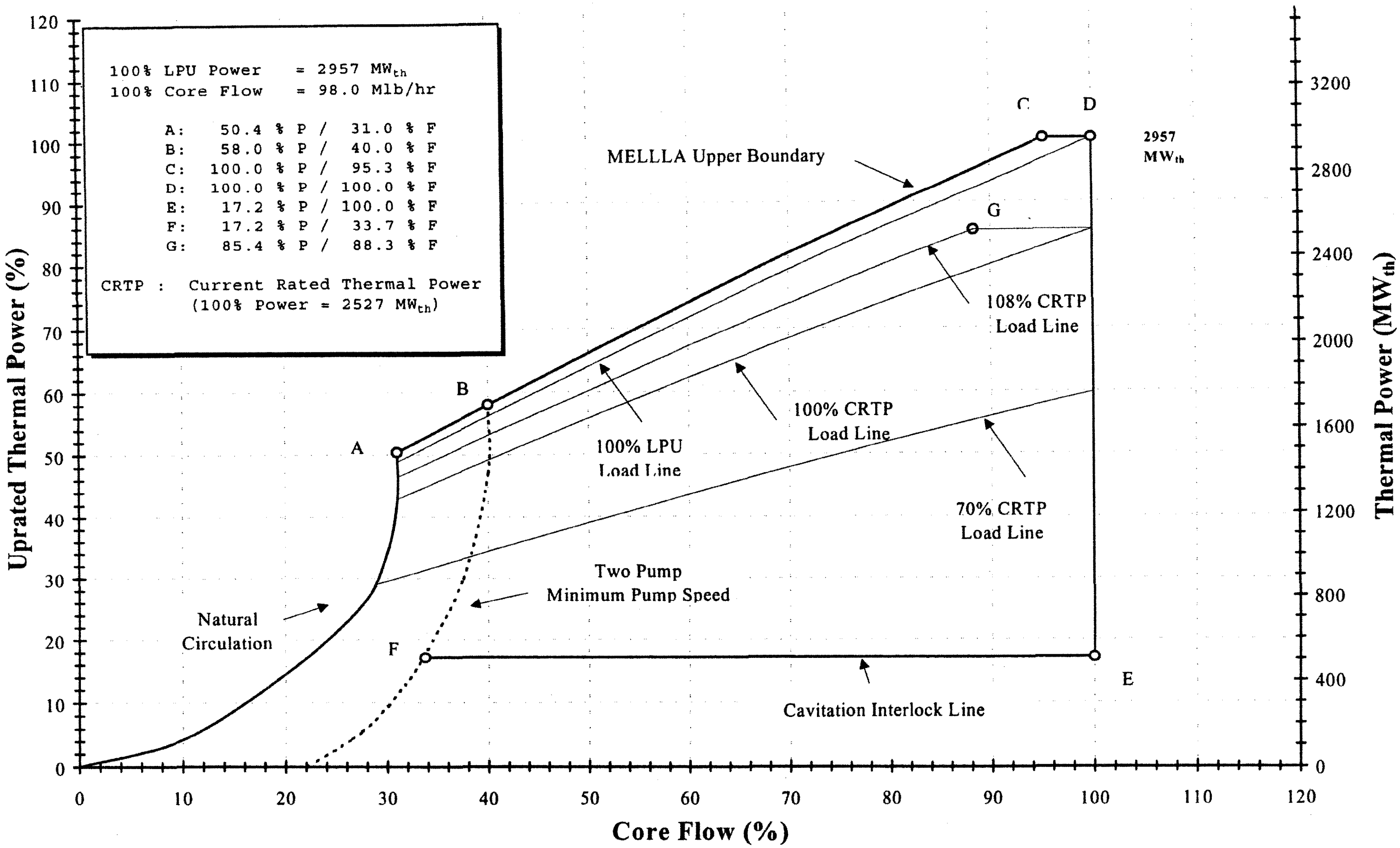
REVISION 4, APRIL 1997



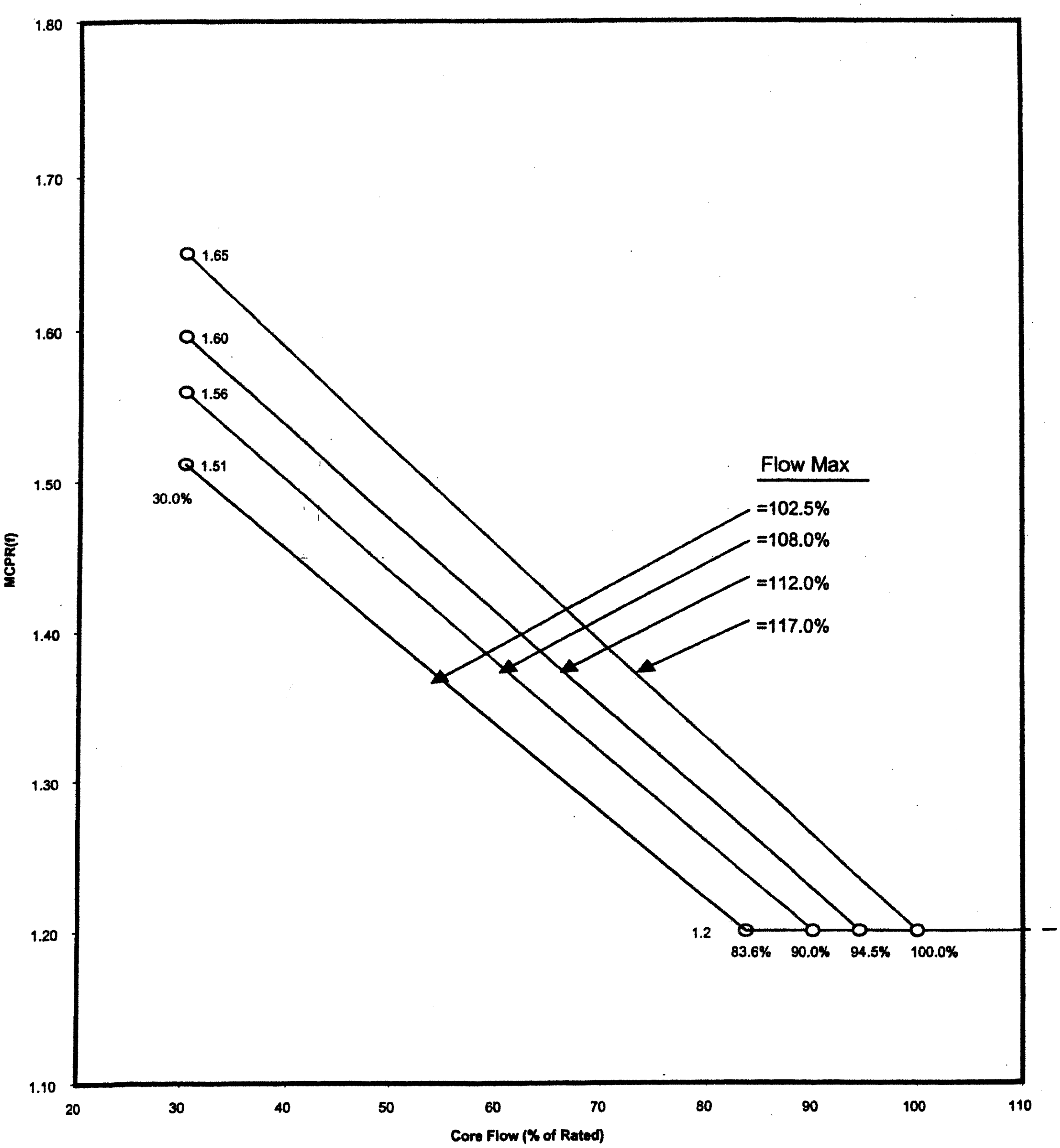
QUAD CITIES STATION UNITS 1 & 2
6-INCH WATER LEVEL SETPOINT CHANGE TRANSIENT ANALYSIS - 1339 MWt (Results are based on initial core.)
FIGURE 4.3-25 REVISION 4, APRIL 1997



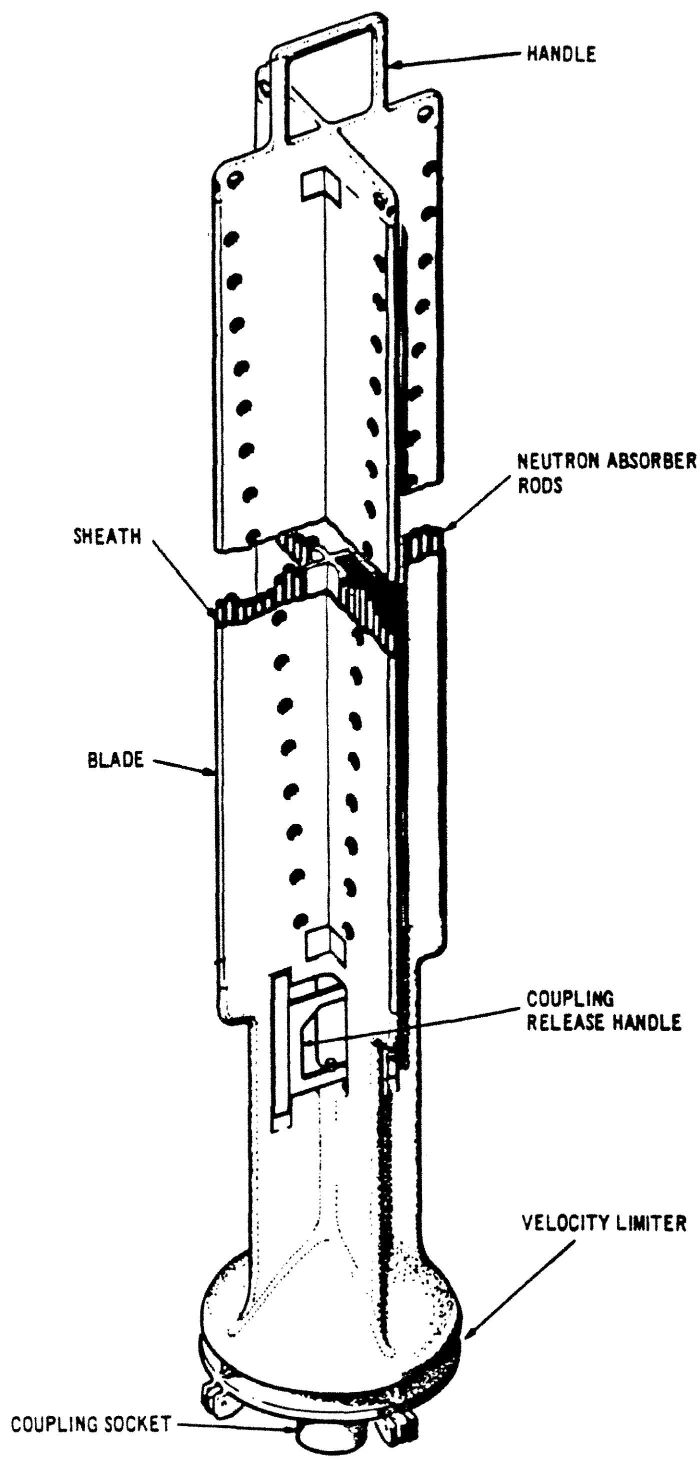
QUAD CITIES STATION UNITS 1 & 2
6-INCH WATER LEVEL SETPOINT CHANGE TRANSIENT ANALYSIS - 1339 MWt (Results are based on initial core.)
FIGURE 4.3-26 REVISION 4, APRIL 1997



QUAD CITIES STATION UNITS 1 & 2
TYPICAL POWER – FLOW MAP
FIGURE 4.4-1



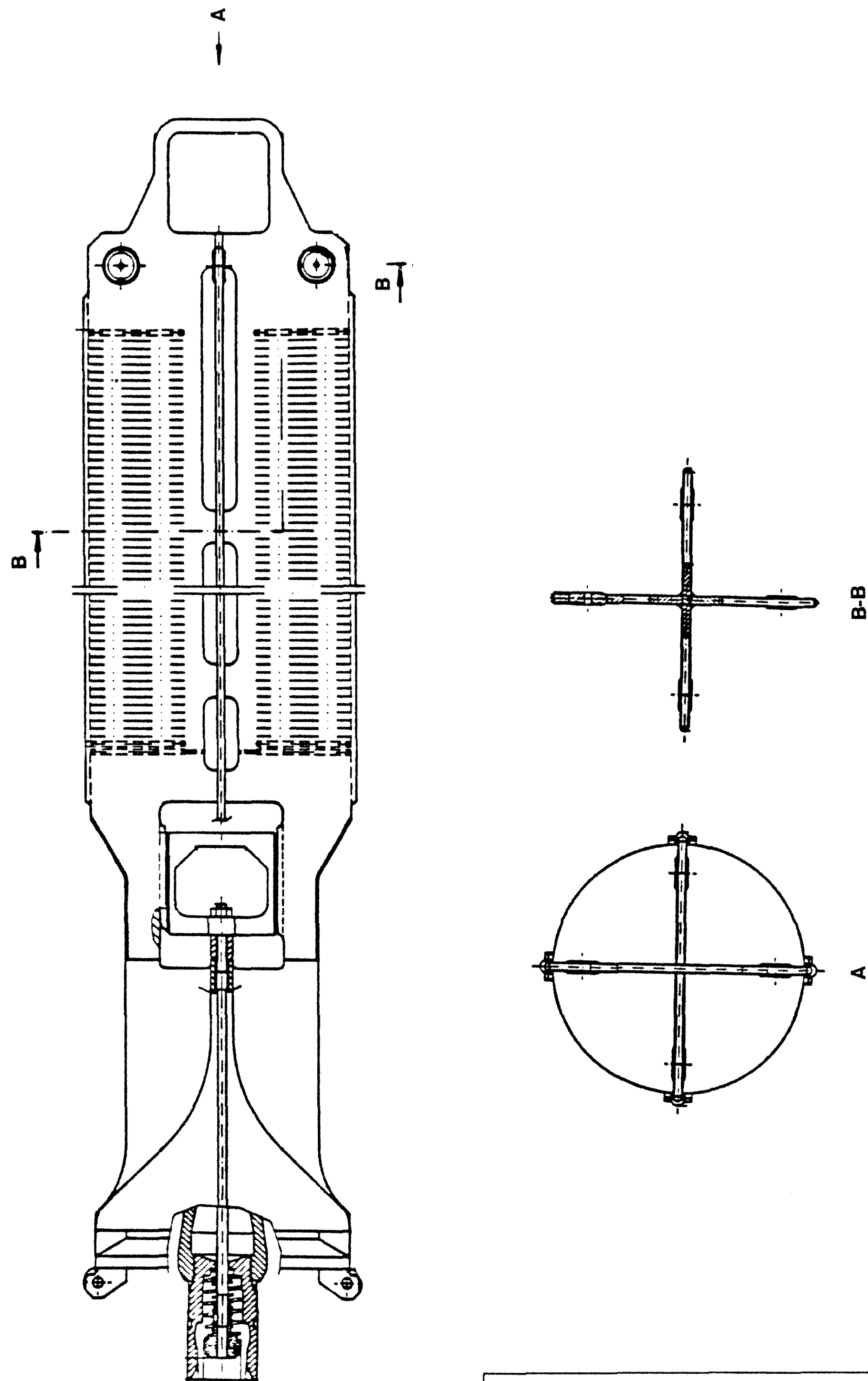
QUAD CITIES STATION Units 1 and 2
GE 14 Flow Dependant MCPR, MCPR(F) (Typical)
Figure 4.4-4



QUAD CITIES STATION
UNITS 1 & 2

ORIGINAL EQUIPMENT
CONTROL ROD - ISOMETRIC

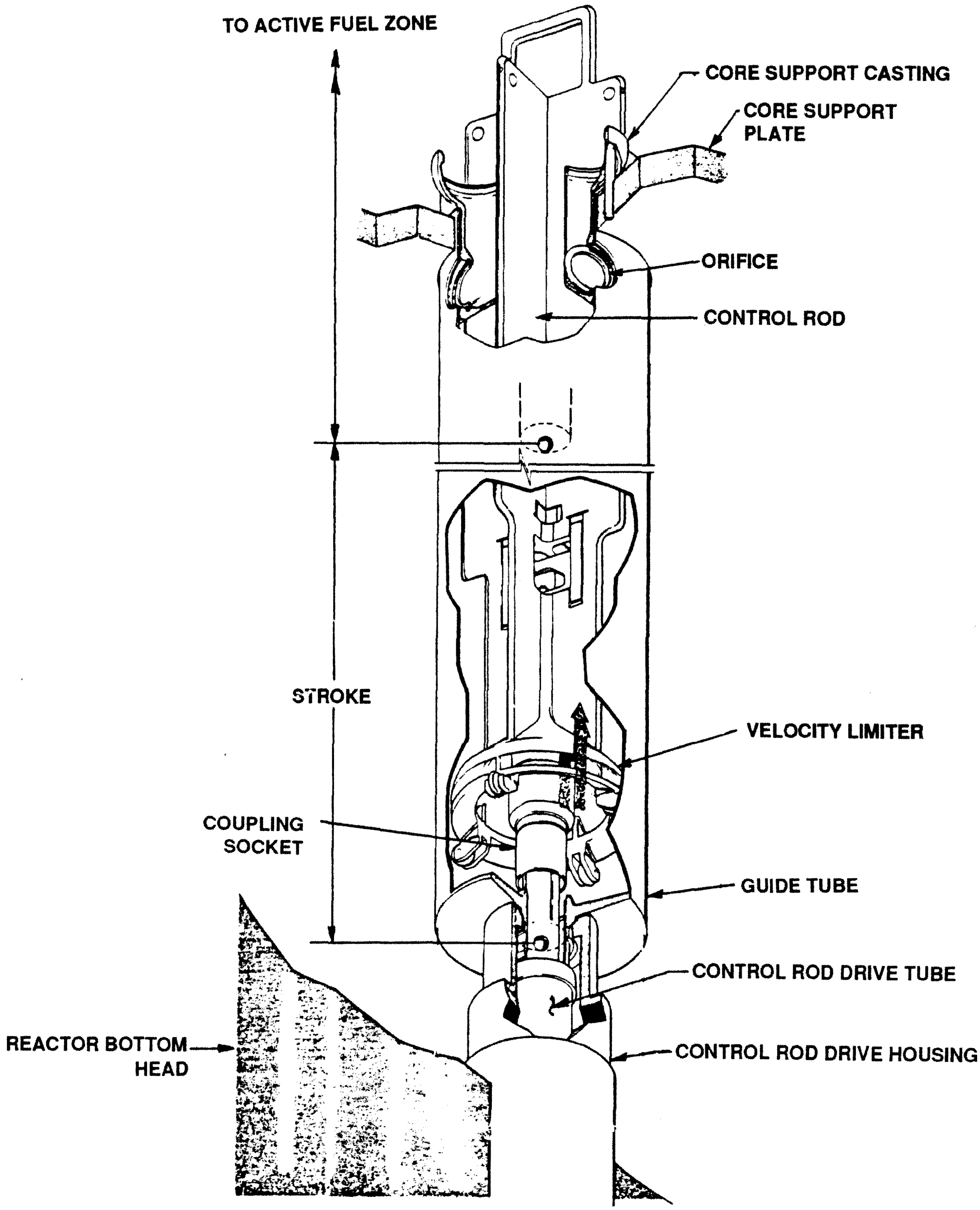
FIGURE 4.6-1



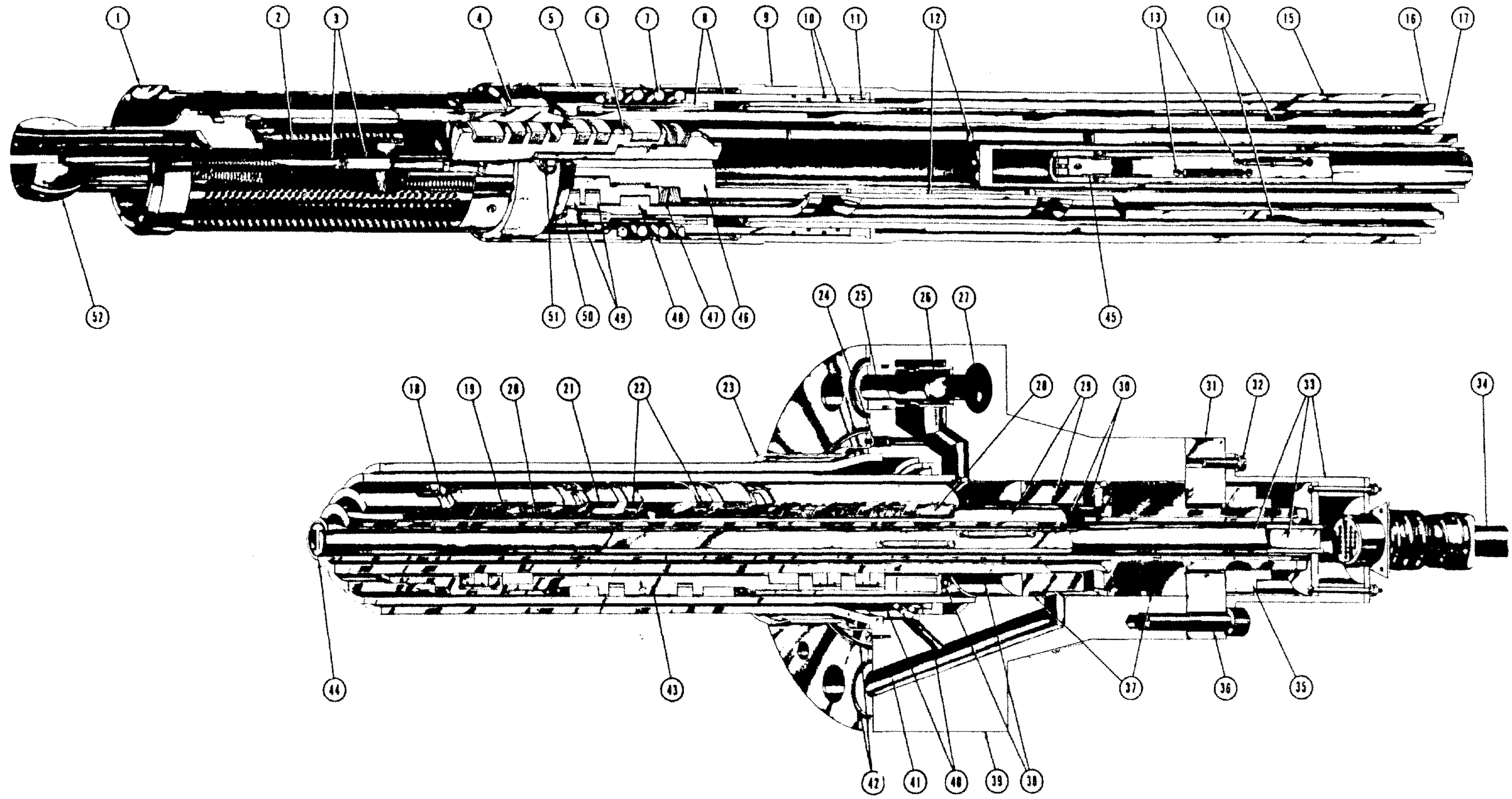
QUAD CITIES STATION
UNITS 1 & 2

ABB-ATOM CONTROL ROD

FIGURE 4.6-2



QUAD CITIES STATION UNITS 1 & 2
ORIGINAL VELOCITY LIMITER ISOMETRIC
FIGURE 4.6-3

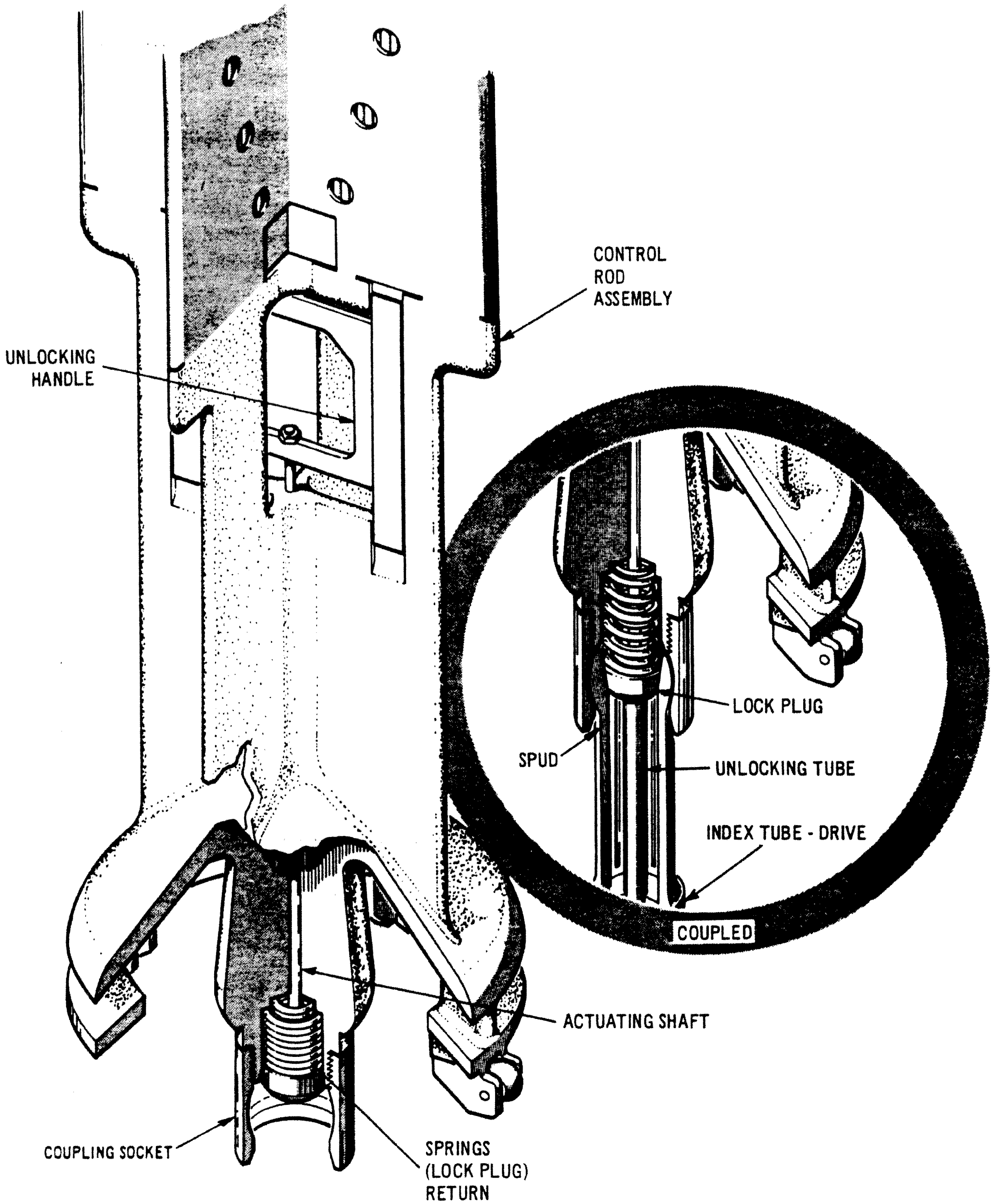


1. EXTERNAL FILTER ASSEMBLY
2. INTERNAL FILTER ASSEMBLY
3. UNCOUPLING ROD ASSEMBLY
4. GUIDE CAP
5. BARREL
6. STOP PISTON
7. COLLET SPRING
8. COLLET AND COLLET PISTON
9. COLLET HOUSING (Part of cylinder, tube and flange)
10. COLLET PISTON SEALS
11. SPACER (Part of cylinder, tube, and flange)
12. BUFFER ORIFICES (Typical)
13. POSITION INDICATOR SWITCHES (Typical)
14. LOCKING GROOVE (Typical)
15. OUTER TUBE (Part of cylinder, tube, and flange)
16. CYLINDER TUBE
17. INDEX TUBE
18. LOCKING BAND (Typical)
19. INTERNAL PISTON SEAL RINGS (Typical)
20. INTERNAL PISTON BUSHINGS (Typical)
21. EXTERNAL PISTON BUSHINGS
22. EXTERNAL PISTON SEALS
23. STRAINER
24. COOLING WATER ORIFICE
25. DRIVE-INSERT WATER INLET (Normal and scram, drive-withdrawal outlet)
26. BALL-SHUTTLE VALVE
27. REACTOR WATER INLET (Through drive housing)
28. SWITCH-ACTUATING MAGNET (Part of drive piston)
29. PISTON TUBE ASSEMBLY
30. DRIVE-WITHDRAW PORTS AND ANNULUS (Also scram outlet)
31. RING FLANGE
32. MACHINE SCREW (Typical)
33. POSITION INDICATOR PROBE
34. POSITION INDICATOR CABLE
35. PISTON TUBE NUT
36. CAP SCREW (Typical)
37. O-RING SEALS
38. DRIVE-INSERT PORTS AND ANNULUS
39. DRIVE FLANGE (Part of cylinder, tube, and flange)
40. UNLOCKING PORT AND ANNULUS (Withdraw pressure to collet piston)
41. DRIVE-WITHDRAW WATER INLET (Also outlet for scram water)
42. METAL O-RING SEAL (Drive to housing)
43. DRIVE PISTON
44. INDICATOR TUBE (Part of piston tube)
45. THERMOCOUPLE (Part of position indicator probe)
46. STUD (Part of piston tube)
47. SPRING WASHERS
48. STOP PISTON BUSHINGS(Typical)
49. STOP PISTON SEAL RINGS (Typical)
50. COLLET FINGER (Typical)
51. COTTER PIN
52. COUPLING SPUD

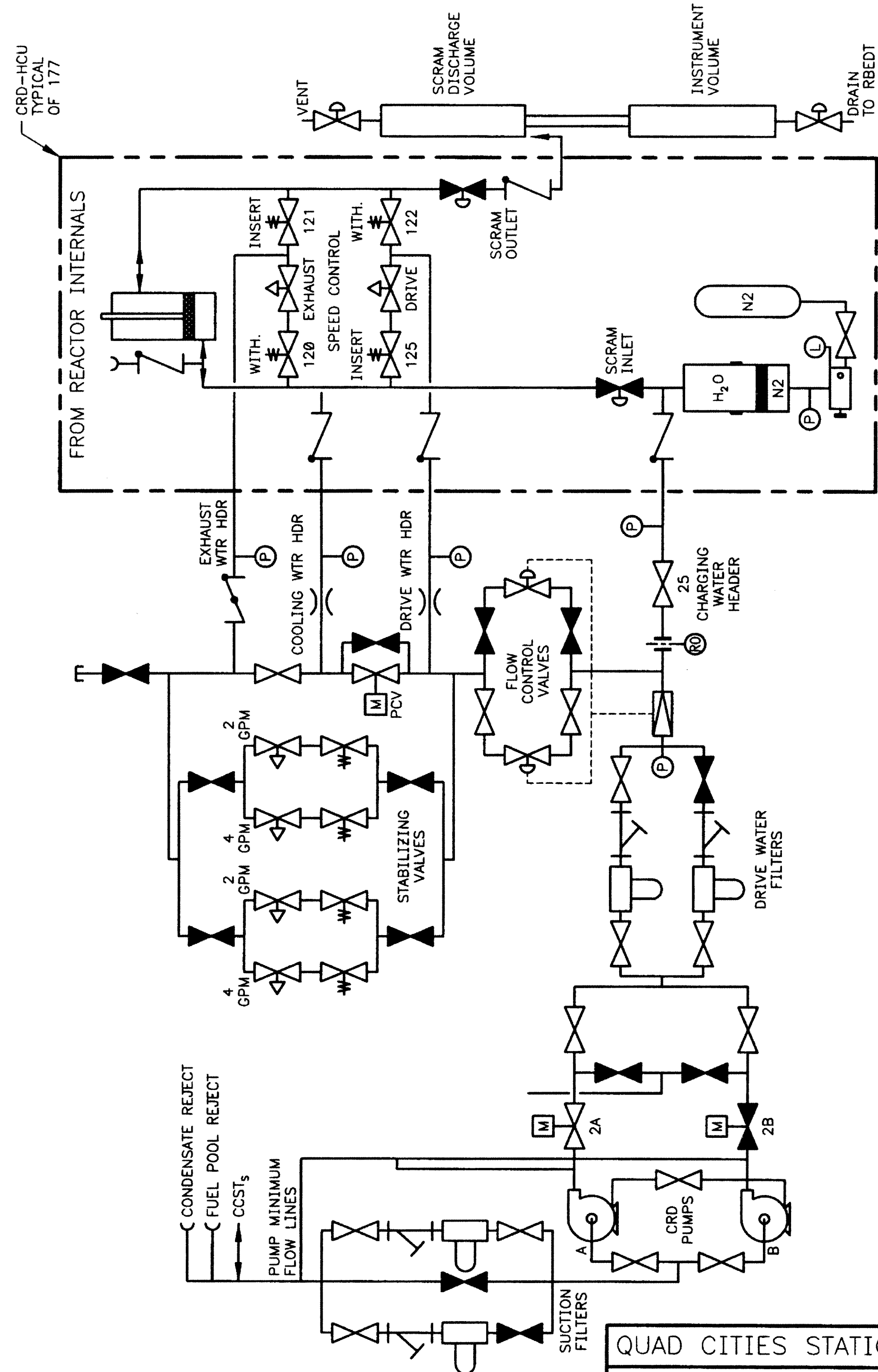
QUAD CITIES STATION
UNITS 1 & 2

CONTROL ROD DRIVE - CUTAWAY

FIGURE 4.6-4



QUAD CITIES STATION UNITS 1 & 2	
CONTROL ROD-TO-DRIVE COUPLING ISOMETRIC	
FIGURE 4.6-5	

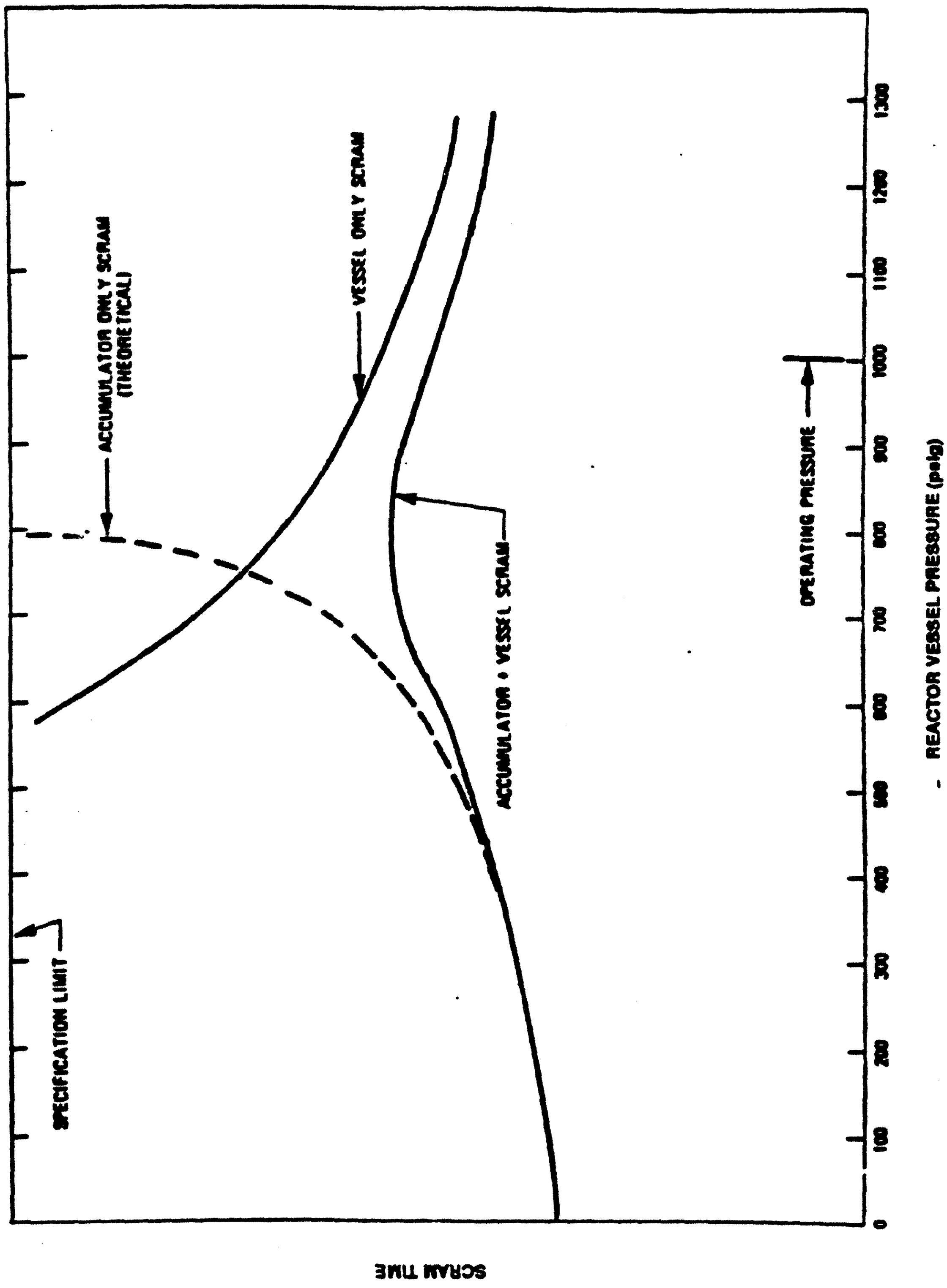


QUAD CITIES STATION UNITS 1 & 2

DIAGRAM OF CONTROL ROD
DRIVE HYDRAULIC PIPING

FIGURE 4.6-6

REVISION 7, JANUARY 2003

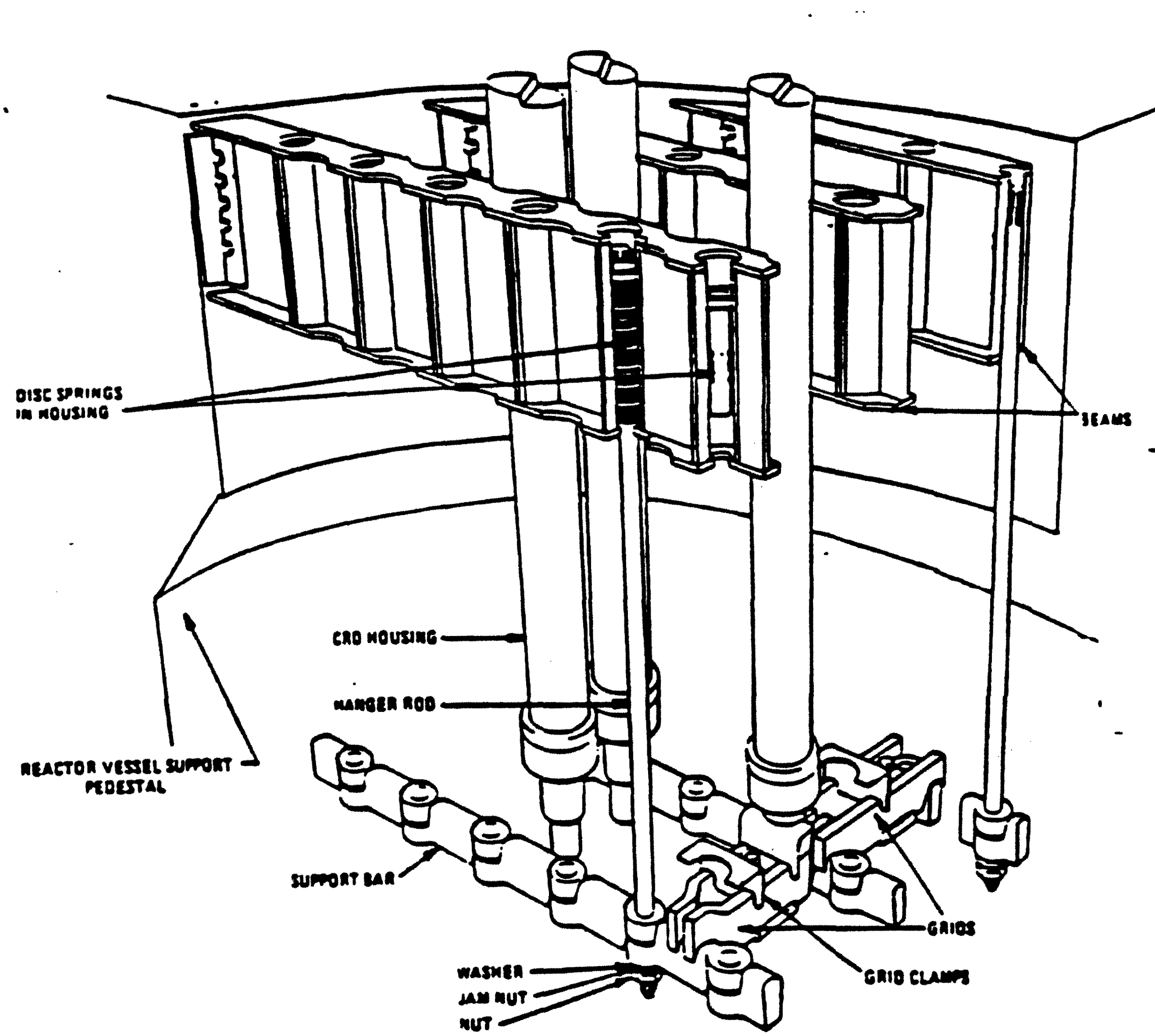


QUAD CITIES STATION
UNITS 1 & 2

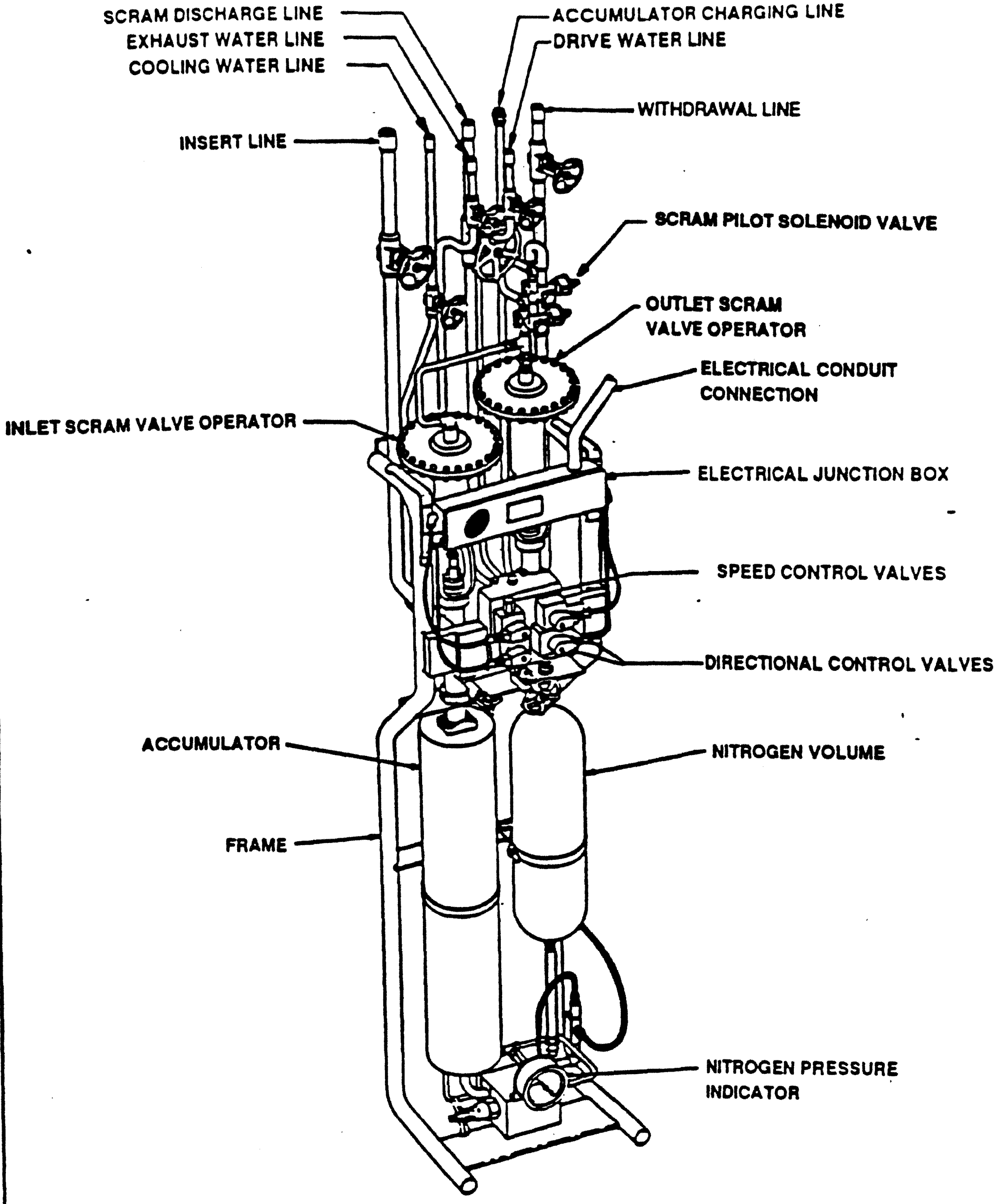
TYPICAL SCRAM TIME CHARACTERISTIC

FIGURE 4.6-7

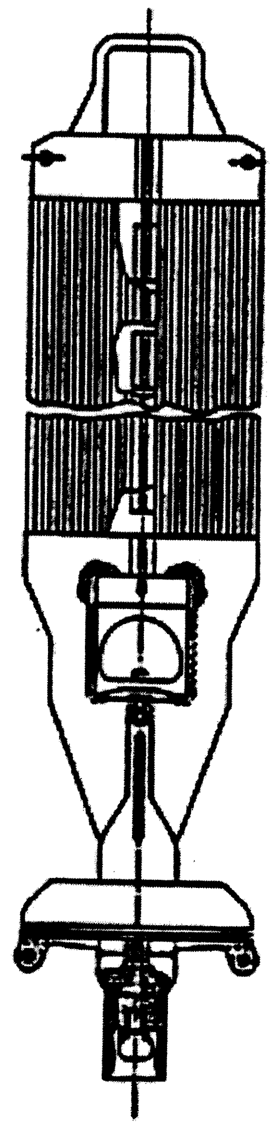
REV.3, DECEMBER 1995



QUAD CITIES STATION UNITS 1 & 2
CONTROL ROD HOUSING SUPPORT ISOMETRIC
FIGURE 4.6-8 REV. 3, DECEMBER 1995



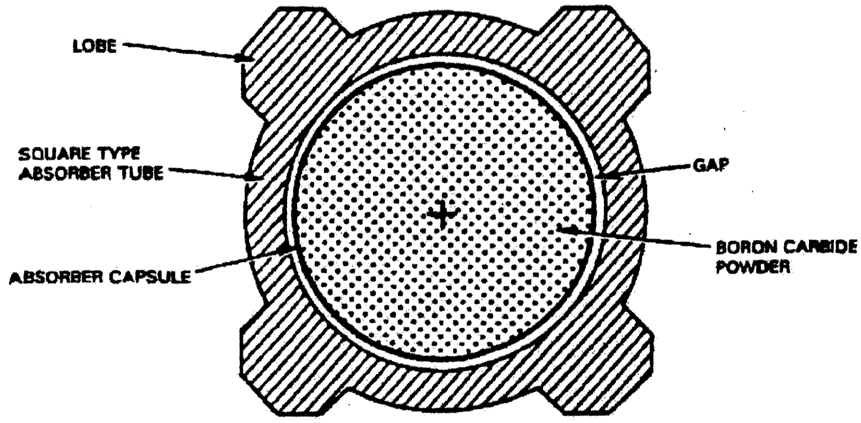
QUAD CITIES STATION UNITS 1 & 2
CONTROL ROD DRIVE HYDRAULIC CONTROL UNIT - ISOMETRIC
FIGURE 4.6-9 REV. 3, DECEMBER 1995



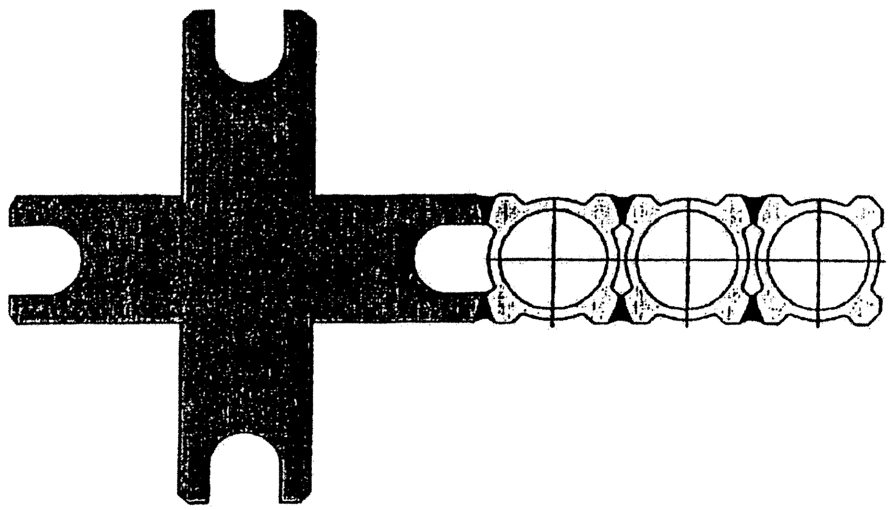
QUAD CITIES STATION
UNITS 1 & 2

GE MARATHON CONTROL BLADE ISOMETRIC

FIGURE 4.6-10



Absorber Tube Cross Section



Absorber Tubes Welded to Tie-Rod

QUAD CITIES STATION UNITS 1 & 2
GE MARATHON CONTROL BLADE DETAILS
FIGURE 4.6-11