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Washington Public Power Supply System
P.O. Box 968 3000 George Washington Way Richland, Washington 99352 (509) 372-5000

March 21, 1983
G02-83-244

Docket No. 50-397

Director of Nuclear Reactor Regulation
Attention: Mr. A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Schwencer:

Subject: NUCLEAR PROJECT NO. 2
REWRITE OF CHAPTER 4, SECTIONS 4.1
THROUGH 4.4 OF THE WNP-2 FSAR

Enclosed are sixty (60) copies of our rewrite to Sections 4.1 through 4.4 of Chapter 4 of the WNP-2 FSAR. The enclosed will be incorporated into Amendment No. 30 to be issued in June 1983.

Very truly yours,

G. D. Bouchey
G..D. Bouchey
Manager, Nuclear Safety and Regulatory Programs

jca
Enclosure

cc: R Auluck - NRC
WS Chin - BPA
A Toth - NRC Site

B001
1/60

CHAPTER 4

REACTOR

4.1 SUMMARY DESCRIPTION

This chapter was prepared utilizing ~~the BWR/4 and BWR/5 Fuel Design Topical, NEDO 20944 and NEDE-20944-P, October 1976~~. ~~Applicable sections of the topical report are referenced as noted in 4.2 Fuel Design, 4.3 Nuclear Design and 4.4 Thermal-Hydraulic Design. The remaining sections of Chapter 4 are herein included. Reference to standardized information contained in the topical report is made consistent with the NRC overall standardization philosophy~~.

The reactor assembly consists of the reactor vessel, its internal components of the core, shroud, steam separator and dryer assemblies, and jet pumps. Also included in the reactor assembly are the control rods, control rod drive housings, and the control rod drives. Figure 5.3-1, Reactor Vessel Internals, shows the arrangement of reactor assembly components. A summary of the important design and performance characteristics is given in 1.3. Loading conditions for reactor assembly components are specified in 3.9.

4.1.1 REACTOR VESSEL

The reactor vessel design and description are covered in 5.3.

4.1.2 REACTOR INTERNAL COMPONENTS

The major reactor internal components are the core (fuel, channels, control blades, and instrumentation), the core support structure (including the shroud, top guide and core plate), the shroud head and steam separator assembly, the steam dryer assembly, the feedwater spargers, the core spray spargers, and the jet pumps. Except for the Zircaloy in the reactor core, these reactor internals are stainless steel or other corrosion resistant alloys. All major internal components of the vessel can be removed except the jet pump diffusers, the jet pump risers, the shroud, the core spray lines, spargers, and the feedwater sparger. The removal of the steam dryers, shroud head and steam separators, fuel assemblies, in-core assemblies, control rods, orificed fuel supports, and control rod guide tubes, can be accomplished on a routine basis.

the General Electric Standard Application for Reactor Fuel, Licensing Topical Report NEDO-24011 and NEDE-24011-P.

The in-core location of the startup and source range instruments provides coverage of the large reactor core and provides an acceptable signal-to-noise ratio and neutron-to-gamma ratio. All in-core instrument leads enter from the bottom and the instruments are in service during refueling. In-core instrumentation is further discussed in 7.6.1.5 and 7.7.1.6.

c. As shown by experience obtained at Dresden-1 and other plants, the operator, utilizing the in-core flux monitor system, can maintain the desired power distribution within a large core by proper control rod scheduling.

d. The Zircaloy-4 reusable channels provide a fixed flow path for the boiling coolant, serve as a guiding surface for the control rods, and protect the fuel during handling operations.

e. The mechanical reactivity control permits criticality checks during refueling and provides maximum plant safety. The core is designed to be subcritical at any time in its operating history with any one control rod fully withdrawn.

f. The selected control rod pitch represents a practical value of individual control rod reactivity worth, and allows ample clearance below the pressure vessel between control rod drive mechanisms for ease of maintenance and removal.

4.1.2.1.2 Core Configuration

The reactor core is arranged as an upright circular cylinder containing a large number of fuel cells and is located within the reactor vessel. The coolant flows upward through the core. The core arrangement (plan view) and the lattice configuration, are shown in Figure 4.~~1-1~~ 2-5.

4.1.2.1.3 Fuel Assembly Description

As can be seen from the referenced figures, the boiling water reactor core is composed of essentially two components--fuel assemblies and control rods. The fuel assembly and control rod mechanical configurations (see Figures 4.2-1, ~~and~~ 4.2-~~8~~2, ~~a. and b.)~~ and Figure 1-1 of Reference 4.1-12) are basically the same as used in Dresden-1 and in all subsequent General Electric boiling water reactors.

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4.1.2.1.3.1 Fuel Rod

A fuel rod consists of UO_2 pellets and a Zircaloy-2 cladding tube. A fuel rod is made by stacking pellets into a Zircaloy -2 cladding tube which is evacuated and back-filled with 3 ATM of helium, and sealed by welding Zircaloy end plugs in each end of the tube.

The BWR fuel rod is designed as a pressure vessel. The ASME Boiler and Pressure Vessel Code, Section III, is used as a guide in the mechanical design and stress analysis of the fuel rod.

The rod is designed to withstand the applied loads, both external and internal. The fuel pellet is sized to provide sufficient volume within the fuel tube to accommodate differential expansion between fuel and clad. Overall fuel rod design is conservative in its accommodation of the mechanisms affecting fuel in a BWR environment. Fuel rod design basis are discussed in more detail in 4.2.1.~~1.2.~~

4.1.2.1.3.2 Fuel Bundle

Each fuel bundle contains 62 fuel rods and two water rods which are spaced and supported in a square (8x8) array by a lower and upper tie plate. The fuel bundle has two important design features:

a. The bundle design places minimum external forces on a fuel rod; each fuel rod is free to expand in the axial direction.

b. The unique structural design permits the removal and replacement, if required, of individual fuel rods.

The fuel assemblies of which the core is comprised are designed to meet all the criteria for core performance and to provide ease of handling. Selected fuel rods in each assembly differ from the others in uranium enrichment. This arrangement produces more uniform power production across the fuel assembly, and thus allows a significant reduction in the amount of heat transfer surface required to satisfy the design thermal limitations.

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4.1.3 REACTIVITY CONTROL SYSTEMS

4.1.3.1 Operation

The control rods perform dual functions of power distribution shaping and reactivity control. Power distribution in the core is controlled during operation of the reactor by manipulation of selected patterns of rods. The rods, which enter from the bottom of the near-cylindrical reactor core, are positioned in such a manner to counter-balance steam voids in the top of the core and effect significant power flattening.

These groups of control elements, used for power flattening, experience a somewhat higher duty cycle and neutron exposure than the other rods in the control system.

The reactivity control function requires that all rods be available for either reactor "scram" (prompt shutdown) or reactivity regulation. Because of this, the control elements are mechanically designed to withstand the dynamic forces resulting from a scram. They are connected to bottom-mounted, hydraulically actuated drive mechanisms which allow either axial positioning for reactivity regulation or rapid scram insertion. The design of the rod-to-drive connection permits each blade to be attached or detached from its drive without disturbing the remainder of the control system. The bottom-mounted drives permit the entire control system to be left intact and operable for tests with the reactor vessel open.

4.1.3.2 Description of Rods

The cruciform shaped control rods contain 76 stainless steel tubes (19 tubes in each wing of the cruciform) filled with vibration compacted boron-carbide powder. The tubes are seal welded with end plugs on either end. 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 individual tubes act as pressure vessels to contain the helium gas released by the boron-neutron capture reaction.

The tubes are held in a cruciform array by a stainless steel sheath extending the full length of the tubes. A top handle, shown in Figure 4.2-~~8~~, aligns the tubes and provides structural rigidity at the top of the control rod. Rollers, housed in the handle, provide guidance for control rod insertion and withdrawal. A bottom casting is also used to provide structural rigidity and contains positioning rollers

and a parachute-shaped velocity limiter. The handle and lower casting are welded into a single structure by means of a small cruciform post located in the center of the control rod. A steel stiffener is located approximately at the midspan of each cruciform wing. The control rods can be positioned at 6-in. steps and have a nominal withdrawal and insertion speed of 3 in/sec.

The velocity limiter (shown in Figure 4.2-30) is a device which is an integral part of the control rod and protects against the low probability of a rod drop accident. It is designed to limit the free fall velocity and reactivity insertion rate of a control rod so that minimum fuel damage would occur. It is a one-way device, in that control rod scram time is not significantly affected.

Control rods are cooled by the core leakage (bypass) flow. The core leakage flow is made up of recirculation flow that leaks through the several leakage flow paths, which are:

a. The area between fuel channel and fuel assembly nosepiece;

b. The area between fuel assembly nosepiece and fuel support piece;

c. Holes in the lower tie plate;

d ~~c~~. The area between fuel support piece and core plate;

e ~~d~~. The area between core plate and shroud;

f ~~e~~. Holes in the core plate near power range monitor instrument guide tubes; ~~and~~

g ~~f~~. Various leakage paths around the control rod guide tubes; and

h. Control rod drive cooling water.

4.1.3.3 Supplementary Reactivity Control

The control requirements of the initial core are designed to be considerably in excess of the equilibrium core requirements because of the long initial operating cycle. The initial core control requirements are met by use of the combined effects of the movable control rods and a supplemental burnable poison. The supplementary burnable poison is gadolinia (Gd_2O_3) mixed with UO_2 in selected fuel rods in each fuel bundle.

spill over onto 4.1-10

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4.1.4 ANALYSIS TECHNIQUES

4.1.4.1 Reactor Internal Components

Computer codes used for the analysis of the internal components are listed as follows:

a. MASS

b. SNAP (MULTISHELL)

c. GASP

d. NOHEAT

e. FINITE

f. DYSEA

g. SHELL 5

h. HEATER

i. FAP-71

j. CREEP-PLAST

Detail descriptions of these programs are given in the following sections:

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4.1.4.3 Reactor Systems Dynamics

The analysis techniques and computer codes used in reactor systems dynamics are described in section 4 of Reference 4.1-10. A complete stability analysis for the reactor coolant system is also provided in 4.4.4.6.

4.1.4.4 Nuclear Engineering Analysis

The analysis techniques are described and referenced in section 3.1 ~~4.3.3.~~ of Reference 4.1-12. The codes used in the analysis are:

Computer Code	Function
Lattice Physics Model	Calculates average few-group cross sections, bundle reactivities, and relative fuel rod powers within the fuel bundle.
BWR Reactor Simulator	Calculates three-dimensional nodal power distributions, exposures and thermal hydraulic characteristics as burnup progresses.

4.1.4.5 Neutron Fluence Calculations

Neutron vessel fluence calculations were carried out using a one-dimensional discrete ordinates Sn transport code with general anisotropic scattering.

This code is a modification of a widely used discrete ordinates code which will solve a wide variety of radiation transport problems. The program will solve both fixed source and multiplication problems. Slab, cylinder, and spherical geometry are allowed with various boundary conditions. The fluence calculations incorporated as an initial starting point neutron fission distributions prepared from core physics data as a distributed source. Anisotropic scattering was considered for all regions. The cross sections were prepared with a 1/E flux weighting, P_L matrices for anisotropic scattering but did not include resonance self-shielding factors. Fast neutron fluxes at locations other than the core mid-plane were calculated using a two-dimensional discrete ordinate code.

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4.1.5 REFERENCES

4.1-1 Crowther, R. L. "Xenon Considerations in Design of Boiling Water Reactors," APED-5640, June 1968.

4.1-2 Beitch, L., "Shell Structures Solved Numerically by Using a Network of Partial Panels," AIAA Journal, Volume 5, No. 3, March 1967.

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4.1-4 I. Farhoomand and E. L. Wilson, "Non-Linear Heat Transfer Analysis of Axisymmetric Solids," SESM Report SESM71-6, University of California at Berkeley, Berkeley, California, 1971.

4.1-5 J. E. McConnelee, "Finite-Users Manual", General Electric TIS Report DF 69SL206, March 1969.

4.1-6 R. W. Clough and C. P. Johnson, "A Finite Element Approximation For the Analysis of Thin Shells," International Journal Solid Structures, Vol. 4, 1968.

4.1-7 "A Computer Program For the Structural Analysis of Arbitrary Three-Dimensional Thin Shells," Report No. GA-9952, Gulf General Atomic.

4.1-8 Burgess, A. B., "User Guide and Engineering Description of HEATER Computer Program," March 1974.

4.1-9 Young, L. J., "FAP-71 (Fatigue Analysis Program) Computer Code," GE/NED Design Analysis Unit R. A. Report No. 49, January 1972.

4.1-10 Carmichael, L. A. and Scatena, G. J., "Stability and Dynamic Performance of the General Electric Boiling Water Reactor," APED-5652.

4.1-11 Y. R. Rashid, "Theory Report for Creep-Plast Computer Program," GEAP-10546, AEC Research and Development Report, January 1972.

4.1-12 "General Electric Standard Application for Reactor Fuel", NEDO-24011 and NEDE-24011-P, January 1982.

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4.2 FUEL SYSTEM DESIGN

See Appendix A, Section A.4.2 of Reference ~~1~~ 4.2-1.

4.2.1 Design Bases

cops ~~consistent with 4.0.0.1~~

See Appendix A, Subsection A.4.2.1 of Reference ~~1~~ 4.2-1.

4.2.2 Description and Design Drawings

cops ~~same throughout (see 4.1.1)~~

See Appendix A, Subsection A.4.2.2 of Reference ~~1~~ 4.2-1.

4.2.2.1 Control Rods

The control rods (Figures 4.2-1 and 4.2-2) perform the dual function of power shaping and reactivity control. A design drawing of the control blade is seen in Figures 4.2-1 and 4.2-2. Power distribution in the core is controlled during operation of the reactor by manipulating selected patterns of control rods. Control rod ~~displacement~~ withdrawal tends to counterbalance steam void effects at the top of the core and results in significant axial power flattening.

The control rod consists of a sheathed cruciform array of stainless steel tubes filled with boron-carbide powder. The control rods are 9.88 inches in total span and are separated uniformly throughout the core on a 12-inch pitch maximum. Each control rod is surrounded by four fuel assemblies.

The main structural member of a control rod is made of Type-304 stainless steel and consists of a top handle, a bottom casting with a velocity limiter and control rod drive coupling, a vertical cruciform center post, and four U-shaped absorber tube sheaths. The top handle, bottom casting, and center post are welded into a single skeletal structure. The U-shaped sheaths are resistance welded to the center post, handle, and castings to form a rigid

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~~4.2.2.1 Control Rods (Continued)~~

housing to contain the boron-carbide-filled absorber rods. Rollers at the top and bottom of the control rod guide the control rod as it is inserted and withdrawn from the core. The control rods are cooled by the core bypass flow. The U-shaped sheaths are perforated to allow the coolant to circulate freely about the absorber tubes. Operating experience has shown that control rods constructed as described above are not susceptible to dimensional distortions.

The boron-carbide (B_4C) powder in the absorber tubes is compacted to about 70 percent of its theoretical density. The boron-carbide contains a minimum of 76.5 percent by weight natural boron. The boron-10 (B-10) minimum content of the boron is 18 percent by weight. Absorber tubes are made of Type-304 stainless steel. Each absorber tube is 0.188 inches in outside diameter and has a 0.025-in. wall thickness. Absorber tubes are sealed by a plug welded into each end. The boron-carbide is longitudinally separated into individual compartments by stainless steel balls at approximately 16-in. intervals. The steel balls are held in place by a slight crimp of the tube. Should boron-carbide tend to ~~compact further~~ sinter in service, the steel balls will ~~distribute~~ keep the resulting void~~s~~ spaces distributed over the length of the absorber tube.

4.2.2.2 Velocity Limiter

The control rod velocity limiter (see Figure 4.2-3) is an integral part of the bottom assembly of each control rod. This engineered safeguard protects against a high reactivity insertion rate by limiting the control rod velocity in the event of a control-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.

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~~4.2.2.2 Velocity Limiter (Continued)~~

The velocity limiter is in the form of two nearly mated conical elements that act as a large clearance piston inside the control rod guide tube. The lower conical element is separated from the upper conical element by four radial spacers 90 degrees apart. ~~and is at a 15-degree angle relative to the upper conical element. with the peripheral separation less than the central separation.~~

The hydraulic drag forces on a control rod are approximately proportional to ~~approximately~~ the square of the rod velocity and are negligible at normal rod withdrawal or rod insertion speeds. However, during the scram stroke the rod reaches high velocity, and the drag forces must be overcome by the drive mechanism.

To limit control rod velocity during dropout but not during scram, the velocity limiter is provided with a streamlined profile in the scram (upward) direction. Thus, when the control rod is scrammed, water flows over the smooth surface of the upper conical element into the annulus between the guide tube and the limiter. In the dropout direction, however, water is trapped by the lower conical element and discharged through the annulus between the two conical sections. Because this water is forced ~~jetted~~ in a partially reversed direction into water flowing upward in the annulus, a severe turbulence is created, and this ~~thereby~~ slows ~~slowing~~ the descent of the control rod assembly to less than 5 ft/sec.

4.2.3 Design Evaluation - Caps

See Appendix A, Subsection A.4.2.3 of Reference ~~1~~ 4.2-1.

4.2.4 Testing, Inspection and Surveillance Plans - Caps

4.2.4.1 General Electric Fuel Testing, Inspection, and Surveillance

See Appendix A, Subsection A.4.2.4 of Reference ~~1~~ 4.2-1.

~~4.2.4 Testing, Inspection, and Surveillance Plans~~

~~4.2.4.1 Testing and Inspection of New Fuel.~~

~~Testing and inspection of new fuel both at the manufacturing facility and at WNP-2 are described in subsections 2.8.1, 2.8.2 and 2.8.3 of Reference 4.2-1. These subsections describe the testing and inspections during the fuel assembly manufacturing process along with enrichment control and burnable poison inspections. In subsection 2.8.1, a description is given of the sampling rate, method and tools of the post-shipment fuel inspection.~~

4.2.4.2 Online Fuel System Monitoring.

WNP-2 has two independent radiation detection systems that are directly capable of detecting fission product releases from failed fuel rods in an online manner. The main steam line radiation (MSLR) monitors actions after detecting gross fuel failures are described in ~~section 7.3 of the FSAR~~ 11.5.2.1.1. Because the MSLR monitors are locate relatively close to the reactor core, they are capable of sensing gross fission product releases in a few seconds.

The offgas system radiation (OGSR) monitors are capable of detecting low-level emissions of noble gases in 2 to 3 minutes after the gases leave the fuel. The OGSR monitors are described in more detail in ~~Section~~ 11.5.2.2.1 ~~of the FSAR~~.

Possible gross fuel failure is indicated by a reactor scram and main steam line valve closure initiated at three times full power background level by the MSLR monitors (see Technical Specification section 2.2; Limiting Safety System Instrumentation setpoints). Gradual fission particle release increase is addressed by Technical Specification 3/4.4.5, Specific Activity.

4.2.4.3 Post-Irradiation Surveillance

~~(see attached)~~

~~4.2.1.3 SURVEILLANCE~~

~~4.?.? Fuel Assembly~~

The following fuel surveillance will be conducted for the WNP-2 unit on fuel discharged during the shutdown period of each refueling outage.

Scope

The fuel surveillance program, developed to provide verification of the reliable performance of the WNP-2 fuel design, will consist of the following inspections and measurements:

a. Visual inspection of the peripheral rods will be performed on at least ~~the two~~ five to ten percent of the highest burnup assemblies of the discharged fuel during each refueling outage to verify assembly and rod structural integrity.

b. If anomalous behavior of the fuel cladding, components of the fuel assembly, or significant rod bow are detected by visual examination, further investigation and measurements of such significant anomalies will be conducted after the refueling outages.

Implementation

a. On-site receiving inspection of all the initial core fuel assemblies and subsequent reloads will be documented. Any significant anomalies detected will be documented. ~~and videotaped.~~

b. Fuel performance history and related plant operation data will be monitored and analyzed during operation.

c. Fuel assemblies scheduled to be permanently discharged will be evaluated prior to the refueling shutdown to identify the range of burnup and operating conditions based on core analysis simulation ~~and in-core~~ measurements.

d. At least ~~the two~~ five to ten percent of the highest burnup assemblies discharged during each refueling outage will be selected for visual inspection. The visual examination of the peripheral rods will include observations for cladding defects, fretting, rod bowing, missing components, corrosion, crud deposition, and geometric distortions. The surface area examined on the selected discharged fuel assemblies will be videotaped.

e. In the event that significant anomalies are observed during the refueling examination, all other discharged assemblies will also be visually inspected during the refueling outage. The results will be analyzed to determine fuel utilization strategy and possible safety implications in accordance with the operating procedures and applicable licensing requirements.

~~4.2.4.3.2 Fuel Assembly (Continued)~~

~~Implementation (Continued)~~

f. If unusual defects are observed, the fuel with the defects and the applicable operational data will be investigated and further appropriate tests and examination of the defected fuel will be performed.

g. Following inspections, an oral report of the results and conclusions will be made prior to startup. In addition, a written report will be submitted to the NRC within 90 days after the startup for the subsequent cycle. Under normal conditions, the report will contain visual examination summaries confirming the reliable performance of the fuel assemblies. In the event that significant anomalies or unusual defects are observed, the report will contain the description and related data of on-site receiving inspection and operational conditions. Evaluation and studies to identify causes for any encountered anomalies or defects will be assessed and the results will be reported to the NRC as they become available.

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← all caps

4.2.5 References

4.2-1 "General Electric Standard Application for Reactor Fuel," (NEDE-24011-P-A, latest approved revision).

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4.3 NUCLEAR DESIGN

See Appendix A, Section A.4.3 of Reference ~~1~~ 4.3-1.

4.3.1 Design Bases — Copo consistent with 4.1.1 (throughout)

See Appendix A, Subsection A.4.3.1 of Reference ~~1~~ 4.3-1.

4.3.2 Description — Copo

See Appendix A, Subsection A.4.3.2 of Reference ~~1~~ 4.3-1.

4.3.2.1 Nuclear Design Description

See Appendix A, Subsection A.4.3.2.1 of Reference ~~1~~ 4.3-1. The initial core loading pattern is provided in Figure 4.3-1. A summary of the fuel bundles loaded is given in Table 4.3-1.

4.3.2.2 Power Distribution

See Appendix A, Subsection A.4.3.2.2 of Reference ~~1~~ 4.3-1.

4.3.2.3 Reactivity Coefficients

See Appendix A, Subsection A.4.3.2.3 of Reference ~~1~~ 4.3-1.

4.3.2.4 Control Requirements

See Appendix A, Subsection A.4.3.2.4 of Reference ~~1~~ 4.3-1.

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4.3.2.4.1 Shutdown Reactivity

To assure that the safety design basis for shutdown is satisfied, an additional design margin is adopted: k-effective is calculated to be less than or equal to 0.99 with the control rod of highest worth fully withdrawn.

The cold shutdown margin for the initial core loading pattern is given in Table 4.3-2.

4.3.2.4.2 Reactivity Variations

The excess reactivity designed into the core is controlled by the control rod system supplemented by gadolinia-urania fuel rods. Enrichment distributions for these rods are given in ~~Appendix D~~ Section 2 of Reference ~~X~~ 4.3-1.

Control rods are used during the cycle partly to compensate for burnup and partly to flatten the power distribution.

Reactivity balances are not used in describing BWR behavior because of the strong interdependence of the individual constituents of reactivity. Therefore, the design process does not produce components of a reactivity balance at the conditions of interest. Instead, it gives the k_{eff} (Table 4.3-2) representing all effects combined. Further, any listing of components of a reactivity balance is quite ambiguous unless the sequence of the changes is clearly defined.

4.3.2.5 Control Rod Patterns and Reactivity Worths

See Appendix A, Subsection A.4.3.2.5 of Reference ~~X~~ 4.3-1.

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4.3.2.6 Criticality of Reactor During Refueling

See Appendix A, Subsection A.4.3.2.6 of Reference ~~1~~ 4.3-1.

4.3.2.7 Stability

See Appendix A, Subsection A.4.3.2.7 of Reference ~~1~~ 4.3-1.

4.3.2.8 Vessel Irradiations

The neutron fluxes at the vessel have been calculated using the one-dimensional discrete ordinates transport code described in ~~Subsection~~ 4.1.4.5. The discrete ordinates code was used in a distributed source mode with cylindrical geometry. The geometry described six regions varying from the center of the core to a point beyond the vessel. The core region was modeled as a single homogenized cylindrical region. The coolant water region between the fuel channel and the shroud was described containing saturated water at 550°F and 1050 psi. The material compositions for the stainless steel in the shroud and the carbon steel in the vessel contain the mixtures by weight as specified in the ASME material specifications for ASME SA-240, 304L, and ASME SA-533 grade B. In the region between the shroud and the vessel, the presence of the jet pumps was ignored. A simple diagram showing the regions, dimensions, and weight fractions are shown in Figure 4.3-2.

The distributed source used for this analysis was obtained from the gross radial power description. The distributed source at any point in the core is the product of the power from the power description and the neutron yield from fission. By using the neutron energy spectrum, the distributed source is obtained for position and energy. The integral over position and energy is normalized to the total number of neutrons in the core region. The core region is defined as a 1 centimeter thick disc with no transverse leakage. The power in this core region is set equal to the maximum power in the axial direction.

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4.3.2.8 ~~Vessel Irradiations (Continued)~~

The neutron fluence is determined from the calculated flux by assuming that the plant is operated 90 percent of the time at 90 percent power level for 40 years or equivalent to 1×10^9 full power seconds. The calculated fluxes and fluence are shown in Table 4.3-3. The calculated neutron flux leaving the cylindrical core is shown in Table 4.3-4.

4.3.3 Analytical Methods cap

See Appendix A, Subsection A.4.3.3 of Reference ~~1~~ 4.3-1.

4.3.4 Changes cap

See Appendix A, Subsection A.4.3.4 of Reference ~~1~~ 4.3-1.

4.3.5 References cap

~~1.~~ 4.3-1 "General Electric Standard Application for Reactor Fuel," (NEDE-24011-P-A, latest approved revision).

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Table 4.3-1

INITIAL CORE FUEL BUNDLES

Fuel Designation	Number Loaded
P8CRB219	432
P8CRB176	240
P8CRB071	92

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Table 4.3-2

CALCULATED CORE EFFECTIVE MULTIPLICATION AND CONTROL SYSTEM WORTH - NO VOIDS, 20°C

Beginning of Cycle, K-effective	
Uncontrolled	1.1165
Fully Controlled	~~0.9328~~ 0.9302
Strongest Control Rod Out	~~0.9758~~ 0.9710
R, Maximum Increase in Cold Core Reactivity with Cycle Exposure, ΔK	0.0047

TABLE 4.3-3

CALCULATED NEUTRON FLUXES (USED TO EVALUATE VESSEL IRRADIATION)

NEUTRON ENERGY MeV	AVERAGE FLUX IN THE CORE n/cm^2-sec	FLUX AT THE CORE BOUNDARY n/cm^2-sec	FLUX AT THE INSIDE SURFACE VESSEL n/cm^2-sec
>3.0	1.5×10^{13}	5.4×10^{12}	3.3×10^{8}
1.0 - 3.0	3.3×10^{13}	1.2×10^{13}	2.8×10^{8}
0.1 - 1.0	5.3×10^{13}	1.7×10^{13}	4.9×10^{8}

Maximum Fluence >1.0 MeV = 1.4×10^{18} $\frac{n}{cm^2}$ 2

1. The calculated flux is a maximum in the axial direction by taking the average over the azimuthal angle.

2. The maximum fluence is calculated using the flux and a capacity factor of 80% or 1×10^9 full power seconds. The fluence includes a azimuthal peaking factor and a factor to cover analytical uncertainties. an

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TABLE 4.3-4

CALCULATED NEUTRON FLUX AT CORE EQUIVALENT BOUNDARY

GROUP	LOWER ENERGY BOUND eV	FLUX n/cm^2-Sec
1	10.0×10^6	4.6×10^{10}
2	6.065×10^6	6.1×10^{11}
3	3.679×10^6	2.1×10^{12}
4	2.231×10^6	4.2×10^{12}
5	1.353×10^6	4.4×10^{12}
6	8.208×10^5	3.9×10^{12}
7	4.979×10^5	4.0×10^{12}
8	3.020×10^5	2.8×10^{12}
9	1.832×10^5	2.3×10^{12}
10	1.111×10^5	1.8×10^{12}
11	6.732×10^4	1.4×10^{12}
12	4.087×10^4	1.1×10^{12}
13	2.478×10^4	1.0×10^{12}
14	1.503×10^4	1.0×10^{12}
15	9.119×10^3	9.6×10^{11}
16	5.531×10^3	9.4×10^{11}
17	3.355×10^3	9.4×10^{11}
18	2.034×10^3	9.1×10^{11}
19	1.010×10^3	1.3×10^{12}
20	2.492×10^2	2.5×10^{12}
21	5.560×10^1	2.6×10^{12}
22	1.240×10^1	2.5×10^{12}
23	0.625	4.0×10^{12}
24	0.0	2.5×10^{13}

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4.4 THERMAL-HYDRAULIC DESIGN

See Appendix A, Section A.4.4 of Reference ~~1~~ 4.4-1.

4.4.1 Design Basis - caps

See Appendix A, Subsection A.4.4.1 of Reference ~~1~~ 4.4-1.

4.4.1.1 Safety Design Bases

See Appendix A, Subsection A.4.4.1.1 of Reference ~~1~~ 4.4-1.

4.4.1.2 Power Generation Design Bases

See Appendix A, Subsection A.4.4.1.2 of Reference ~~1~~ 4.4-1.

4.4.1.3 Requirements for Steady-State Conditions

For purposes of maintaining adequate thermal margin during normal steady-state operation, the MCPR must not be less than the required MCPR operating limit, and the MLHGR must be maintained below the design LHGR for the plant. This does not specify the operating power nor does it specify peaking factors. These parameters are determined subject to a number of constraints including the thermal limits given previously. The core and fuel design basis for steady-state operation (i.e., MCPR and LHGR limits) have been defined to provide margin between the steady-state operating conditions and any fuel damage condition to accommodate uncertainties and to assure that no fuel damage results even during the worst anticipated transient condition at any time in life. The design steady-state MCPR operating limit and the peak LHGR are given in Table 4.4-1.

4.4.1.4 Requirements for Transient Conditions

See Appendix A, Subsection A.4.4.1.4 of Reference ~~1~~ 4.4-1.

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4.4.1.5 Summary of Design Bases

See Appendix A, Subsection A.4.4.1.5 of Reference ~~1~~ 4.4-1.

4.4.2 Description of Thermal-Hydraulic Design of Reactor Core -Copy

See Appendix A, Subsection A.4.4.2 of Reference ~~1~~ 4.4-1.

4.4.2.1 Summary Comparison

An evaluation of plant performance from a thermal and hydraulic standpoint is provided in ~~Subsection~~ 4.4.3.

A tabulation of thermal and hydraulic parameters of the core is given in Table 4.4-1.

4.4.2.2 Critical Power Ratio

See Appendix A, Subsection A.4.4.2.2 of Reference ~~1~~ 4.4-1.

4.4.2.3 Linear Heat Generation Rate (LHGR)

See Appendix A, Subsection A.4.4.2.3 of Reference ~~1~~ 4.4-1.

4.4.2.4 Void Fraction Distribution

The core average and maximum exit void fractions in the core at rated condition are given in Table 4.4-1. The axial distribution of core void fractions for the average radial channel and the maximum radial channel (end of node value) for the core are given in Table 4.4-2. The core average and maximum exit values ~~is~~ are also provided. Similar distributions for steam quality are provided in Table 4.4-3. The core average axial power distribution used to produce these tables is given in Table 4.4-4.

4.4.2.5 Core Coolant Flow Distribution and Orificing Pattern

See Appendix A, Subsection A.4.4.2.5 of Reference ~~1~~ 4.4-1.

4.4.2.6 Core Pressure Drop and Hydraulic Loads

See Appendix A, Subsection A.4.4.2.6 of Reference ~~1~~ 4.4-1.

4.4.2.7 Correlation and Physical Data

See Appendix A, Subsection A.4.4.2.7 of Reference ~~1~~ 4.4-1.

4.4.2.8 Thermal Effects of Operational Transients

See Appendix A, Subsection A.4.4.2.8 of Reference ~~1~~ 4.4-1.

4.4.2.9 Uncertainties in Estimates

See Appendix A, Subsection A.4.4.2.9 of Reference ~~1~~ 4.4-1.

4.4.2.10 Flux Tilt Considerations

See Appendix A, Subsection A.4.4.2.10 of Reference ~~1~~ 4.4-1.

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

See Appendix A, Subsection A.4.4.3 of Reference ~~1~~ 4.4-1.

4.4.3.1 Plant Configuration Data

4.4.3.1.1 Reactor Coolant System Configuration

The reactor coolant system is described in Section 5.4 and shown in isometric perspective in Figure 5.4-1. The piping sizes, fittings and valves are listed in Table 5.4-1.

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4.4.3.1.2 Reactor Coolant System Thermal Hydraulic Data

The steady-state distribution of temperature, pressure and flow rate for each flow path in the reactor coolant system is shown in Figure 5.1-1.

4.4.3.1.3 Reactor Coolant System Geometric Data

Coolant volumes of regions and components within the reactor vessel are shown in Figure 5.1-2.

Table 4.4-5 provides the flow path length, height, liquid level, minimum elevations, and minimum flow areas for each major flow path volume within the reactor vessel and recirculation loops of the reactor coolant systems.

Table 4.4-6 provides the lengths and sizes of all safety injection lines to the reactor coolant system.

4.4.3.2 Operating Restrictions on Pumps

5.4-3a and 5.4-3b

Expected recirculation pump performance curves are shown in Figures 5.4-3. These curves are valid for all conditions with a normal operating range varying from approximately 20 percent to 115 percent of rated pump flow.

The pump characteristics, including considerations of NPSH requirements, are the same for the conditions of two-pump and one-pump operation as described in ~~Subsection~~ 5.4.1. Subsection 4.4.3.3 gives the operating limits imposed on the recirculation pumps by cavitation, pump loads, bearing design flow starvation, and pump speed.

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4.4.3.3 Power-Flow Operating Map

4.4.3.3.1 Limits for Normal Operation

A boiling water reactor must operate with certain restrictions because of pump net positive suction head (NPSH), overall plant control characteristics, core thermal power limits, etc. The power-flow operating map for the power range of operation is shown in Figure 4.4-1. The nuclear system equipment, nuclear instrumentation, and the reactor protection system, in conjunction with operating procedures, maintain normal operations within the area of this map. ~~for normal operating conditions.~~ The boundaries on this map are as follows:

<u>Natural Circulation Line</u>: The operating state of the reactor moves along this line for the normal control rod withdrawal sequence in the absence of recirculation pump operation.

<u>Rated Flow Control Line</u>: This line passes through 100% power at 100% flow. The operating state for the reactor follows this line (or similar ones) during recirculation flow changes with a fixed control rod pattern. The line is based on a constant xenon concentration at rated power and flow.

<u>Constant Position Lines for Flow Control Valve (FCV)</u>: These lines show the change in flow associated with power changes while maintaining flow-control valves at a constant position.

<u>Cavitation Protection Line</u>: This line results from the recirculation pump, flow control valve, and jet pump NPSH requirements.

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4.4.3.3.2 Regions of the Power Flow Map

Region I — This region defines the system operational capability with the recirculation pumps and motors being driven by the low frequency motor-generator set at 25% speed. Flow is controlled by the flow control valve and power changes during normal startup and shutdown will be in this region. The normal operating procedure is to start up along curve C - FCV wide open at 25% speed.

Region II — This region shows the area where the 25% pump speed and 100% pump speed operating regimes overlap. The switching sequence from the low frequency m-g set to 100% speed will be done in this region.

Region III — This is the low power area of the operating map where cavitation can be expected in the recirculation pumps, jet pumps, or flow control valves. Operation within this region is precluded by system interlocks which trip the main motor from the 100% speed power source to the 25% speed power source.

Region IV — This represents the normal operating zone of the map where power changes can be made by either control rod movement or by core flow changes through use of the flow control valve.

4.4.3.3.3 Design Features for Power-Flow Control

The following limits and design features are employed to maintain power-flow conditions to the required values shown in Figure 4.4-1.

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4.4.3.3.3 Design Features for Power-Flow Control (Continued)

a. Minimum power limits at intermediate and high core flows. To prevent cavitation in the recirculation pumps, jet pumps and flow control valves, the recirculation system is provided with an interlock to trip off the 60 Hz power source and close the 15Hz power source if the difference between steam line temperature and recirculation pump inlet temperature is less than a preset value (10.7°F). This differential temperature is measured using high high accuracy RTDs with a sensing error of less than 0.2°F at the two standard deviation (2σ) confidence level. This action is initiated electronically through a 15-second time delay. The interlock is active while in both the automatic and manual operation modes.

b. Minimum power limit at low core flow. During low power, low loop flow operations, the temperature differential interlock may not provide sufficient cavitation protection to the flow control valves. Therefore, the system is provided with an interlock to trip off the 60Hz power source and close the 15Hz power source if the feedwater flow falls below a preset level (28% of rated) and the flow control valves are below a preset position (19% open). The feedwater flow rate and recirculation flow control valve position are measured by existing process control instruments. The speed change action is electronically initiated. This interlock is active during both automatic and manual modes of operation.

c. Pump bearing limit. For pumps as large as the recirculation pumps, practical limits of pump bearing design

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4.4.3.3.3 Design Features for Power-Flow Control (Continued)

require that minimum pump flow be limited to 20% of rated. To assure this minimum flow, the system is designed so that the minimum flow control valve position will allow this rate of flow.

d. Valve position. To prevent structural or cavitation damage to the recirculation pump due to pump suction flow starvation, the system is provided with an interlock to prevent starting the pumps, or to trip the pumps if the suction or discharge block valves are at less than 90% open position. This circuit is activated by a position limit switch and is active before the pump is started, during manual operation mode and during automatic operation mode.

4.4.3.3.3.1 Flow Control

The principal modes of normal operation with valve flow control low frequency motor generator (LFMG) set are summarized as follows. The recirculation pumps are started on 100% speed power source in order to unseat the pump bearings. Suction and discharge block valves are full open and the flow control valve is in the minimum position. When the pump is at full speed, the main power source is tripped and the pump allowed to coast down to 25% speed where the LFMG set will power the pump and motor. The flow control valve is then opened to the maximum position at which point reactor heatup and pressurization can commence. When operating pressure has been established, reactor power can be increased. This power-flow increase will follow a line within Region I of the flow control map shown in Figure 4.4-1.

When reactor power is greater than approximately 30% of rated, the low feedwater flow interlock is cleared and the main recirculation

4.4.3.3.1 Flow Control (Continued)

pumps can be switched to the 60Hz power source. The flow control valve is closed to the minimum position before the speed change to prevent large increases in core power and a potential flux scram. This operation occurs within Region II of the operating map. The system is then brought to the desired power-flow level within the normal operating area of the map (Region IV) by opening the flow control valves and by withdrawing control rods.

Control rod withdrawal with constant flow control valve position will result in power/flow changes along lines of constant C_V (constant position). Flow control valve movement with constant control rod position will result in power/flow changes along, or nearly parallel to, the rated flow control line, CF.

4.4.3.4 Temperature-Power Operating Map (PWR)

Not applicable.

4.4.3.5 Load-Following Characteristics

Large negative operating reactivity coefficients, inherent in the boiling water reactor, provide the following important advantages:

a. Good load-following with well damped behavior and little undershoot or overshoot in the heat transfer response,

b. Load-following with recirculation flow control, and

c. Strong damping of spatial power disturbances.

Design of the boiling water reactor includes the ability to follow load demands over a reasonable range without requiring operator

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~~4.4.3.5 Load-Following Characteristics (Continued)~~

action. Reactor power can be controlled automatically by flow control over approximately a 35% power range at, for example, approximately 1/2% per second for a 10% step-load change.

4.4.3.6 Thermal and Hydraulic Characteristics Summary Table

The thermal-hydraulic characteristics are provided in Table 4.4-1 for the core and tables of Section 5.4 for other portions of the reactor coolant system.

4.4.4 Evaluation - Cop

See Appendix A, Subsection A.4.4.4 of Reference ~~1~~ 4.4-1.

4.4.5 Testing and Verification - cop

See Appendix A, Subsection A.4.4.5 of Reference ~~1~~ 4.4-1.

4.4.6 Instrumentation Requirements - Cop

See Appendix A, Subsection A.4.4.6 of Reference ~~1~~ 4.4-1.

4.4.6.1 Loose Parts

The loose part detection sensors are mounted on the exterior of primary coolant system and are located at natural collection points where loose parts will be most likely to impact. The general locations are:

a. Main Steam Line A & B (25 26" line): 2 sensors.

b. Feedwater line A & B (12" line): 2 sensors.

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~~4.4.6.1 Loose Parts (Continued)~~

c. Recirculation Water Outlet A & B (24" line):
2 sensors

d. Reactor Vessel Bottom Head (3/4" to 1" CRD lines):
4 sensors

See 7.7.1.12 for further information.

4.4.7 References - cap

4.4-1 ~~1.~~ "General Electric Standard Application for Reactor Fuel," (NEDE-24011, latest approved revision).

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THERMAL AND HYDRAULIC DESIGN CHARACTERISTICS OF THE REACTOR CORE

General Operating Conditions	
Reference design thermal output, Mwt	3,323 ~~8~~
Power level for engineered safety features, Mwt	3,489 ~~8~~
Steam flow rate, at 420°F final feedwater temperature millions lb/hr	14.30
Core coolant flow rate, millions lb/hr	108.5
Feedwater flow rate, millions lb/hr	14.26
System pressure, nominal in steam dome, psia	1020 ~~8~~
System pressure, nominal core design, psia	1035 ~~8~~
Coolant saturation temperature at core design pressure, $^{\circ}$F	549
Average power density, kW/liter	49.15
Maximum Linear Heat Generation Rate kW/ft	13.4
Average Linear Heat Generation Rate kW/ft	5.4
Core total heat transfer area, ft^2	74,871
Maximum heat flux, Btu/hr-sq ft	361,500

TABLE 4.4-1 (Continued)

<u>General Operating Conditions</u>

Average heat flux, Btu/hr-sq ft	145,100
Design operating minimum critical power ratio (MCPR)	1.24 (see Table 15.0-3)
Core inlet enthalpy at 420°F FFWT, Btu/lb	527.6
Core inlet temperature, at 420°F FFWT, °F	533
Core maximum exit voids within assemblies, %	0.76 ~~0~~
Core average void fraction, active coolant	0.418
Maximum fuel temperature, °F	3,435 ~~0~~
Active coolant flow area per assembly, in.^{2} (BOL)	15.824
Core average inlet velocity, ft/sec	6.88
Maximum inlet velocity, ft/sec	7.28
Total core pressure drop, psi	24.74
Core support plate pressure drop, psi	20.32
Average orifice pressure drop	
Central region, psi	6.03
Peripheral region, psi	16.54
Maximum channel pressure loading, psi	13.28

TABLE 4.4-2

VOID DISTRIBUTION

Core Average Value = 0.409
Maximum Exit Value = 0.760
Active Fuel Length = 150 inches

	Node	Core Average (Average Node Value)	Maximum Channel (End of Node Value)
Bottom of Core	1	0.000	0.000
	2	0.000	0.007
	3	0.007	0.073
	4	0.041	0.183
	5	0.104	0.287
	6	0.179	0.371
	7	0.254	0.443
	8	0.323	0.500
	9	0.379	0.545
	10	0.425	0.582
	11	0.462	0.611
	12	0.492	0.636
	13	0.517	0.655
	14	0.537	0.672
	15	0.555	0.686
	16	0.571	0.699
	17	0.585	0.711
	18	0.598	0.722
	19	0.610	0.733
	20	0.622	0.742
	21	0.631	0.750
	22	0.638	0.756
	23	0.644	0.759
Top of Core	24	0.646	0.760

TABLE 4.4-3

FLOW QUALITY DISTRIBUTION

Core Average Value = 0.072
Maximum Exit Value = 0.282
Active Fuel Length = 150 inches

	Node	Core Average (Average Node Value)	Maximum Channel (End of Node Value)
Bottom of Core	1	0.000	0.000
	2	0.000	0.000
	3	0.000	0.003
	4	0.001	0.012
	5	0.004	0.026
	6	0.010	0.042
	7	0.019	0.061
	8	0.029	0.081
	9	0.040	0.100
	10	0.051	0.119
	11	0.062	0.137
	12	0.072	0.153
	13	0.080	0.169
	14	0.089	0.182
	15	0.097	0.196
	16	0.105	0.208
	17	0.112	0.221
	18	0.119	0.233
	19	0.126	0.246
	20	0.134	0.258
	21	0.140	0.268
	22	0.145	0.275
	23	0.149	0.280
Top of Core	24	0.151	0.282

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TABLE 4.4-4

AXIAL POWER DISTRIBUTION USED TO GENERATE VOID AND QUALITY DISTRIBUTIONS

	NODE	AXIAL POWER FACTOR
Bottom of Core	1	0.38
	2	0.69
	3	0.93
	4	1.10
	5	1.21
	6	1.30
	7	1.47
	8	1.51
	9	1.49
	10	1.44
	11	1.36
	12	1.28
	13	1.16
	14	1.06
	15	1.01
	16	0.97
	17	0.94
	18	0.97
	19	0.96
	20	0.91
	21	0.77
	22	0.59
	23	0.38
Top of Core	24	0.12

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TABLE 4.4-5

REACTOR COOLANT SYSTEM GEOMETRIC DATA

	Flow Path Length (in.)	Height and Liquid Level (in.)	Elevation of Bottom of Each Volume* (in.)	Minimum Flow Areas (sq ft)
A. Lower Plenum	216	216 216	 -172.5	71.5
B. Core	164	164 164	44.0	142.0
C. Upper Plenum and Separators	178	178 178	208.0	49.5
D. Dome (Above Normal Water Level)	312	312 0	386.0	343.5
E. Downcomer Area	321	321 321	 -51.0	 79.5
F. Recirculation Loops and Jet Pumps (one loop)	108.5 ft.	403 403	-394.5	132.5 in^2

*Reference point is recirculation nozzle outlet centerline.

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~~AL to confirm table~~

TABLE 4.4-6

LENGTHS AND SIZES OF SAFETY INJECTION LINES

		LINE O.D. (in.)		LINE Length (ft)
I. HPCS Line				
	A. Pump discharge to valve*	16"		331'-5 15/16"
	B. Inside containment to RPV	12 3/4"		104'-0 1/4"
			TOTAL	435'-6 3/16"
II. LPCI Lines				
	A. LOOP A			
		18"		352'-0 1/8"
	1. Pump discharge to valve*	14"		23'-0 ½"
	2. Inside containment to RPV	14"		91'-2"
			TOTAL	466'-2 5/8"
	B. LOOP B			
		18"		339'-5 5/16"
	1. Pump discharge to valve*	14"		19'-7 3/4"
	2. Inside containment to RPV	14"		91'-1 9/16"
			TOTAL	450'-2 5/8"
	C. LOOP C			
		18"		71'-4 3/8"
	1. Pump discharge to valve*	14"		139'-0 4/16"
	2. Inside containment to RPV	14"		91'-1 9/16"
			TOTAL	301'-6 3/16"
III. LPCS Line				
	A. Pump discharge to valve*	16"		225'-1 5/8"
	B. Inside containment to RPV	12 3/4"		106'-4 7/8"
			TOTAL	331'-6 1/2"

*Valve located as near as possible to outside of containment wall.

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Table 4.4-7

STABILITY ANALYSIS RESULTS

Rod Line Analyzed: 105%

Decay Ratio:

Total System Stability	1.0
Reactor Core Stability	0.702
Channel Hydrodynamic Performance	0.494

CHAPTER 4.0

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