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III - REACTOR

1.0 SUMMARY DESCRIPTION

The sections included in the "Reactor" section describe and evaluate those systems most pertinent to the fuel barrier and the control of core reactivity. The "Fuel Mechanical Design" section describes the mechanical aspects of the fuel material (uranium dioxide), the fuel cladding, the fuel rods, and the arrangement of fuel rods in bundles. Of particular interest is the ability of the fuel to serve as the initial barrier to the release of radioactive material. The mechanical design of the fuel is sufficient to prevent the escape of significant amounts of radioactive material during normal modes of reactor operation.

The "Reactor Vessel Internals Mechanical Design" section describes both the arrangements of the supporting structure for the core and the reactor vessel internal components which are provided to properly distribute the coolant delivered to the reactor vessel. In addition to their main function of coolant distribution, the reactor vessel internals separate the moisture from the steam leaving the vessel and provide a floodable inner volume inside the reactor vessel that allows sufficient submergence of the core under accident conditions to prevent additional damage to the fuel and the gross release of fission products from the fuel. The reactor vessel internals are designed to allow the control rods and Core Standby Cooling Systems to perform their safety functions during abnormal operational transients and accidents.

The "Reactivity Control Mechanical Design" section describes the mechanical aspects of the moveable control rods. These are provided to control core reactivity. The Control Rod Drive Hydraulic System is designed so that sufficient energy is available to force the control rods into the core under conditions associated with abnormal operational transients and accidents. Control rod insertion speed is sufficient to prevent fuel damage as a result of any abnormal operational transient.

Control Rod Housing Supports are located underneath the reactor vessel near the control rod housings. These supports limit the travel of a control rod in the event that a control rod housing is ruptured. They prevent a significant nuclear excursion as a result of the housing failure, thus protecting the fuel barrier and the primary system.

The "Nuclear Design" section describes the nuclear aspects of the reactor core. The design of the boiling water reactor 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, flow distribution, void content, bundle power, and operating pressure. All of these conditions are dynamic functions of operating conditions. However, design analyses and calculations, verified by comparison with data from operating plants, are performed for specific steady state, transient, and accident conditions. Included in the "Nuclear Design" section are summaries of results of the steady-state analyses for the fuel cycle, reactivity control, and control rod worths. Also included are discussions of the reactivity coefficients and spatial xenon characteristics of the core. Transient and accident analysis results are presented in Section XIV, "Station Safety Analysis".

The Thermal – Hydraulic stability analysis provides a discussion on regional and core wide thermal – hydraulic instabilities; thermal – hydraulic/reactivity feedback mechanisms and the core response to such instabilities. Evaluations performed reveal core wide versus regional instabilities are unlikely. For core wide instabilities APRM flow bias scram will prevent exceeding safety limits. This analysis of the nuclear system stability demonstrates the reactor can be operated safely without danger of compromising any radioactive material barriers or fuel safety limits because of instability.

The "Thermal and Hydraulic Design" section describes the thermal and hydraulic characteristics of the core. The low coolant saturation temperature, high heat transfer coefficient, and neutral water chemistry of the boiling water reactor are significant advantages in minimizing Zircaloy temperatures and associated temperature-dependent hydride pickup. This results in improved cladding performance at long exposures. The relatively uniform fuel cladding temperatures throughout the boiling water reactor core minimize migration of the hydrides to cold cladding zones and reduce the thermal stresses. A discussion of fuel failure mechanisms and the parameters associated with fuel damage is included in the section.

The Standby Liquid Control System provides a redundant, independent, and different way from the control rods to make the reactor subcritical, even in the cold condition. The insertion of control rods is expected always to assure prompt shutdown of the reactor. The Standby Liquid Control System can bring the reactor, at any time in a cycle, from full power and minimum control rod inventory (which is defined to be at the peak of the xenon transient) to a subcritical condition with the reactor in the most reactive xenon-free state.

2.0 FUEL MECHANICAL DESIGN

2.1 Safety Objective

The Safety Objective of the nuclear fuel is to ensure, in conjunction with the core nuclear characteristics, the core thermal and hydraulic characteristics, the plant equipment characteristics, and the instrumentation and protection systems, that fuel damage will not result in the release of radioactive materials in excess of the requirements of 10CFR20, 50, or 100.

2.2 Safety Design Basis

In meeting the power generation objectives, the nuclear fuel is utilized as the initial barrier to the release of fission products. The fission product retention capability of the nuclear fuel is designed to assure (in conjunction with the core nuclear characteristics, the core thermal and hydraulic characteristics, the plant equipment characteristics, and the capability of the nuclear instrumentation and reactor protection system) that fuel damage limits will not be exceeded during normal modes of reactor operation and abnormal operational occurrences.

2.3 Power Generation Objective

The objective of the nuclear fuel is to provide the heat source to the reactor coolant in producing steam for direct use in the turbine generator.

2.4 Power Generation Design Basis

The nuclear fuel shall provide a high integrity assembly of fissionable material, which can be arranged in a critical array. The assembly must be capable of efficiently transferring the generated fission heat to the circulating coolant water while maintaining structural integrity containing the fission products.

2.5 Description

The reactor core is comprised of numerous core cells. A typical core cell consists of a control rod and four fuel assemblies which immediately surround it (see Figure III-2-4). The four fuel assemblies are lowered into the core cell and, when seated, springs mounted at the tops of the channels force the channels into the corners of the cell such that the sides of the channels contact the grid beams (see Figure III-2-4).[1] Each core cell is associated with a four-lobed fuel support piece. Around the outer edge of the core, certain core cells have only three fuel assemblies and a "fuel support plug". This fuel support plug is a "dummy" assembly and contains no fuel. The fuel support plugs act as one-quarter of the control rod channel which is needed to guide the control rod during control rod movement. Also around the outer edge of the core, certain fuel assemblies are not immediately adjacent to a control rod and are supported by individual peripheral fuel support pieces.

The fuel assembly, shown in Figure III-2-1, consists of a fuel bundle and a channel which surrounds it. The current reload fuel bundle design is designated as GNF2. The GNF2 fuel bundles contain 92 fuel rods and two large centrally located water rods. The rods are spaced and supported in a square (10x10) array by the upper and lower tieplates and eight spacers. The lower tieplate has a nose piece which has the function of supporting the fuel assembly in the reactor. The upper tieplate has a handle for transferring the fuel bundle from one location to another. The identifying assembly serial number is engraved on the top of the handle. No two assemblies bear the same serial number. A boss projects from one side of the handle to aid in ensuring proper fuel assembly orientation. Both upper and lower tieplates are fabricated from Type-304 stainless steel castings. Inconel X-750 fuel rod spacers equipped with Inconel X-750 springs are employed to maintain rod-to-rod spacing in the GNF2 bundle. Finger springs located between the lower tieplate and the channel are utilized to control the bypass flow through that flow path in the GE14 dummy bundles; these were removed in the GNF2 bundle.[1] Fuel channels are made with both NSF and improved Zircaloy-2 materials, instead of Zircaloy-4, for improved resistance to corrosion in a steam environment. Zircaloy-2 has a thinner corrosion layer which: (1) lowers hydrogen pickup; (2) improves margin to thermal limits; (3) reduces risk of plant radiation buildup and worker exposure and; (4) provides greater resistance to damage from handling fuel and seismic events while in cold shutdown. NSF channel material has similar corrosion resistance as Zircaloy-2, but exhibits less channel bow which decreases the risk of control blade to channel interference.

Each fuel rod placed in a batch of fuel consists of high density ceramic uranium dioxide fuel pellets stacked within Zircaloy-2 cladding which is evacuated, backfilled with helium and sealed with Zircaloy end plugs welded in each end. GNF2 cladding tubes have an inner liner of high purity zirconium which minimizes the effect of interactions between the cladding and the pellets. This is referred to as "barrier fuel". The fuel rod cladding thickness is adequate to be "free-standing", i.e., capable of withstanding external reactor pressure without collapsing onto the pellets within. Although most fission products are retained within the UO_2, a fraction of the gaseous products is released from the pellet and accumulates in a plenum at the top of the rod. Sufficient plenum volume is provided to prevent excessive internal pressure from these fission gases or other gases liberated over the design life of the fuel. A plenum spring, or retainer, is provided in the plenum space to exert a downward pressure on the pellets (see detail of Figure III-2-1).

A complete description of the current approved fuel bundle designs is found in GE Nuclear Energy, NEDE 31152 P, Fuel Bundle Designs.[112]

Three types of fuel rods are used in a reload fuel bundle: part-length rods, tie rods and standard rods (Figure III-2-1). The eight tie rods in each bundle have lower end plugs which thread into the lower tie plate casting and threaded upper end plugs which extend through the upper tie plate casting. A stainless steel hexagonal nut and locking tab are installed on the upper end plug to hold the fuel bundle together. These tie rods support the weight of the bundle only during fuel handling operations when the assembly hangs by the handle. During operation, the fuel assembly is supported by the lower tieplate. Seventy rods in the GNF2 bundles are standard rods. The end plugs of the standard rods have shanks which fit into bosses in the tie plates. An Inconel-X expansion spring is located over the upper end plug shank of each rod in the assembly to keep the rods seated in the lower tie plate while allowing independent axial expansion by sliding within the holes of the upper tie plate. There are fourteen part-length rods. Part-length rods are dispersed throughout the bundle lattice.

The GNF2 fuel bundles contain two water rods. Each water rod is a hollow Zircaloy tube with several holes punched around the circumference near each end to allow coolant to flow through. The water rod is not of uniform diameter.

One water rod positions the eight Inconel X-750 fuel spacers axially. This spacer-positioning water rod is equipped with a square bottom end plug and with tabs which are welded to the exterior. The rod and spacers are assembled by sliding the rod through the appropriate spacer cell with the welded tabs oriented in the direction of the corner of the spacer cell. It is then rotated so that the tabs are above and below the spacer structure. Once in position the water rod is prevented from rotating by the engagement of its square lower end plug with the lower tie plate hole.

The primary function of the fuel spacer is to provide lateral support and spacing of the fuel rods, with consideration of thermal-hydraulic performance, fretting wear, strength, neutron economy, and producibility. The spacer represents an optimization of all these considerations.

Finger springs are employed to control the bypass flow through the channel-to-lower tieplate flow path for the dummy fuel assemblies. These springs have been used in all initial core and some reload fuel of all BWR/4 plants. These finger spring seals, located between the lower tie plate and the channel, provide control over the flow through this path due to channel wall deflections by maintaining a nearly constant flow area as the channel wall deforms.

The upper and lower tieplates serve the functions of supporting the weight of the fuel and positioning the rod ends during all phases of operation and handling. Both the upper and lower tieplates are shown in Figure III-2-1. All BWR/4 reactors have two alternate path bypass flow holes located in the lower tieplate (see Figure III-2-1). These holes are drilled to augment flow in the bypass region.[1]

The fuel pellets are manufactured by compacting and sintering uranium dioxide powder into right cylindrical pellets with flat ends and chamfered edges. Ceramic uranium dioxide is chemically inert to the cladding at operating temperatures and is resistant to attack by water. Several U-235 enrichments are used in the fuel assemblies to reduce the local peak-to-average fuel rod power ratios. Selected fuel rods within each reload bundle also incorporate small amounts of gadolinium as burnable poison. Gd_2O_3 is uniformly distributed in the UO_2 pellet and forms a solid solution.[5]

A separate licensing topical report entitled "BWR Fuel Channel Mechanical Design and Deflection," NEDE-21354-P (PROPRIETARY) and NEDO-21354, September 1976, provides a complete description and analytical results for channels supplied by the General Electric Company used in conjunction with the reload fuel. The following functional description is included for completeness.

The fuel channel performs the following functions:

a. forms the fuel bundle flow path outer periphery for bundle coolant flow;

b. provides surfaces for control rod guidance in the reactor core;

c. provides structural stiffness to the fuel bundle during lateral loading applied from fuel rods through the fuel spacers;

d. in conjunction with the fingersprings (if present) and bundle lower tieplate, controls coolant flow leakage at the channel/lower tieplate interface;

e. transmits fuel assembly seismic loading to the top guide and fuel support of the core internal structures;

f. provides a heat sink during loss-of-coolant accident (LOCA); and

g. provides a stagnation envelope for in-core fuel sipping.

The channel is open at the bottom and makes a sliding seal fit on the lower tieplate surface. The upper end of the fuel assemblies in a four-bundle cell are positioned in the corners of the cell against the top guide beams by the channel fastener springs (Figure III-2-4). At the top of the channel, two diagonally opposite corners have welded tabs, one of which supports the weight of the channel from a threaded raised post on the upper tieplate. One of these raised posts has a threaded hole. The channel is attached using the threaded channel fastener assembly, which also includes the fuel assembly positioning spring. Channel-to-channel spacing is provided for by means of spacer buttons located on the upper portion of the channel adjacent to the control rod passage area.

The GE14 fuel channel enclosing the fuel bundle has a square cross section with a 5.278-in. (nominal) inside width and round corners, each having a 0.45-in. (nominal) inside radius. The nominal length of the fuel channel is 166.906 in.[6]

The GNF2 fuel channel enclosing the fuel bundle has a square cross section with a 5.284-in. (nominal) inside width and round corners, each having a 0.45-in. (nominal) inside radius. The maximum length of the fuel channel is 167.58 in.

Proper orientation of fuel assemblies in the reactor core is readily verified by visual observation and is assured by verification procedures during core loading. Five separate visual indications of proper fuel assembly orientation exist:

a. The channel fastener assemblies, including the spring and guard used to maintain clearances between channels, are located at one corner of each fuel assembly adjacent to the center of the control rod.

b. The identification boss on the fuel assembly handle points toward the adjacent control rod.

c. The channel spacing buttons are adjacent to the control rod blades.

d. The assembly identification numbers on the fuel assembly handles are all readable from the direction of the center of the cell.

e. There is cell-to-cell symmetry.

Experience has demonstrated that these design features are clearly visible so that any misoriented fuel assembly would be readily distinguished during core loading verification.

2.6 Safety Evaluation

This USAR section contains historical information as indicated by the italicized text. USAR Section I-3.4 provides a more detailed discussion of historical information. The information being presented in this section as historical has been preserved as it was originally submitted to the NRC in the CNS FSAR.

Fuel damage is defined as perforation of the fuel rod cladding which would permit the release of fission products to the reactor coolant. The mechanisms which could cause fuel damage in reactor operational transients are: (a) severe overheating of the fuel rod cladding caused by inadequate cooling, (b) rupture of the fuel rod cladding due to strain caused by relative expansion of the UO_2 pellet, (c) fuel melting, (d) manufacturing defects and (e) debris in coolant. Cladding failure due to overpressure from vaporization of UO_2 following a rapid reactivity transient is not considered to be an operational transient and is discussed in Section III-6.0, "Nuclear Design".

Margin against the mechanism of severe overheating is discussed in greater detail in Section III-7.0, "Thermal and Hydraulic Design".

Bundle-specific operating limits are established to ensure that actual fuel operation is maintained within the fuel rod thermal-mechanical design and safety bases. The Maximum Linear Heat Generation Rate defines the maximum pellet operating limit. The anticipated operational occurrences are analyzed assuming maximum overpower limits to ensure the 1% cladding circumferential plastic strain and centerline melt criterion are met for a given cycle operation. Further explanation is found in Section 2 of NEDE-24011-P-A.[113]

The fuel damage limit of 1% plastic strain is based on tensile tests of cladding irradiated in the Vallecitos Boiling Water Reactor (VBWR).[8] Tensile tests of irradiated Dresden Unit 1 Type I Zircaloy support this limit. Two fuel rods irradiated in the General Electric Test Reactor (GETR) showed 1% plastic strain to correspond to approximately 28 kW/ft.[9]

The fuel operating limit and the fuel damage limit were originally established based on operating experience and experimental tests covering the complete range of design power and exposure levels. Table III-2-3 presents a summary of power reactor production fuel experience and Tables III-2-4 and III-2-5 show the range of development fuel irradiations which have already been completed, or are in progress. This experience has been used in establishing design features and in the analysis of performance characteristics. A number of Zircaloy fuel rods with 0.56-inch diameter and 0.030-inch cladding had been operated in the Vallecitos Boiling Water Reactor (VBWR) to peak exposures in excess of 10,000 MWd/T at a linear heat generation greater than 19.9 kW/ft. A large number of other fuel rods of smaller diameter had operated in VBWR at lower linear heat generation but to higher heat fluxes and exposure. Dresden Nuclear Power Station Unit 1 provided the largest body of operating experience on Zircaloy-clad fuel. Maximum Dresden Type I assembly average exposures had reached 21,500 MWd/T with peak fuel rod segments having attained 30,500 MWd/T. A limited number of what would be considered high performance (high exposure or power, or both) developmental fuel assemblies had been irradiated in the Dresden Unit 1 reactor core. The SA-1 fuel assembly with a peak exposure of approximately 36,500 MWd/T was discharged after more than five years of successful operation in the Dresden reactor. Initial operation of these fuel rods at a peak LHGR of 16.0 kW/ft was achieved in VBWR. The maximum LHGR achieved in Dresden operation was 13.0 kW/ft early in life. Dresden Unit 1 Type III fuel provided experience with the through rod and *spring spacer design. Type III assembly average exposures up to 18,600 MWd/T with peak local exposures of 27,000 MWd/T and peak linear heat generation of 15.4 kW/ft had been attained.*

To better determine the effects of exposure at high power generation, several capsules containing Zircaloy-clad fuel specimens with characteristics similar to current boiling water reactor fuel designs have been irradiated in the General Electric Test Reactor to exposures as high as 70,000 MWd/T and peak linear heat generation rates up to 29.4 kW/ft.

Each generic fuel design is reviewed and approved by the USNRC prior to use. See GE topical reports NEDE-24011-P-A and NEDE-31152P [112][113] for design details. Each reactor core reload, employing specific enrichment and gadolinia distributions, is analyzed to demonstrate that it meets design requirements and can be operated safely within the bounds of the cycle specific thermal operating limits. These limits are described in the Core Operating Limits Report (COLR) in accordance with Technical Specification Section 5.6.5.

The models used for fuel rod thermal analyses are described in GESTARII, General Electric Standard Application for Reactor Fuel.[113]

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. The limit is patterned after ANSI/ANS-57.5-1981. The figure of merit employed is the Design Ratio, where:

$$\text{Design ratio} = \frac{\text{Effective Stress}}{\text{Stress Limit}} \text{ or } \frac{\text{Effective Strain}}{\text{Strain Limit}}$$

Section 2 of NEDE-24011-P-A[113] describes the thermal and mechanical evaluations for all GE current designs. It has been determined that the stress/strain for GNF2 fuel limit is met.

Adequate free volume is provided in the fuel pellets and between the pellets and cladding to prevent overstraining of the cladding due to relative thermal expansion or UO_2 swelling from long term irradiation. Pellet-to-cladding gap has been specified such that the thermal expansion, any phase changes as a result of fuel melting, and fuel densification power spiking accommodates for worst case dimensional tolerances throughout life. The fuel rods experiencing the maximum expected power and exposure conditions are evaluated using the GE fuel rod thermal-mechanical performance model (GESTR-MECHANICAL)[112] to determine the maximum uniform cladding plastic strain during anticipated operational occurrences. For GNF2 fuel, the PRIME thermal-mechanical performance model is used. Section 2 of NEDE-24011-P-A describes the thermal and mechanical evaluations for all GE current design. The results of this evaluation show that less than 1% cladding circumferential plastic strain will result.

Fuel rod internal pressure is due to the helium which is backfilled during rod fabrication, the volatile content of the UO_2, and the fraction of gaseous fission products which are released from the UO_2. 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. The fuel rods experiencing the maximum expected power and exposure conditions are evaluated using the GE fuel rod thermal-mechanical performance model (GESTR-MECHANICAL) to determine the cladding creep-out rate due to internal gas pressure during normal steady-state operation. For GNF2 fuel, the PRIME thermal-mechanical performance model is used. Section 2 of NEDE-24011-P-A describes the thermal and mechanical evaluations for all GE current design. The results of this evaluation show that the cladding creep-out rate is lower than the fuel pellet irradiation swelling rate.

TABLE III-2-3
SUMMARY OF LEADING EXPERIENCE ON OPERATING PRODUCTION FUEL
ZIRCALOY-CLAD UO_2 PELLET FUEL

	Exposure[b] Mwd/MT Peak Pellet/ Average Assembly	Time[b] In-Core (Years)	Max Heat Flux[a,f] (Btu/hr-ft^2)	Peak LHGR (kW/ft) a,f	Fuel Rod Dia. (in.)	Clad Thickness (mils)	Pellet-to-Clad Gap (Nominal) (mils)	Active Fuel Length (in.)	Fission Gas Plenum (Volume Per Unit Fuel Vol)	Number Segments or Rods
Dresden Type I[g]	33,800/23,500[c]	9.25	350,000	15.2	0.567	30	3	106.5	0.013	144
Dresden Type IIIB	29,600/15,400	4.25	360,000	15.4	0.555	35	7.5	109.0	0.040	5,868
Dresden Type IIIF	27,800/15,600	3.95	360,000	15.5	0.5625	35	10	108.25	0.048	3,360
Dresden V	15,700/ 7,400	1.95	360,000	15.5	0.5625	35	10	108.25	0.048	3,816
Garigliano Type I	24,700/12,700	6.40	252,000	10.3	0.534	30	5	105.7	0.031	11,502
Garigliano Type II	8,800/ 5,400	1.20	320,000	14.6	0.593	37	11	107.0	0.030	3,456
KAHL	21,400/11,000[h]	7.60	296,000	13.0	0.5690	33	5	59.8	0.017	1,368
Humboldt										
Type II	20,300/11,700	4.00	325,000	12.1	0.186	33	10	79.0	0.043	6,468
Type III	5,060/ 2,500	0.75	389,000	16.8	0.563	32	11	79.0	0.062	1,872
KRB	17,100/10,100	3.10	367,000	15.8	0.5625	35	10	130.0	0.058	10,512
JPDR	8,200/ 3,800	5.60	205,000	13.0	0.557	30	5	56.75	0.100	5,184
Consumers										
BRP-B	29,500/19,500	3.70	434,000	15.0	0.449[d]	34	8	70.0	0.048	1,694
BRP-E	13,400/ 6,600	1.50	410,000	17.7	0.5625	40	11	70.0	0.048	2,926
BRP-EG	5,300/ 3,500	0.75	410,000	17.7	0.5625	40	11	70.0	0.048	1,694
Tarapur	2,700/ 1,400	0.85	365,000	15.8	0.5625	35	10.5	144.0	0.059	27,832
Current BWR[e]	45,000/27,500	5.0	428,000	18.5	0.563	32	11	144.0	0.11	18,032

[a] At rated power.
[b] Dresden and JPDR data as of September 5, 1969. All other data, excepting that for KAHL, is as of December 31, 1969.
[c] Lead Type I Assembly.
[d] 109 rods with 0.449-inch diameter, 12 corner rods with 0.344-inch diameter.
[e] Typical design as opposed to proven performance in preceding entries.
[f] License limit.
[g] 100 assemblies (14,400) segments discharged January 1, 1967, with average exposure of 12,600 MWd/MT.
[h] As of August 31, 1968.

TABLE III-2-4
GENERAL ELECTRIC DEVELOPMENTAL IRRADIATIONS
ZIRCALOY-CLAD 95% TD UO_2 PELLET FUEL RODS

Name	Reactor	No. of Rods	Fuel Rod Dia. (in.)	Clad Wall Thickness (in.)	Pellet-to-Clad Gap (mils)	Peak Heat Flux (Btu/hr-ft^2)	Peak LHGR (kW/ft)	Peak Exposure (MWd/MT)	Status
Dresden Prototype	VBWR	9	0.565	0.030	3.0-16.0	460,000	19.94	12,000	Completed
Fuel Cycle (R&D)[a]	VBWR	144	0.242	0.022	2.0-8.0	509,000	16.6	13,800	Completed
Dresden Prototypes	VBWR	52	0.565	0.028	5.0-8.0	407,000	17.64	10,000	Completed
High Performance	GETR	12	0.565	0.030	4.0-6.0	630,000	27.0	1,500	Completed[h]
UO_2[b]						1,126,000	49.0		
High Performance UO_2[b]	GETR	2	0.565	0.030	4.0-11.0	1,355,000	58.0	14,000	Completed
SA-1[c]	Dresden I	98	0.424	0.022	4.0-8.0	400,000	13.0	40,000	Completed
D-1,2,3[d]	Consumers	363	0.424	0.030	7.0	434,000	14.2	30,000	Completed
D-50[f]	Consumers	36	0.570	0.035	12.0	507,000	22.0	15,400	Under Evaluation[g,]
D-52,53	Consumers	58	0.700	0.040	13.0	525,000	27.0	4,600	Under Evaluation

[a] USAEC Contract AT (04-3)-189 Project Agreement 11.
[b] USAEC Contract AT (04-3)-189 Project Agreement 17.
[c] USAEC Contract AT (04-3)-189 Project Agreement 41.
[d] USAEC Contract AT (04-3)-361.
[e] Hollow Pellet.
[f] USAEC Contract AT (04-3)-189 Project Agreement 50.
[g] Eight fuel rods failed during second operating cycle due to abnormal crud and scale deposition.
[h] One rod failure at 49 kW/ft.

TABLE III-2-5

GENERAL ELECTRIC DEVELOPMENTAL IRRADIATIONS
ZIRCALOY-CLAD 95% TD UO_2 PELLET CAPSULES
GENERAL ELECTRIC TEST REACTOR

Capsule	Number Of Rods	Fuel Rod Dia. (In.)	Clad Wall Thickness (In.)	Pellet-to-Clad (mils)	Peak Heat Flux (Btu/h-ft^2)	Peak LHGR (kW/ft)	Peak Exposure[a] MWd/MT)	Status
A	3	0.425	0.024-0.032	1.4-10.2	750,000	24.5	77,000	Complete
	1	0.488	0.032	11.2	785,000	29.4	34,000	Complete
B	6	0.489	0.034	7.7-11.6	503,000	18.8	41,000	Continuing
C	5	0.557	0.036	2.0-15.0	404,000	17.7	34,000	Continuing[b]
D	5	0.557	0.036	2.0-1.2	540,000	23.0	37,000	Complete

[a]Peak exposure status is as of 11/30/69.
[b]Two sound rods removed for detailed Radioactive Materials Laboratory examination with □ 30,000 MWd/MT exposure.

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Flow-induced fuel rod vibrations depend primarily on flow velocity and fuel rod geometry. For the range of flow rates and geometrical variations for the plant, vibrations do not exceed an amplitude of 0.002 inch at the expected frequency of 36 Hz. The maximum vibrational amplitude occurs midway between spacers due to the constraint of the spacer contact points. The stress levels resulting from the vibrations are negligibly low and well below the endurance limit of all affected components.[16] Spacer design is based on reactor testing where it was identified that the use of an active spring force by the spacer on the fuel rod eliminates the potential for any fretting wear. Subsequent spacer designs have been based on these test results and the concept of an active spring force. GE14 uses ferrule spacer design.

The fuel assembly and the fuel rod cladding are evaluated to ensure the strains due to cyclic loading will not exceed the fatigue capability. The fatigue life analysis is based on the estimated number of temperature, pressure, and power cycles. The fatigue evaluations use the GE fuel rod thermal−mechanical performance model (GESTR−MECHANICAL). Section 2 of NEDE−24011−P−A describes the thermal and mechanical evaluations for all GE current design. The results of this evaluation show that the material fatigue capability criterion is met.

The fuel rod is evaluated to ensure that fuel rod failure due to fuel melting will not occur. The fuel rod is evaluated to ensure that fuel melting during normal steady-state operations and whole core anticipated operational occurrences (AOO) is not expected to occur. For local AOO, a small amount of calculated fuel pellet center melting may occur but is limited by 1% cladding circumferential plastic strain.

The fuel rods experiencing the most severe expected power and exposure conditions are evaluated with GESTR−MECHANICAL to determine the fuel centerline temperature during the maximum whole core AOO. The results of this evaluation indicate that fuel melting is not expected to occur for the fuel designs. In addition, each cycle's AOO results are evaluated compared to this maximum result to assure that cycle operation is bounded.

Reuse of fuel channels can result in excessive channel box bow impacting thermal limits, and control rod scram times. CNS has committed to use new channels for each new fuel bundle received. [98]

The nuclear fuel pellets and cladding serve as the initial barrier from release of fission products to the reactor coolant. A value of 1% of plastic strain of Zircaloy cladding is defined as the limit below which overstraining of the cladding is not expected to occur. Design evaluations show power required to produce 1% plastic strain in the cladding or fuel centerline melting is not expected to occur during normal operation or anticipated operational occurrences for each cycle of operation.

Based on this information, the nuclear fuel provides a high integrity assembly of fissionable material capable of transferring heat to the reactor coolant. The fuel maintains its structural integrity, retains its fission products and acts as a fission product barrier during operation as required by the Safety Design Basis, Section 2.3.

2.7 Inspection and Testing

Inspection and testing of fuel during fabrication is performed in accordance with the vendor's approved quality assurance program.

Dimensional measurements and visual inspections of critical areas are verified at the reactor site by NPPD personnel on each fuel bundle.

Each fuel assembly is assured of proper orientation within a four-assembly cell based on visual verification of the direction of the fuel assembly identification number and the projecting boss on the upper tie plate handle. The identification number is engraved on the top of the upper tie plate handle for each fuel assembly to be readable from the direction of the center of the cell. The boss of the upper tie plate handle projects from only one side of the handle towards the center of the cell.

Loading of the core is recorded with sufficient resolution to back-check the fuel orientation. The recording is retained during the subsequent operating cycle.[21]

3.0 REACTOR VESSEL INTERNALS MECHANICAL DESIGN

3.1 Safety Objective

The reactor vessel internals shall be designed to direct and control coolant flow distribution and to provide a floodable volume in which the core can be adequately cooled, thus limiting fuel damage during loss of coolant accidents.

The reactor vessel internals and fuel assemblies shall be designed such that their deflections and deformations are limited to assure that the control rod movement is not impaired.

3.2 Safety Design Bases

The reactor vessel internals shall meet the following safety design bases:

a. Shall be arranged to provide a floodable volume in which the core can be adequately cooled in the event of a breach in the nuclear system process barrier external to the reactor vessel.

b. Deformation of internals shall be limited to assure that the control rods and core standby cooling systems can perform their safety functions.

c. Mechanical design of applicable internals shall assure that safety design bases a and b, above, are satisfied so that the safe shutdown of the plant and removal of decay heat are not impaired.

3.3 Power Generation Objectives

Reactor vessel internals (exclusive of fuel, control rods, and in-core nuclear instrumentation) are provided to achieve the following objectives:

a. Maintain partitions between regions within the reactor vessel to provide correct coolant distribution, thereby allowing power operation without fuel damage.

b. Provide positioning and support for the fuel assemblies, control rods, in-core flux monitors, and other vessel internals to assure that control rod movement is not impaired.

3.4 Power Generation Design Bases

The reactor vessel internals shall be designed to the following power generation design bases:

a. They shall provide the proper coolant distribution during all anticipated normal operating conditions to allow power operation of the core without fuel damage.

b. They shall be arranged to facilitate refueling operations.

c. They shall be designed to facilitate inspection.

3.5 Description

The reactor vessel internals include the following components:

Core shroud
Shroud head and steam separator assembly
Shroud support ring
Core Support (core plate)
Top guide
Fuel support pieces
Control rod guide tubes
Jet pump assemblies
Steam dryers
Feedwater spargers
Vessel head nozzles
Differential pressure and Standby Liquid Control line
In-core flux monitor guide tubes
Surveillance sample holders

The overall arrangement of the internals within the reactor vessel is shown in Figure III-3-1.

Although not mandatory, the design of the reactor vessel internals is in accordance with the intent of Section III of the ASME Boiler and Pressure Vessel Code. The material used for fabricating most of the reactor vessel internals is solution heat-treated, unstabilized Type 304 austenitic stainless steel conforming to ASTM specifications. Weld procedures and welders are qualified in accordance with the intent of Section IX of the ASME Boiler and Pressure Vessel Code.

In order to insure that the reactor pressure vessel (RPV) and internals are capable of withstanding anticipated thermal stresses, the design of the RPV and internals was based upon certain thermal cycle specifications obtained from General Electric by Combustion Engineering. Combustion Engineering used these values to perform a fatigue analysis of the RPV and internals based upon USAS B31.7-1969. The fatigue analysis has been re-evaluated for past operating cycles and for projected future cycles to demonstrate that ASME Code-limiting conditions are not exceeded for 60 years of operation. In addition, a reconstituted stress analysis has been performed for the RPV components that were identified as the limiting candidates for environmentally assisted fatigue (see Section K-2.1.15). These components involve cyclic application of loads and thermal conditions, as well as operating experience and/or identified fatigue margin.

In compliance with Technical Specification 5.5.5, NPPD monitors operational transient occurrences listed on Table III-3-1 on a refueling cycle basis, and their affect on the RPV and pressure boundary piping fatigue stress to ensure that fatigue cumulative usage factors remain less than 1.0.

The floodable inner volume of the reactor vessel can be seen in Figure IV-3-5. It is the volume inside the core shroud up to the level of the jet pump suction inlet. The boundary of the inner volume consists of the following:

a. The jet pumps from the jet pump suction inlet down to the shroud support ring.

b. The shroud support ring, which forms a barrier between the outside of the shroud and the inside of the reactor vessel.

c. The reactor vessel wall below the shroud support ring.

d. The core shroud up to the level of the jet pump suction inlet.

TABLE III-3-1

REACTOR VESSEL FATIGUE TRANSIENT OCCURRENCES

FATIGUE EVENT

Boltup

Design Hydrostatic Test (1250 psig)

Startup

Turbine Toll and Increase to Rated Power

Reduction to 75%

Reduction to 50%

Rod Worth Test

Loss of Feedwater Heater – Turbine Trip at 25% Power

Loss of Feedwater Heater – Feedwater Heater Bypass

Loss of Feedwater Pump, Isolation Valve Closed

Turbine-Generator Trip, Feedwater Stays On, Isolation Valves Stay Open

Reactor Overpressure with Delay Scram, Feedwater Stays On, Isolation Valves Stay Open

Single Relief or Safety Valve Blowdown

Other Scrams

Rated Power Normal Operation

Improper Start of Cold Recirculation Loop

Sudden Start of Pump in Cold Recirculation Loop

Reduction to 0 Percent Power

Hot Standby

Shutdown

Shutdown Vessel Flooding

Hydrostatic Test (1563 psig)

Unbolt

3.5.1 Core Structure

The core structure surrounds the active core of the reactor and consists of the core shroud, shroud head and steam separator assembly, core support, and top guide. This structure is used to form partitions within the reactor vessel, to sustain pressure differentials across the partitions, to direct the flow of the coolant water, and to locate laterally and support the fuel assemblies, control rod guide tubes, and steam separators. Figure III-3-2 shows the reactor vessel internal flow paths.

3.5.1.1 Core Shroud

The core shroud is a stainless steel cylindrical assembly that provides a partition to separate the upward flow of coolant through the core from the downward recirculation flow. This partition separates the core region from the downcomer annulus, thus providing a floodable region following a recirculation line break. The volume enclosed by the core shroud is characterized by three regions, each with a different shroud diameter. The upper shroud has the largest diameter and surrounds the core discharge plenum, which is bounded by the shroud head on top and the top guide below. The central portion of the shroud surrounds the active fuel and forms the longest section of the shroud. This section has an intermediate diameter and is bounded at the bottom by the core support. The lower shroud, surrounding part of the lower plenum, has the smallest diameter and, at the bottom, is welded to the reactor vessel shroud support ring. (See Section IV-2, "Reactor Vessel and Appurtenances Mechanical Design".) The shroud support ring has two openings allowing access to the lower head area during construction. The two opening covers were welded closed at the end of construction.

3.5.1.2 Shroud Head and Steam Separator Assembly

The shroud head and steam separator assembly is bolted to the top of the upper shroud to form the top of the core discharge plenum. This plenum provides a mixing chamber for the steam-water mixture before it enters the steam separators. Individual stainless steel axial flow steam separators, shown in Figure III-3-3, are attached to the top of standpipes that are welded into the shroud head. The steam separators have no moving parts. In each separator, the steam-water mixture rising through the standpipe passes vanes that impart a spin to establish a vortex separating the water from the steam. The separated water flows from the lower portion of the steam separator into the recirculation flow downcomer annulus.

3.5.1.3 Core Support

The core support (core plate) consists of a circular stainless steel plate stiffened with a rim and beam structure. Perforations in the plate provide lateral support and guidance for the control rod guide tubes, in-core flux monitor guide tubes, peripheral fuel support pieces, and any test capsules installed in the startup neutron source locations. The last two items are also supported vertically.

The entire assembly is bolted to a support ledge, between the central and lower portions of the core shroud. Alignment pins that bear against the shroud are used to correctly position the assembly before it is secured.

The 88 core plate bypass holes are plugged in order to eliminate in-core instrument vibration that was causing damage to fuel channels.

The plug body guides the device into the bypass flow holes, provides a shoulder to support the plug and forms a seal against water flow.

The shaft extends through the body. At the bottom, a latch is attached to the shaft and bears on the bottom of the core plate. A spring acts against the body and shaft during normal operation to provide the force necessary to offset the pressure differential acting on the body.

The spring material is Inconel X-750, spring-temper-conditioned with age hardened heat treatment. All other parts are Type-304 stainless steel.

A stress analysis was performed on the plug in the core support plate. Normal operating conditions, pressure and thermal transients, and installation/removal operations were considered in the analysis. The results show acceptable stress levels in all plug components during normal operation, and pressure, and thermal transients. Plug assembly cycles produce extreme fibre torsional shear stresses in the spring near yield strength. Spring tests show that a loss of spring free length results from the assembly cycle. This loss is predictably small and will not detract from the functional adequacy of the spring.

Some load relaxation will be experienced after plug installation. Elevated temperature testing has shown this relaxation to be a small percentage of the total preload. Creep in the stainless steel latch was experimentally investigated and was found to have negligible effect on plug preload. The combined effect on plug preload of: (1) the plug assembly cycles, (2) the spring relaxation, and (3) latch creep, will result in a final operating preload margin of nearly three times the operating static pressure differential across the plug, and a margin of two over the worst-case transient static pressure differential. For this reason, the core plate bypass plugs have a service life of thirty two effective full power years (EFPY).[92]

The analysis for the load required to extract the plug from the core plate shows that the assembly will maintain its integrity during this operation.[26]

Adequate flow to the bypass region is provided by two holes in the bundle lower tie plate. These holes provide the required cooling flow to the bypass region without increasing vibration of core neutron monitoring equipment.

3.5.1.4 Top Guide

The top guide is formed by a series of stainless steel beams joined at right angles to form square openings. Each opening provides lateral support and guidance for four fuel assemblies. Holes are provided in the bottom of the beams to anchor the in-core flux monitor guide tubes and startup neutron sources. The top guide is positioned with alignment pins that bear against the shroud.

3.5.2 Fuel Support Pieces

The fuel support pieces, shown in Figure III-3-4, are of two basic types, namely peripheral and four-lobed. The peripheral fuel support pieces, which are welded to the core support assembly, are located at the outer edge of the active core and are not adjacent to control rods. Each peripheral fuel support piece will support one fuel assembly and contains an orifice assembly. Each four-lobed fuel support piece will support four fuel assemblies and is provided with four orifice plates. See Section III-2 for discussion on fuel support plugs (dummy bundles).

The four-lobed fuel support pieces rest in the top of the control rod guide tubes and are supported laterally by the core support. A slot on the side of the fuel support piece fits around a pin on the core support. This arrangement provides proper orientation for the fuel support piece (Figure III-3-4). The control rods pass through slots in the center of the four-lobed fuel support pieces. A control rod and the four adjacent fuel assemblies represent a core cell. (See Section III-2, "Fuel Mechanical Design".)

Correct distribution of core coolant flow among the fuel assemblies is accomplished by the use of a fixed orifice at the inlet of each fuel assembly. The orifices are located in the fuel support piece. They serve to control the flow distribution and the coolant conditions within prescribed bounds throughout the design range of core operations. The core is divided into two orifice zones. The outer zone is a narrow, reduced-power region around the periphery of the core. The inner zone consists of the core center region. No other control of flow is used or needed. A blank is placed in the orifice slot for the fuel support plugs (dummy bundles). This ensures no coolant flow bypasses the core. A hold down pin is provided in these fuel support pieces to hold the original fuel support plug seated, however current design does not require this pin to maintain the fuel support plug seated. The hold down pin also prevents a fuel bundle from being seated into a location with no flow.

3.5.3 Control Rod Guide Tubes

The control rod guide tubes, located inside the vessel, extend from the top of the control rod drive housings up through holes in the core support. Each tube is designed as the lateral guide for a control rod and as the vertical support for a four-lobed fuel support piece and the four fuel assemblies surrounding the control rod. The bottom of the guide tube is supported by the control rod drive housing (see Section IV-2, "Reactor Vessel and Appurtenances Mechanical Design"), which in turn transmits the weight of the guide tube, fuel support piece, and fuel assemblies to the reactor vessel bottom head. A thermal sleeve is inserted into the control rod drive housing from below and is rotated to lock the control rod guide tube in place. A key is inserted into a locking slot in the bottom of the control rod drive housing to hold the thermal sleeve in position.

3.5.4 Jet Pump Assemblies

The jet pump assemblies are located in two semi-circular groups in the downcomer annulus between the core shroud and the reactor vessel wall. Each stainless steel jet pump consists of a driving nozzle, suction inlet, throat or mixing section, and diffuser (see Figure III-3-5). The driving nozzle, suction inlet, and throat are joined together as a removable unit, and the diffuser is permanently installed. High pressure water from the recirculation pumps (see Section IV-3, "Reactor Recirculation System") is supplied to each pair of jet pumps through a riser pipe welded to the recirculation inlet nozzle thermal sleeve. A riser brace is welded to cantilever beams extending from pads on the reactor vessel wall.

The nozzle entry section is connected to the riser by a metal-to-metal, spherical-to-conical seal joint. Firm contact is maintained by a holddown assembly. The throat section is supported laterally by a restrainer bracket attached to the riser. There is a slip-fit joint between the throat and diffuser. The diffuser is a gradual conical section changing to a straight cylindrical section at the lower end.

During the installation of the 316NG Recirculation System Piping (see Section IV-3.4), a number of jet pumps could not be seated properly due to an inlet mixer and riser misalignment. To correct this misalignment on jet pumps number 1, 2, 9, and 10, the jet pump restrainer bracket radial positioning screws have been modified or removed, and gravity actuated restrainer wedges are installed where needed. [74] [75]

JET PUMP NO.	VESSEL SIDE	SHROUD SIDE
1	Wedge	No Wedge
2	Wedge	Wedge
9	Wedge	Wedge
10	Wedge	No Wedge

3.5.5 Steam Dryers

The steam dryers remove moisture from the wet steam leaving the steam separators. The extracted moisture drips down the dryer vanes to the collecting troughs, then flows through tubes into the downcomer annulus (see Figure III-3-6). A skirt extends from the top of the steam dryers to the steam separator standpipe, below the water level. This skirt forms a seal between the wet steam plenum and the dry steam flowing from the top of the dryers to the steam outlet nozzles.

The steam dryer and shroud head are positioned in the vessel with the aid of vertical guide rods. The dryer assembly rests on steam dryer support brackets attached to the reactor vessel wall. Upward movement of the dryer assembly is restricted by steam dryer hold-down brackets attached to the reactor vessel top head.

3.5.6 Feedwater Spargers

The feedwater spargers have three concentric thermal sleeves, the innermost of which conducts feedwater to the sparger arms. The arms are attached to the sleeve by a forged tee, are fastened to the reactor vessel wall at their end points by brackets, and are designed to deliver feedwater uniformly to the annular area between the core shroud and the vessel wall. In so doing, they provide subcooling for the jet pumps and help maintain a uniform core power distribution. The sparger arms discharge feedwater into the vessel through elbows mounted on top and fitted with converging discharge nozzles. These features reduce temperature stratification in the sparger and flow separation around the periphery of the flow holes at low feedwater flow. The spargers prevent the existence of significant flux asymmetrics or non-uniform enthalpy distributions. This sparger design protects the feedwater nozzle from high frequency thermal cycles, which can initiate nozzle cracks. Its design reduces potential for thermal fatigue cracking and vibration which cause sparger arm cracks.

Bypass leakage flow in the feedwater nozzle bore will be reduced substantially by two piston-ring seals and an interference fit. Water leaking past the first seal would pass into the vessel through the annulus between the inner sleeve and the "mid-thermal" sleeve, which is supported at its upstream end by a slotted attachment to the inner sleeve. Attached to the "mid-thermal" sleeve is an outer sleeve that is fitted tightly in the nozzle bore at the upstream end to prevent vibratory motion and fatigue damage of the sparger assembly. The secondary piston-ring seal at that tight interference joint reduces potential bypass flow to nearly zero because the pressure drop is very low across the secondary seal. In addition, the three concentric sleeves will prevent formation of a cold boundary layer of water in the annulus next to the nozzle bore.[28][89]

For further information regarding the feedwater nozzles, see Section IV-2, "Reactor Vessel and Appurtenances Design."

3.5.7 Core Spray Lines

Two core spray lines enter the reactor vessel through the two core spray nozzles. (See Section IV-2, "Reactor Vessel and Appurtenances Mechanical Design.") The lines divide immediately inside the reactor vessel. The two halves are routed to opposite sides of the reactor vessel and are supported by clamps attached to the vessel wall. The lines are then routed downward into the downcomer annulus and pass through the upper shroud immediately below the flange. The flow divides again as it enters the center of the semicircular header, which is routed halfway around the inside of the upper shroud. The ends of the two headers are supported by brackets designed to accommodate thermal expansion. The line routing and supports are designed to accommodate differential movement between the shroud and vessel. The other core spray line is identical except that it enters the opposite side of the vessel and the headers are at a slightly different elevation in the shroud. The correct spray distribution pattern is provided by a combination of distribution nozzles pointed radially inward and downward from the headers. (See Chapter VI, "Emergency Core Cooling Systems.")

3.5.8 Differential Pressure and Standby Liquid Control Line

The differential pressure and standby liquid control line serves a dual function within the reactor vessel--to inject standby liquid control solution into the coolant stream (discussed in Section III-9, "Standby Liquid Control System") and to sense the differential pressure across the core support assembly (described in Section IV-2, "Reactor Vessel and Appurtenances Mechanical Design"). This line enters the reactor vessel at a point below the core shroud as two concentric pipes. In the lower plenum, the two pipes separate. The inner pipe terminates near the lower shroud with a perforated length below the core support assembly. It is used to sense the pressure below the core support during normal operation and to inject liquid control solution when required. This location facilitates good mixing and dispersion. The inner pipe also reduces thermal shock to the vessel nozzle should the standby liquid control system be actuated. The outer pipe terminates immediately above the core support and senses the pressure in the region outside the fuel assembly channels.

3.5.9 In-Core Flux Monitor Guide Tubes

The in-core flux monitor guide tubes extend from the top of the in-core flux monitor housings (see Section IV-2, "Reactor Vessel and Appurtenances Mechanical Design") in the lower plenum to the top of the core support. The power range detectors for the power range monitoring units and the dry tubes for the source range monitoring and intermediate range monitoring (SRM/IRM) detectors are inserted through the guide tubes. The guide tubes are held in place below the top guide by spring tension. A latticework of clamps, tie bars, and spacers gives lateral support and rigidity to the guide tubes. The bolts and clamps are welded, after assembly, to prevent loosening during reactor operation.

3.5.10 Initial Startup Neutron Sources

Startup neutron sources were used for initial start-up to provide an initial neutron source high enough to indicate on source range neutron monitors. These sources are no longer required to provide a neutron source so they are no longer installed in the core.

The startup source core locations are occasionally used for irradiation of materials test capsules in the CNS core.

3.5.11 Surveillance Sample Holders

The surveillance sample holders are welded baskets containing impact and tensile specimen capsules (see Section IV-2, "Reactor Vessel and Appurtenances Mechanical Design"). Each capsule contains iron, nickel, and copper wire dosimeters. The baskets hang from brackets that are attached to the inside wall of the reactor vessel and extend to mid-height of the active core. The radial positions are chosen to expose the specimens to the same environment and maximum neutron fluxes experienced by the reactor vessel itself while avoiding jet pump removal interference or damage.

3.6 Safety Evaluation

3.6.1 Evaluation Methods

To determine that the safety design bases are satisfied, responses of the reactor vessel internals to loads imposed during normal, upset, emergency, and faulted seismic conditions are examined. The effects on the ability to insert control rods, cool the core, and flood the inner volume of the reactor vessel is determined.

The ASME Boiler and Pressure Vessel Code, Section III for Class A Vessels, is used as a guide to determine limiting stress intensities and cyclic loading for the reactor vessel internals. When buckling is not a possible failure mode and stresses are within those stated in the ASME Code, either the elastic stability of the structure or the resulting deformation is examined to determine whether the safety design bases are satisfied.

3.6.1.1 Events to be Evaluated

Examination of the spectrum of conditions for which the safety design basis must be satisfied reveals three significant events:

a. Loss-of-coolant accident: a break in a recirculation line. The accident results in pressure differentials, across some of the reactor vessel internals, that exceed normal loads.

b. Steam line break accident: a break in one main steam line between the reactor vessel and the flow restrictor. The accident results in significant pressure differentials across some of the reactor vessel internals.

c. Earthquake: subjects the reactor vessel internals to significant forces as a result of ground motion.

Analysis of other conditions existing during normal operation, abnormal operational transients, and accidents shows that the loads affecting most of the reactor vessel internals are less severe than the three postulated events listed above.

3.6.1.2 Pressure Differential During Rapid Depressurization

A digital computer code[29] is used to analyze the transient conditions in the reactor vessel following the loss-of-coolant accident and the design basis steam line break accident. The analytical model of the vessel consists of nine nodes, which are connected to the necessary adjoining nodes by flow paths having the required resistance and inertial characteristics. The program solves the energy and mass conservation equations for each node to give the depressurization rates in the various regions of the reactor.

The nine nodes are: 1) the subcooled lower plenum, 2) the core, 3) the upper plenum, 4) the separation zone, 5) the downcomer, 6) the broken loop (if applicable), 7) the intact loop, 8) the core bypass region, and 9) the steam dome.

The flow resistances are evaluated from the irreversible pressure drops associated with known flow rates. If the accident being considered is a rupture in the recirculation loop, an additional flow path exists through the diffusers of the inoperative jet pumps.

A one-dimensional, incompressible form of the momentum equation is solved to obtain the flows in the reactor vessel internal flow paths.

Figure III-3-7 shows the reactor nodes. The pressure differentials acting on the major components are defined as follows:

Component	Pressure Loading
Core Plate and Guide Tube	(1)-(3)
Baffle Plate and Lower Shroud	(1)-(5)
Upper Shroud and Shroud Head	(3)-(9)
Channel Box	(2)-(3)
Dryer	(4)-(5)

3.6.2 Recirculation Line and Steam Line Break

3.6.2.1 Accident Definition

The recirculation line break is the same as the design basis loss-of-coolant accident described in Section VI, "Core Standby Cooling Systems", and Section XIV, "Station Safety Analysis". A sudden, complete circumferential break is assumed to occur in one recirculation loop.

The analysis of the steam line break assumes a sudden, complete circumferential break of one main steam line between the reactor vessel and the main steam line restrictor. This is not the same accident as that described in Section XIV, "Station Safety Analysis". A steam line break upstream of the flow restrictors produces a larger blowdown area than a break downstream of the restrictors. The larger blowdown area results in greater pressure differentials across the reactor assembly internal structures.

Both the recirculation line break and steam line break have been examined for a spectrum of initial reactor operating conditions. These studies show that a recirculation line break at any initial reactor condition would be very mild insofar as resultant pressure differentials are concerned. (See Section III-5.2.2.)

The steam line break accident produces significantly higher pressure differentials across the reactor assembly internal structures than does the recirculation line break. This results from the higher reactor depressurization rate associated with the steam line break. The depressurization rate is proportional to the mass flow rate and the excess of fluid escape enthalpy above saturated water enthalpy h_f. Mass flow rate is inversely proportional to escape enthalpy, h_e, and the depressurization rate is approximately proportional to $1 - h_f/h_e$. Consequently, depressurization rate decreases as h_e decreases, that is, the depressurization rate is less for mixture flow than for steam flow. Therefore, the steam line break is the design basis accident for internal pressure differentials.

For a steam line break accident, a low initial reactor power level results in a more severe pressure transient across some components than would be the case at maximum power. This is because the difference between energy removal through the break and energy addition to the reactor vessel inventory increases as the reactor power decreases. Thus the depressurization rate following a steam line break would increase with decreasing initial reactor power level.

The steam line break inside the drywell is analyzed for two sets of initial conditions in the reactor. Case 1 assumes the reactor to be operating at 105-percent rated steam flow condition. For Case 2, the reactor is assumed to be at 23-percent power with 110-percent rated recirculation flow at the time of the break. The initial values of key nuclear system parameters are as follows:

	Case 1	Case 2
Core power (MWt)	2486	537
Steam rate (lb/hr)	10×10^6	1.9×10^6
Recirculation flow (lb/hr)	77.2×10^6	81×10^6

The recirculation line break is analyzed at the Case 1 initial conditions. Note that a steam line break at low reactor power would impose less severe requirements on the CSCS, because there would be less stored heat in the core. The assumed initial conditions of Case 2 represent the most severe possible situation. If a steam line break accident occurred during a reactor startup (at low power and natural recirculation flow) the resulting loads would be similar but less severe.

The maximum differential pressures across the reactor assembly internals resulting from the postulated accidents are shown in Table III-3-2. Figures III-3-8 and III-3-9 show the differential pressures for various internals as a function of time.

In order to assure that the reactor and internals are adequately designed to resist the oscillatory nature of the blowdown forces (Figures III-3-8 and III-3-9), a dynamic analysis was made of the reactor internal components being acted upon by the applied forces. The component natural periods are determined from a comprehensive dynamic model of the RPV and internals, including hydrodynamic mass effect of the water inside the RPV. The dynamic model of the reactor and internals used is shown in Figure III-3-10.

TABLE III-3-2

PRESSURE DIFFERENCES ACROSS REACTOR VESSEL INTERNALS

Component	Initial Steady Status Values Case 1	Initial Steady Status Values Case 2	Maximum During Steam Line Break Case 1	Maximum During Steam Line Break Case 2	Maximum During Recirc. Line Break Case 1
Core plate and guide tube	22	20	30	32	19
Baffle plate and lower shroud	31	25	49	53	27
Upper shroud and shroud head	9	5	30	31	10.5
Channel box	10	5	15	12	10
Dryer*	0.3	0.011	7	10	0.3

(values in psid)

*Because the only consequence of a dryer failure would be (possibly) interference with isolation valve closure and because isolation valve closure is not essential for a break inside the drywell, the dryer differential is evaluated for a steam line break outside the primary containment.

The natural frequency for the first five modes for a typical 218 size reactor are:

Mode	Vertical Frequency
1	18.5 Hz
2	27.2
3	58.1
4	68.9
5	71.4

Since these frequencies are more than ten times as high as the highest frequency components of the pressure differences shown in Figures III-3-8 and III-3-9, the resonant amplification is less than 1.01.

The dynamic analysis is performed by determining the natural frequencies and mode shapes of the internal components under consideration. The oscillatory forces are then applied to these components to determine the dynamic load response. The dynamic loads are combined with appropriate normally occurring loads and used for design.

A dynamic analysis of the intact loop due to LOCA is not performed as the effects are considered to be minor. The rapid depressurization of the reactor vessel following a LOCA would not create temperature changes or pressure surges in the piping systems of any pipe stress significance.

The flow transient in the recirculation system due to pump trip and coast down following a LOCA is essentially the same as that of a pump trip during pre-operational testing.

A summary of the results of the dynamic analysis is included in the loading combinations listed in Table C-3-7.[30]

3.6.2.2 Response of Reactor Internals to Pressure Differences

The maximum differential pressures are used, in combination with other structural loads, to determine the total loading on the various reactor vessel internals. The internals are then evaluated to assess the extent of deformation and collapse, if any. Of particular interest are (a) the responses of the guide tubes and the metal channels around the fuel bundles, and (b) the potential leakage around the jet pump joints.

The recirculation line break and the steam line break event result in the largest hydraulic loading on control rod guide tubes. The following information concerning the control rod guide tubes has been provided in Amendment No. 14 to Browns Ferry Nuclear Plant, Docket No. 50-259 and is applicable to the CNS control rod guide tubes:

a. The pressure differential developed across the control rod guide tube wall versus time,

b. The lateral hydraulic loading experienced by a control rod guide tube versus time,

c. The column load on the control rod guide tube versus time,

d. The control rod displacement versus time, when time zero is taken as the instant of the recirculation line break,

e. An analysis considering the possibility of elastic instability on tube collapse due to the partial effects of the above hydraulically developed loads.[27]

The guide tube is evaluated for collapse caused by externally applied pressure. A number of formulas are used to calculate the collapse pressure of the lower shroud and core support assembly. Use of ASME curves indicates the extreme sensitivity to wall thickness. For the minimum wall thickness of 10-in. Schedule 10 pipe, the ASME curves give a collapse load of 45 psi. Using the average wall thickness, the collapse pressure is increased to more than 70 psi. Using empirical relations for tubes over the critical length, the calculated collapse pressure is more than 100 psi. The ASME curves calculate that the collapse pressure is reached at 54 psi for a wall thickness of 0.150-in., which is 6 mils over the minimum for 10-in. Schedule 10 pipe. The calculated total loading for the guide tubes is considerably below the collapse loading, and it can be concluded that no failure occurs. The analysis also indicates that the control rods are 70 to 80 percent inserted at the time the maximum external pressure is applied to the guide tubes.

The fuel channel load resulting from an internally applied pressure is evaluated, utilizing a fixed-beam analytical model under a uniform load. Tests to verify the applicability of the analytical model indicate that the model is conservative. The fuel channels may deform sufficiently outward to cause some interference with movement of the control rod blade. There are approximately 15 factors, such as fuel channel deformation, core support tolerance, and top fuel guide beam location that determine the clearance between the control rod blade and fuel channel. If each tolerance factor is assumed to be at the worst extreme of the tolerance range, then a slight interference would develop under an 18-psi pressure difference across the channel wall. However, the maximum calculated pressure differential is only 13 psid. A roller at the top of the control rod guides the blade as it is inserted. The clearance between channels is 70 mils less than the diameter of the roller, causing it to slide or skid instead of roll. As the rod is inserted approximately halfway, the control rod sheath tends to push inward on the channel and cause the control rod surface to contact the channel surface. A "worst case" study indicates the possibility of a 50-mil interference. Starting with Cycle 24, some of the control rod blades will have the rollers at the top of the blade replaced by spacer pads. These lie flatter against the blade than the rollers and so will be bounded by the results for the rollers.

The possibility of a worst case developing is extremely remote. A statistical analysis utilizing a normal distribution for each of the 15 variables indicates that no interference occurs within 3σ limits, where σ is the standard deviation in a point distribution of events (3σ lies in the 0.995 percentile of probability of nonoccurrence). However, even if interference occurs, the result is negligible. About one pound of lateral force is required to deflect the channel inboard one mil. The friction force developed is an extremely small percentage of the total force available to the control rod drives.

The previous discussion presupposes that the control rod has not moved when the fuel channel experiences the largest magnitude of pressure drop. Analysis indicates that the rod is about 70 to 90 percent inserted. If the rod is beyond 70 percent inserted, then no interference is likely to develop because all the channel deformation is in the lower portion of the fuel channel, whereas the roller is at the top of the rod. It is concluded that the main steam line break accident can pose no significant interference to the movement of control rods.

Jet Pump Joints: An analysis has been performed to evaluate the potential leakage from within the floodable inner volume of the reactor vessel during the recirculation line break and subsequent LPCI reflooding.[103] The two possible sources of leakage are:

a. Jet pump throat to diffuser joint
b. Jet pump nozzle to riser joint

The jet pump to shroud support joint is welded and therefore is not a potential source of leakage. The slip joints for all jet pumps leak no more than a total of 225 gpm. The jet pump bolted joint, by analysis, is shown to leak no more than 582 gpm for the pumps through which the vessel is being flooded.

The summary of maximum leakage is:

Total leakage through all throat to diffuser joints	225 gpm
Total leakage through all nozzle to riser joints	582 gpm
Total Maximum Rate	807 gpm

LPCI capacity is sized to accommodate 3000 gpm leakage at these locations. It is concluded that the reactor vessel internals retain sufficient integrity during the recirculation line break accident to allow reflooding of the inner volume of the reactor vessel.

3.6.3 Earthquake

The seismic loads on the reactor vessel internals are based on a dynamic analysis of a model similar to that shown in Figure III-3-10. Seismic analysis is performed by coupling this lumped mass model of the reactor vessel and internals with the building model to determine the system natural frequencies and node shapes. The relative displacement, acceleration, and load response is then determined by either the time history method or the response spectrum method.

In the time history method, the dynamic response is determined for each node of interest and added algebraically for each instant of time. Resulting response time histories are then examined, and the maximum value of displacement, acceleration, shears, and moments are used for design calculations.

In the response spectrum method, the relative displacements, accelerations, shears, and moments are determined for each node of interest. The root-mean-square of these individual responses are then used for design calculations.

3.6.4 Conclusions

Response analyses of the reactor vessel internals show that deformations are sufficiently limited to allow both adequate control rod insertion and proper operation of the core standby cooling systems. Sufficient integrity of the internals is retained during accident conditions to allow successful reflooding of the reactor vessel inner volume. The analyses considered various loading combinations, including loads imposed by external forces. Thus, safety design bases 3.2a, b, and c are satisfied.

3.7 Inspection and Testing

This USAR section contains historical information as indicated by the italicized text. USAR Section I-3.4 provides a more detailed discussion of historical information. The information being presented in this section as historical has been preserved as it was originally submitted to the NRC in the CNS FSAR.

3.7.1 Inspection and Testing Prior to Commercial Operation

The reactor coolant system, which includes the reactor vessel internals, is thoroughly cleaned and flushed before fuel is loaded initially.

During the preoperational test program, operational readiness tests are performed on various systems. In the course of these tests such reactor vessel internals as the feedwater spargers, the core spray lines, the vessel head cooling spray nozzle, and the standby liquid control system line are functionally tested.

Steam separator-dryer performance tests are made during the startup test program of the first plant to use a specific design concept to determine carry-under and carry-over characteristics. Samples are taken from the inlet and outlet of the steam dryers and from the inlet to the main steam lines at various reactor power levels, water levels, and recirculation flow rates. Moisture carry-over is determined from sodium-24 activity in these samples and in reactor water samples. Carry-under is determined from measured flows, and temperatures are determined by heat balances.

Vibration analysis of reactor vessel internals is included in the design to eliminate failures caused by vibration. When necessary, vibration is measured during startup tests to determine the vibration characteristics of specific reactor internals and the recirculation loops under forced recirculation flow. Vibratory responses are recorded at various recirculation flow rates using strain gages, accelerometers, and linear differential transducers as appropriate.

The vibration analyses and tests are designed to determine any potential, hydraulically induced equipment vibrations and to verify that the structures do not fail because of fatigue. The structures are analyzed for natural frequencies, node shapes, and vibrational magnitudes that could lead to fatigue at these frequencies. With this analysis as a guide, the reactor internals are instrumented and tested to ascertain that there are no gross instabilities. The cyclic loadings are evaluated using, as a guide, the cyclic stress criteria of the ASME Code, Section III. These field tests are only performed on reactor internals that represent a significant departure from design configurations or operating conditions previously tested and found to be acceptable. Field test data are correlated with the analyses to ensure validity of the analytical techniques on a continuing basis.[31]

3.7.2 Inspection and Testing After Commencing Commercial Operation

Quality control methods are used during the fabrication and assembly of reactor vessel internals to assure that the design specifications are met.

The reactor vessel and internals have been designed to assure adequate working space and access for inspection of selected components and locations. Criteria for selecting the components and locations to be inspected are based on the probability of a defect occurring or enlarging at a given location and include areas of known stress concentrations and locations where cyclic strain or thermal stress might occur.

The plant design allows for inservice inspection to the extent practical in accordance with ASME Section XI. The Inservice Inspection (ISI) Program for Cooper Nuclear Station describes the ISI testing that is performed (See Appendix J).

The reactor vessel and internals are in scope for License Renewal per 10 CFR 54.4(a)(1) and (a)(2) and were subject to aging management review.

Aging effects are managed by the following Aging Management Programs: Bolting Integrity (see USAR Section K-2.1.2), BWR CRD Return Line Nozzle (see USAR Section K-2.1.4), BWR Feedwater Nozzle (see USAR Section K-2.1.5), BWR Penetrations (see USAR Section K-2.1.6), BWR Stress Corrosion Cracking (see USAR Section K-2.1.7), BWR Vessel ID Attachment Welds (see USAR Section K-2.1.8), BWR Vessel Internals (see USAR Section K-2.1.9), Inservice Inspection - ISI (see USAR Section K-2.1.19), Reactor Head Closure Studs (see USAR Section K-2.1.32), Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) (see USAR Section K-2.1.37), and Water Chemistry Control - BWR (see USAR Section K-2.1.39). The following Time-Limited Aging Analyses are applicable: Neutron Fluence (see USAR Section K-2.2.1.1) and Metal Fatigue (see USAR Section K-2.2.2.1).

4.0 REACTIVITY CONTROL MECHANICAL DESIGN

4.1 Safety Objective

The safety objective of the reactivity control mechanical design is to provide a means to quickly terminate the nuclear fission process in the core so that damage to the fuel barrier is limited.

4.2 Safety Design Bases

The reactivity control mechanical design shall include control rods and gadolinia burnable poison in selected fuel rods within fuel assemblies. Safety design bases are as follows:

a. The control rods and gadolinia-urania fuel rods shall have sufficient mechanical strength to prevent displacement of their reactivity control material.

b. The control rods shall have sufficient strength and be so designed as to prevent deformation that could inhibit their motion.

c. Each control rod shall have a device to limit its free-fall velocity sufficiently to avoid damage to the nuclear system process barrier by the rapid reactivity increase resulting from a free fall of any single control rod from its fully inserted position to the position where the drive was withdrawn.

4.3 Power Generation Objective

The power generation objective of the reactivity control mechanical design is to provide a means to control power generation in the fuel.

4.4 Power Generation Design Basis

The reactivity control mechanical design shall include reactivity control devices (control rods and gadolinia burnable poison) that contain and position the material that controls the excess reactivity in the core.

4.5 Description

4.5.1 Gadolinia-Urania Fuel Rods

To meet the reactivity control requirements of any core load with excess reactivity, selected fuel rods in each reload bundle incorporate small amounts of gadolinium as burnable poison.[1] Gd_2O_3 is uniformly distributed in the UO_2 and forms a solid solution in the appropriate high enrichment fuel rod. During the fuel cycle the presence of the high cross-section Gd isotopes and the position of the gadolinia-urania fuel rods within the fuel assembly results in relatively low heat generation in those rods.

Addition of small amounts of Gd_2O_3 to UO_2 affects both the conductivity and melting temperatures of the solid solution. Below 1800°C the conductivity is reduced relative to that of pure UO_2, but above 1800°C there is essentially no effect. At no temperature is the conductivity of the solid solution less than the minimum conductivity of pure UO_2. Melting temperature of the solid solution is also below that of pure UO_2. However, the combined effect of these changes is not large enough to cause the fuel rods containing gadolinia to approach a molten condition at any power output that does not cause melting in the highest powered pure UO_2 rods.

4.5.2 Control Blades

The control blades perform the dual function of power shaping and reactivity control. Power distribution in the core is controlled during operation of the reactor by manipulating selected patterns of control blades. The control blades are arranged to counterbalance steam void effects at the top of the core which results in significant power flattening.

The operational lifetime of the control blades is limited by the burnup of boron-10 from neutron absorption, and boron loss due to absorber tube cracking. The nuclear lifetime limit is reached when the loss of total reactivity equals 10% of the associated initial control blade worth. The 10% change in relative reactivity is defined as occurring over a significant section of the control blade. A significant section of the control blade is defined as one-fourth of the active absorber length. The mechanical lifetime limit is defined as the time at which the internal helium pressure from the boron-10 (neutron, alpha) reaction results in stresses in any absorber tube of the control rod reaching the most restrictive design limit.

Based on experimental data, a helium release model is used to correlate the fraction of generated helium released from the boron-carbide with the boron-10 burnup fraction. This model predicts a release fraction that starts at 4 percent for zero boron-10 burnup and increases to approximately 20 percent release at 100 percent boron-10 burnup.

The actual replacement of the original control blades depends on the loss of reactivity control capability and gas pressure buildup and varies among control blades. The marathon design control blades have a higher burnup (lifetime) than the original design blades. The replacement of original control blades by marathon control blades was completed in RE29.

Marathon Control Blades

The marathon control blades (see Figure III-4-1a) consist of a cruciform array of stainless steel tubes loaded with boron-carbide powder capsules, some empty tubes, and hafnium metal filled segments. The control blades are 9.78 inches in total span and are separated uniformly throughout the core on a 12 inch pitch. Each control blade is surrounded by four fuel assemblies.

The marathon control blade main structural member 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, and a segmented vertical cruciform center post. The top handle, bottom casting, and center post are welded into a single unit forming a rigid housing containing the capsules. Rollers at the top and bottom of the control blade guide the blade as it is inserted and withdrawn from the core. The control blades are cooled by the core bypass flow that flows over the stainless steel tubes that house the absorber capsules. Operating experience has shown that control blades constructed as described above are not susceptible to dimensional distortions. Marathon blades used starting with Cycle 24 have spacer pads in lieu of rollers at the top. This design feature was encompassed by the Marathon blade topical report and subsequent approval by the NRC. The spacer pads fall within the same envelope as the rollers at the top.[117]

The marathon design control blade has boron-carbide (B_4C) powder in absorber capsules that are loaded 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 boron is 18.0 percent by weight. Absorber tubes are made of Type 304 stainless steel. Each absorber tube is about 0.3 inches square in outside dimensions. Absorber tubes are sealed by welding each end. The boron-carbide is longitudinally separated into individual compartments by using a number of capsules over the length.

4.5.3 Control Rod Velocity Limiter

The control rod velocity limiter (see Figure III-4-2) 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.

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 over the length of the control rod stroke. 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 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 jetted in a partially reversed direction into water flowing upward in the annulus, a severe turbulence is created, thereby slowing the descent of the control rod assembly to less than 5 ft/sec at 70°F.[33]

4.6 Safety Evaluation

4.6.1 Evaluation of Control Blades and Gadolinia-Urania Fuel Rods

The description of the control blades and gadolinia-urania fuel rods illustrates that they meet the design basis requirements.

4.6.2 Evaluation of Control Rod Velocity Limiter

The control rod velocity limiter limits the free fall velocity of the control rod to a value that cannot result in nuclear system process barrier damage. This velocity is evaluated by the rod-drop accident analysis in Section XIV, "Station Safety Analysis".

4.7 Inspection and Testing

4.7.1 Gadolinia-Urania Fuel Rods

The same rigid quality control requirements observed for standard UO_2 fuel are employed in manufacturing 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 are identified by a stamp on the pellet. The amount of Gd_2O_3 in fuel powder and pellets is verified by testing samples using X-ray fluorescence. Individual rods, which may contain more than one pellet composition, are assembled based on rigid physical and administrative controls, normally including a computer interactive inventory and assembly system with overchecks to assure proper assembly. The Gd_2O_3 content of each fuel rod is verified based on X-ray fluorescence scanning of pellets or fuel rod scanning for Gd_2O_3 content and proper assembly. Correct placement of gadolinia-bearing rods within the fuel assembly is further assured by the extended upper end plug shanks and enrichment control symbols for these rods.

All assemblies and rods for a given project are inspected to assure overall accountability of fuel quantity and rod placement for the project.[1]

4.7.2 Control Blade Absorber Tubes

(1) Material integrity of the tubes and end plug are verified by inspection.

(2) The boron-10 fraction of the boron content of each lot of boron-carbide is verified.

(3) Weld integrity of the finished absorber tube is verified by helium leak-testing.

5.0 CONTROL ROD DRIVE MECHANICAL DESIGN

5.1 Safety Objective

The safety objective of the Control Rod Drive (CRD) mechanical design is to insert the control rods with sufficient speed to limit fuel barrier damage.

5.2 Safety Design Bases

5.2.1 Reactivity Control Mechanical Design

The reactivity control mechanical design shall meet the following safety design bases:

a. Design shall provide for a sufficiently rapid control rod insertion so that no fuel damage results from any abnormal operating transient.

b. Design shall include positioning devices, each of which individually support and position a control rod.

c. Each positioning device shall:

(1) Prevent its control rod from withdrawing as a result of a single malfunction.

(2) Be designed so that no single failure of a component will prevent its control rod from being inserted.

(3) Be individually operated so that a failure in one positioning device does not affect the operation of any other positioning device.

(4) Be individually energized when rapid control rod insertion (scram) is signaled so that failure of power sources external to the positioning device does not prevent other positioning devices' control rods from being inserted.

(5) Be locked to its control rod to prevent undesirable separation.

5.2.2 Scram Discharge Volume Design

a. The scram discharge volume shall have sufficient capacity to receive and contain water exhausted by a full reactor scram without adversely affecting CRD scram performance.

(1) No single active failure of a component or service function shall prevent a reactor scram.

(2) No single active failure shall cause an uncontrolled loss of reactor coolant.

(3) The scram discharge headers shall be sized in accordance with GE-OER-54 and shall be hydraulically coupled to the instrumented volume(s) in a manner to

permit operability of the scram level instrumentation prior to a loss of system function.

The system shall be analyzed for maximum in-leakage to ensure system function is not lost prior to initiation of automatic scram.

(4) Vent and drain valves shall be provided to contain the scram discharge water, with a single active failure in order to minimize operation exposure.

A) Power-operated vent and drain valve shall close under loss of air and/or electric power.

B) Power-operated valve position indication shall be provided in the Control Room.

b. Scram discharge volume level instrumentation shall be provided with automatic scram initiation prior to sufficient volume capacity degradation.

(1) Scram instrumentation shall be capable of detecting water accumulation in the instrumented volume(s) assuming a single active failure in the instrumentation system or plugging of an instrument line.

A) System piping flow path reduction shall be analyzed to assure system reliability and operability under all modes of operation.

(2) Scram instrumentation shall aid an operator in the detection of water accumulation in the instrument volume(s) prior to scram initiation.

(3) Scram discharge system instrumentation shall provide redundancy, to operate reliably under all conditions, and shall not be adversely affected by hydrodynamic forces or flow characteristics.

5.3 <u>Power Generation Objective</u>

The power generation objective of the CRD mechanical design is to position the control rod within the core to control power generation.

5.4 <u>Power Generation Design Basis</u>

The reactivity control mechanical design shall provide for positioning the control rods to control power generation in the core.

5.5 <u>Description</u>

The CRD mechanisms are part of the CRD system. The CRD system hydraulically operates the CRD mechanisms using processed condensate water or demineralized water as hydraulic fluid. The CRD mechanisms operate manually to position the control rods but act automatically to rapidly insert the control rods during abnormal conditions requiring rapid shutdown (scram).

The control rods, CRD mechanisms, and that part of the CRD hydraulic system necessary for scram are designed as Class I (seismic) equipment. The piping and valves in the CRD system that are required to effect a scram and serve as part of the reactor coolant pressure boundary are designed and fabricated to high quality levels as defined in Appendix D(1), "Construction Phase Quality Assurance Program."

5.5.1 Control Rod Drive Mechanisms

The CRD mechanism (drive) used for positioning the control blade in the reactor core is a double-acting, mechanically latched, hydraulic piston cylinder configuration using process condensate water as its operating fluid. (See Figures III-5-1, III-5-2, III-5-3, and III-5-4.) The individual drives are mounted on the bottom head of the reactor pressure vessel. The drives do not interfere with refueling and are operative even when the head is removed from the reactor vessel. The drives are also readily accessible for inspection and servicing. The bottom location makes maximum utilization of the water in the reactor as a neutron shield and gives the least possible neutron exposure to the drive components. Using reactor water from the condensate system as the operating fluid eliminates the need for special hydraulic fluid. Drives are able to utilize simple piston seals whose leakage does not contaminate the reactor water but does cool the drive mechanisms and their seals.

The drives are capable of inserting or withdrawing a control blade at a slow, controlled rate as well as providing rapid insertion during a scram. A mechanism on the drive locks the control rod in 6-in. increments of stroke over the 12-ft length of the core.

A coupling spud at the top end of the drive index tube (piston rod) engages and locks into a mating socket at the base of the control rod. The weight of the control rod is sufficient to engage and lock this coupling. Once locked, the drive and rod form an integral unit that must be manually unlocked by specific procedures before the components can be separated.

The drive holds its control rod in distinct latch positions until the hydraulic system actuates movement to a new position. An alarm annunciates if the withdraw overtravel limit on the drive is reached. Normally, the seating of the control rod at the lower end of its stroke prevents reaching the drive withdraw overtravel limit. If the drive can reach the withdraw overtravel limit, this means the control rod is uncoupled from its drive.

The individual rod indicators, grouped in one control panel display, correspond to relative rod locations in the core. A separate, smaller display is located just below the large display on the vertical part of the bench board. This display presents the positions of the control rod selected for movement and the other rods in the affected rod group. For display purposes the control rods are considered in groups of four adjacent rods centered around a common core volume. Each group is monitored by four LPRM strings (see Section VII-5, "Neutron Monitoring System"). Rod groups at the periphery of the core may have less than four rods. The four rod display shows the positions, in digital form, of the rods in the group to which the selected rod belongs. A white light on the control panel display indicates which of the four rods is the one selected for movement. For further information, see Section VII-7, "Reactor Manual Control System."

Figure III-5-2 illustrates the operating principle of a drive. Figures III-5-3 and III-5-4 illustrate the drive in more detail. The main components of the drive and their functions are described below.

5.5.1.1 Drive Piston

The drive piston is mounted at the lower end of the index tube. This tube functions as a piston rod. The drive piston and index tube make up the main moving assembly in the drive. The drive piston operates between positive end stops, with a hydraulic cushion provided at the upper end only. The piston has both inside and outside seal rings and operates in an annular space between an inner cylinder (fixed piston tube) and an outer cylinder (drive cylinder). Because the type of inner seal used is effective in only one direction, the lower sets of seal rings are mounted with one set sealing in each direction.

A pair of nonmetallic bushings prevents metal-to-metal contact between the piston assembly and the inner cylinder surface. The outer piston rings are segmented step-cut seals with expander springs holding the segments against the cylinder wall. A pair of split bushings on the outside of the piston prevents contact with the cylinder wall. The effective piston area for downtravel, or withdrawal, is approximately 1.2 sq. in. versus 4.0 sq. in. for uptravel, or insertion. This difference in driving area tends to balance the control rod weight and assures a higher force for insertion than for withdrawal.

5.5.1.2 Index Tube

The index tube is a long hollow shaft made of nitrided Type-304 stainless steel. Circumferential locking grooves, spaced every six inches along the outer surface, transmit the weight of the control rod to the collet assembly.

5.5.1.3 Collet Assembly

The collet assembly serves as the index tube locking mechanism. It is located in the upper part of the drive unit. This assembly prevents the index tube from accidentally moving downward. The assembly consists of the collet fingers, a return spring, a guide cap, a collet housing (part of the cylinder, tube, and flange), and the collet piston seals.

Locking is accomplished by six fingers mounted on the collet piston at the top of the drive cylinder. In the locked or latched position the fingers engage a locking groove in the index tube.

The collet piston is normally held in the latched position by a force of approximately 150 lb. supplied by a spring. Metal piston rings are used to seal the collet piston from reactor vessel pressure. The collet assembly will not unlatch until the collet fingers are unloaded by a short, automatically sequenced, drive-in signal. A pressure, approximately 265 psi above reactor vessel pressure, is applied to the collet piston to overcome spring force, slide the collet up against the conical surface in the guide cap, and spread the fingers out so they do not engage a locking groove. The collet piston is nitrided to minimize wear.

A guide cap is fixed in the upper end of the drive assembly. This member provides the unlocking cam surface for the collet fingers and serves as the upper bushing for the index tube. It is nitrided to provide a compatible bearing surface for the index tube.

If reactor water is used during a scram to supplement accumulator pressure, it is drawn through a filter on the guide cap.

5.5.1.4 Piston Tube

The piston tube is an inner cylinder, or column, extending upward inside the drive piston and index tube. The piston tube is fixed to the bottom flange of the drive and remains stationary. Water is brought to the upper side of the drive piston through this tube. A series of orifices at the top of the tube provides progressive water shutoff to cushion the drive piston at the end of its scram stroke.

5.5.1.5 Stop Piston

A stationary piston, called the stop piston, is mounted on the upper end of the piston tube. This piston provides the seal between reactor vessel pressure and the space above the drive piston. It also functions as a positive end stop at the upper limit of control rod travel. A stack of spring washers just below the stop piston helps absorb the final mechanical shock at the end of control rod travel. The piston rings are similar to the outer drive piston rings. A bleedoff passage to the center of the piston tube is located between the two pairs of rings. This arrangement allows seal leakage from the reactor vessel (during a scram) to be bled directly to the discharge line. The lower pair of seals is used only during the cushioning of the drive piston at the upper end of the stroke.

5.5.1.6 Position Indicator Probe

The center tube of the drive mechanism forms a well to contain the position indicator probe. This probe is an aluminum extrusion attached to a cast aluminum housing. Mounted on the extrusion are hermetically sealed, magnetically operated, position indicator switches. Each switch is sheathed in a braided glass sleeve, and the entire probe assembly is protected by a thin-walled stainless steel tube. The switches are actuated by a ring magnet located at the bottom of the drive piston. The drive piston, piston tube, and indicator tube are all of nonmagnetic stainless steel, allowing the individual switches to be operated by the magnet as the piston passes. One switch is located at each position corresponding to an index tube groove, thus allowing indication at each latching point. An additional switch is located at each midpoint between latching points to indicate the intermediate positions during drive motion. Thus, indication is provided for each 3 inches of travel. Duplicate switches are provided for the full-in and full-out positions. One additional switch (an overtravel switch) is located at a position below the normal full-out position. Because the limit of downtravel is normally provided by the control rod itself as it reaches the backseat position, the drive can pass this position and actuate the overtravel switch only if it is uncoupled from its control rod. A convenient means is thus provided to verify that the drive and control rod are coupled after installation of a drive or at any time during plant operation.

5.5.1.7 Flange and Cylinder Assembly

A flange and cylinder assembly is made up of a heavy flange welded to the drive cylinder. A sealing surface on the upper face of this flange makes the seal to the drive housing flange. Teflon coated, stainless steel o-rings are used for these seals. In addition to the reactor vessel seal, the two hydraulic control lines to the drive are sealed at this face. A drive can thus be replaced without removing the control lines, which are permanently welded into the housing flange. The drive flange contains the integral ball, or two-way, check (ball-shuttle) valve. This valve directs either the reactor vessel pressure or the driving pressure, whichever is higher, to the underside of the drive piston. Reactor vessel pressure is admitted to this valve from the annular space between the drive and drive housing through passages in the flange. A screen is provided to intercept foreign material at this point.

Water used to operate the collet piston passes between the outer tube and the cylinder tube. The inside of the cylinder tube is honed to provide the surface required for the drive piston seals.

Both the cylinder tube and outer tube are welded to the drive flange. The tops of these tubes have a sliding fit to allow for differential expansion.

5.5.1.8 Coupling Spud and Lock Plug

The upper end of the index tube is threaded to receive a coupling spud. The coupling (see Figure III-5-1) accommodates a small amount of angular misalignment between the drive and the control rod. Six spring fingers allow the coupling spud to enter the mating socket on the control rod. A plug then enters the spud and prevents uncoupling.

Two means of uncoupling are provided. With the reactor vessel head removed, the lock plug can be raised against the spring force of approximately 50 lb. by a rod extending up through the center of the control rod to an unlocking handle located above the control rod velocity limiter. The control rod, with the lock plug raised, can then be lifted from the drive.

The lock plug can also be pushed up from below, if it is desired to uncouple a drive without removing the reactor pressure vessel head for access. In this case, the central portion of the drive mechanism is pushed up against the uncoupling rod assembly, which raises the lock plug and allows the coupling spud to disengage the socket as the drive piston and index tube are driven down.

The control rod is heavy enough to force the spud fingers to enter the socket and push the lock plug up, allowing the spud to enter the socket completely and the plug to snap back into place. Therefore, the drive can be coupled to the control rod using only the weight of the control rod. However, with the lock plug in place, a force in excess of 50,000 lb. is required to pull the coupling apart.

5.5.1.9 Materials of Construction

Factors that determine the choice of construction materials are discussed below:

The index tube must withstand the locking and unlocking action of the collet fingers. A compatible bearing combination must be provided that is able to withstand moderate misalignment forces. The reactor environment limits the choice of materials suitable for corrosion resistance. The column and tensile loads can be satisfied by an annealed 300 series stainless steel. The wear and bearing requirements are provided by Malcomizing the completed tube. To obtain suitable corrosion resistance, a carefully controlled process of surface preparation is employed.

The coupling spud is made of Inconel 750 that is aged for maximum physical strength and the required corrosion resistance. Because misalignment tends to cause chafing in the semispherical contact area, the entire part is protected by a thin vapor-deposited chromium plating (electrolyzed). This plating also prevents galling of the threads attaching the coupling spud to the index tube.

Inconel 750 is used for the collet fingers, which must function as leaf springs when cammed open to the unlocked position. Colmonoy 6 hard facing provides a long-wearing surface, adequate for design life, to the area contacting the index tube and unlocking cam surface of the guide cap.

A composite material is selected for seals and bushings on the drive piston and stop piston. The material is inert and has a low friction coefficient when water-lubricated.

Because loss of strength is experienced at higher temperatures, the drive is supplied with cooling water to hold temperatures below 250°F. The composite material is relatively soft, which is advantageous when an occasional particle of foreign matter reaches a seal. The resulting scratches in the seal reduce sealing efficiency until worn smooth, but the drive design can tolerate considerable water leakage past the seals into the reactor vessel.

All drive components exposed to reactor vessel water are made of AISI 300 series stainless steel except the following:

1. Seals and bushings on the drive piston and stop piston are a composite material.

2. All springs and members requiring spring action (collet fingers, coupling spud, and spring washers) are made of Inconel 750.

3. The ball check valve is a Haynes Stellite cobalt-base alloy.

4. Elastomeric O-ring seals are ethylene propylene.

5. Collet piston rings are Haynes 25 alloy.

6. Certain wear surfaces are hard faced with Colmonoy 6.

7. Nitriding by a proprietary new Malcomizing process, electrolyzing (a vapor deposition of chromium), and chromium plating are used in certain areas where resistance to abrasion is necessary.

8. The drive piston head is made of Armco 17-4Ph.

Pressure-containing portions of the drives are designed and fabricated in accordance with requirements of Section III of the ASME Boiler and Pressure Vessel Code.

5.5.2 Control Rod Drive Hydraulic System

The CRD hydraulic system (Burns and Roe Drawing 2039) supplies and controls the pressure and flow to and from the CRDs. One supply subsystem supplies water to the hydraulic control units (HCUs) at the correct flow. Each HCU controls the water flow to and from its associated CRD during normal operation (with and without CRD motion) and reactor scram. The water discharged from the drives during a scram flows through the HCUs to the scram discharge volume (SDV). The water discharged from a drive during a normal control rod positioning operation flows through its HCU and into the exhaust header.

The CRD return line (CRDRL) was originally installed to return exhaust water from the CRD movement and water in excess of system requirements to the reactor vessel. This exhaust water is much cooler than reactor water and could have caused cracking in the CRDRL nozzle and on the wall of the reactor vessel beneath the nozzle. Therefore, in 1978 the CRDRL was removed and the vessel wall and nozzle were inspected and found to be in satisfactory condition. The CRDRL nozzle was capped with an inconel cap. The open return line was capped both inside and outside the drywell.[35]

With no return line flow, the higher pressure in the cooling water header caused a reverse flow through the exhaust water header and the HCU 121 withdrawal exhaust valves. Therefore, equalizing valves were installed in 1981 between the exhaust header and the cooling water header, thus reducing the duty on the HCU 121 withdrawal exhaust valves. The carbon steel pipe, valves and flow elements were also replaced with stainless steel components thus eliminating carbon steel corrosion products from the reverse flow and returning the service of the CRD and HCU filters to normal.[36]

Special Test Procedure 81-1 was performed during the 1981 refueling and maintenance outage to demonstrate the CRD System flow capacity. With two CRD pumps in operation, a flow capacity in excess of 160 gpm was achieved at a reactor pressure above 1000 psig. With one CRD pump in operation, a flow capacity of 140 gpm was achieved at a reactor pressure of 990 psig. Based on a generic GE analysis the CNS post-scram decay and residual heat from the fuel would require a makeup injection capability of approximately 135 gpm. Therefore, the CRD System with the CRDRL removed meets the requirements of NUREG 0619, as committed in the CNS response to NRC Generic Technical Activity A-10.[37]

The CRD hydraulic system (Burns and Roe Drawing 2039) also supplies flow to the Cold Reference Leg Continuous Backfill subsystem of the NBI System. See section VII-8.5.2 for further discussion of the Cold Reference Leg Continuous Backfill subsystem. The CRD hydraulic system also supplies water to the RWCU pump seals and the Recirculation pump seals for the purpose of purging the seals.

5.5.2.1 CRD Supply and Discharge Subsystems

The CRD hydraulic supply and discharge subsystems (Burns and Roe Drawing 2039, Reactor Controls Drawing CP001, Sheet 1, and General Electric Drawing 719E580BB) control the pressure and flow required to operate the CRD mechanisms. These hydraulic requirements, identified by the function they perform, are as follows:

1. The CRD HCU scram accumulator must hold a charging pressure of approximately 1400 to 1500 psig and deliver an adequate water volume to its associated CRD underpiston area upon scram. Flow to the scram accumulators is required only during scram reset or system startup.

2. CRD drive pressure of approximately 265 psi above reactor vessel pressure is required for normal CRD movement. A flow rate of approximately 4 gpm during CRD insert movement and 2 gpm during CRD withdraw movement is required.

3. Cooling water to the drives is required at approximately 20 psi above reactor vessel pressure and at a flow rate of 0.20 to 0.34 gpm per drive unit. Cooling water can be interrupted for short periods without damaging the drive.

4. Each scram discharge header pipe is sized to receive and contain all the water discharged by its associated CRDs during a reactor scram including water leakage past the CRD seals. A SDV of at least 3.34 gal. per drive is required. The SDV is normally required to contain air at atmospheric pressure, except during scram when it is filled with water and until the scram signal is cleared.

5. The SDV drain and vent valves must isolate the SDV from secondary containment atmosphere during scram such that no single active failure will cause uncontrolled loss of reactor coolant.

The CRD hydraulic supply and discharge subsystems provide the required functions with the pumps, filters, valves, instrumentation, and piping shown in Burns and Roe Drawing 2039 and described in the following paragraphs.

Redundant components are included, where necessary, to assure continuous system operation if an in-service component requires maintenance. Piping and equipment pressure parts in the CRD hydraulic supply and discharge subsystems are designed and fabricated in accordance with Appendix C.

5.5.2.2 CRD Pumps

One CRD pump pressurizes the system with water from the condensate storage tank or from the Demineralized Water System while the second pump is maintained in standby. Each pump is installed with a suction strainer and filter. A discharge check valve prevents bypassing flow through the nonoperating pump.

If the pump discharge valve is inadvertently closed, the flow is diverted through a minimum-flow bypass connection into the condensate storage tank. Consequently, pump overheating is avoided.

The nuclear system leakage limit is based upon CRD pump capacity. This is discussed further in Section IV-10, "Nuclear System Leakage Rate Limits."

5.5.2.3 Pump Discharge Filters

The on-line filter removes foreign material larger than 50 microns absolute (25 microns nominal) from the hydraulic supply subsystem water. The filter installation allows the addition of 2 ft. minimum thickness of temporary external radiation shielding for personnel protection. A differential pressure indicator and Control Room alarm monitor the filter element as it collects foreign material. A strainer in the filter discharge line protects the HCUs and CRDs in the event of filter element failure.

A check valve assembly is installed downstream of the CRD pump discharge filters. The assembly is Class IS and consists of two check valves in series, an isolation valve, and two test lines. The assembly is installed to minimize potential post LOCA-DBA bypass leakage through the CRD System outside secondary containment.

5.5.2.4 Flow Control Valves

Accumulator charging pressure is established by the discharge pressure of the CRD pump and maintained by the on-line system flow control valve during normal operation. During a scram the scram inlet and outlet valves open and permit the accumulators liquid volume to discharge into the CRDs underpiston area, due to the stored energy in the HCU nitrogen accumulators. The resulting pressure decrease in the charging water header allows the CRD pump to "run out" (i.e., flow rate to increase substantially) into the CRDs via the charging water header. The flow sensing system upstream of the accumulator charging header detects high flow and closes the flow control valve. This action maintains increased flow through the charging water header.

Pressure in the charging header is monitored in the Control Room with a pressure indicator and high pressure alarm.

During normal operation the flow control valve maintains a constant system flow rate. This flow is used for drive flow, drive cooling, and system stability.

5.5.2.5 Drive Water Pressure Control Valve

Drive water pressure required in the drive header is maintained by the drive pressure control valve, which is manually adjusted from the Control Room. A flow rate of approximately 6 gpm (the sum of the flow rate required to insert and withdraw a control rod) normally flows from the drive water pressure staging header through two solenoid-operated stabilizing valves (arranged in parallel) and then through the equalizing valves and into the exhaust water header. The flow through one stabilizing valve equals the drive insert flow; that of the other stabilizing valve equals the drive withdrawal flow. When operating a drive, the required flow is diverted to that drive by closing the appropriate stabilizing valve. Thus, flow through the drive pressure control valve is always constant.

Flow indicators in the drive water header and in the line downstream from the stabilizing valves allow the flow rate through the stabilizing valves to be adjusted when necessary. Differential pressure between the reactor vessel and the drive pressure header is indicated in the Control Room.

5.5.2.6 CRD Cooling Water

The cooling water header is located upstream from the equalizing valves. With the CRDRL removed, the cooling water flow is the same as the CRD System flow. Coordination and adjustment of CRD System flow, charging water pressure, and drive water pressure maintain the cooling water at proper flow and pressure.

The flow through the flow control valve is virtually constant. Therefore, once adjusted, the drive pressure control valve can maintain its required pressure independent of reactor pressure. Changes in setting of the pressure control valves are required only to adjust for changes in the cooling requirements of the drives, as their seal characteristics change with time. A flow indicator in the Control Room monitors cooling water flow. A differential pressure indicator in the Control Room indicates the difference between reactor vessel pressure and drive cooling water pressure. Although the drives can function without cooling water, seal life is shortened by long term exposure to reactor temperatures. The temperature of each drive is monitored by the PMIS, and excessive temperatures are alarmed in the Control Room.

5.5.2.7 Scram Discharge Volumes

Two Scram Discharge Volumes (SDVs) are used to receive the water displaced by the CRD mechanisms during a reactor scram. The SDVs will also limit the loss of reactor water discharged from all the drives during a scram. They are also used to contain the reactor water that leaks past the drives following a scram. These volumes are provided in the scram discharge header.

During normal plant operation each scram discharge header is empty, and the drain and vent valves are open. A scram signal or ARI signal initiates the closure of the drain and vent valves. There are two (2) valves in series on each SDV vent and drain. Position indicator switches on the vent and drain valves actuate valve lights in the Control Room. The limit switches for each series of valves are interconnected to provide positive indication that the vent and drain paths are either "open" or "closed". The close circuit limit switches for each set of valves are connected in parallel, closure of either valve will produce the "close" light indication in the Control Room. This verifies that the vent and drain paths are isolated to prevent a potential uncontrolled loss of reactor coolant. The open circuit limit switches for each set of valves are wired in series, both valves must open in order for the "open" light indication in the Control Room to energize. This verifies that the vent and drain paths are open to prevent an undetected buildup of fluid in the SDV.

During a scram, each SDV partly fills with water discharged from above the drive pistons. While scrammed, the CRD seal leakage from the reactor continues to flow into the SDV until the volume pressure equals the reactor vessel pressure. A check valve in each HCU prevents reverse flow from the volume to the drive. When the initial scram signal is cleared from the Reactor Protection System (RPS), the SDV scram signals are overridden with a key lock override switch, and the SDVs are drained. The key lock switch operation is further discussed in Section VII-2.3.8, "Scram Bypasses".

A test pilot valve allows the SDV vent and drain valves to be tested without disturbing the RPS. If desired for diagnostics or maintenance planning, outlet scram valve leakage can be detected by elevated scram discharge riser temperatures for each HCU.

5.5.2.8 Scram Discharge Instrument Volumes

Each SDV drains through an 8-inch line to a corresponding scram discharge instrument volume (SDIV). The two SDIVs have the instrumentation required to measure the amount of water accumulated in the SDVs. The piping between each SDV and its SDIV is designed to resolve issues documented in IEB 80-17. Adequate hydraulic coupling between the SDV and its SDIV is necessary to ensure that the SDV high level scram is actuated while sufficient free volume remains in the SDV to ensure full control rod insertion for all control rods. The 8-inch line size for this connection between the SDV and SDIV was determined to adequately meet this requirement. Two differential pressure level transmitters and three level-measuring switches set at the same low, intermediate, and high levels on each SDIV prevent operating the reactor without sufficient free volume present to accommodate the water discharged during a scram.

At the first (lowest) level as water begins to fill either SDIV, the lower level set point of the level transmitters will initiate alarms in the Control Room. At the second level, one level switch on each SDIV initiates a rod withdrawal block to prevent further withdrawal of any control rod. At the third (highest) level, a combination of the high level set point of the level transmitters and the two remaining level switches on each SDIV initiates a scram to shut down the reactor while sufficient free volume is still present to receive the scram discharge water. The level transmitters and level switches meet the requirements for redundancy and diversity in that each RPS trip system (A and B) on each SDIV has one level transmitter and one level switch for scram initiation. After a scram and before reactor operation can be resumed, the level transmitters and level switches must be cleared by draining the SDVs.[38] For further information, see Section VII-2, "Reactor Protection System."

5.5.2.9 Hydraulic Control Units

Each hydraulic control unit (HCU) furnishes pressurized water, on signal, to a drive unit. The drive then positions its control rod as required. Operation of the electrical system that supplies scram and normal control rod positioning signals to the HCU is described in Section VII-7, "Reactor Manual Control System."

The basic components in each HCU are manual, pneumatic, and electrical valves; an accumulator; related piping; electrical connections; filters; and instrumentation (see Burns and Roe Drawing 2039, Reactor Controls Drawing CP001, Sheet 1, and Figure III-5-8). These components and their functions are described below.

5.5.2.10 HCU Directional Control Valves

The HCU Insert Supply Directional Control Valve (DCV) is solenoid-operated and opens on CRD insert signal. The valve supplies drive water to the bottom side of the associated CRD piston.

The HCU Insert Exhaust DCV also opens by solenoid on an insert signal. The valve discharges water from above the associated CRD piston to the exhaust water header.

The HCU Withdraw Supply DCV is solenoid-operated and opens on a withdraw signal. The valve supplies drive water to the top of the associated CRD piston. At the beginning of the CRD withdrawal sequence the CRD is given a short insert signal, actuating the DCVs associated with rod insertion to unlatch the collet fingers.

The solenoid-operated Withdraw Exhaust DCV opens on a withdraw signal and discharges water from below the associated CRD piston to the exhaust header. It also serves as the settle valve. The valve opens following any normal drive movement (insert or withdraw) to allow the control rod and its drive to settle back into the nearest latch position.

5.5.2.11 HCU Speed Control Valves

The speed control valves regulate the control rod insertion and withdrawal rates during normal CRD motion. They are manually adjustable flow control valves used to regulate the water flow to and from the volume beneath the main drive piston. A correctly adjusted valve does not require readjustment except to compensate for changes in piston seal leakage.

5.5.2.12 HCU Scram Pilot Valves

The scram pilot valves are operated from the RPS trip system. One scram pilot valve controls both the scram inlet valve and the scram exhaust valve by connecting (to close) or venting (to open) air from the scram air header. The scram pilot valves are three-way, dual solenoid-operated, normally energized valves. On loss of electrical signal to both solenoids on the scram pilot valve, such as the loss of external AC power, the inlet port closes and the exhaust port opens on the scram pilot valve. The scram pilot valves (Burns and Roe Drawing 2039 and Reactor Controls Drawing CP001, Sheet 1) are arranged so that the trip system signal must be removed from both solenoids before air pressure can be discharged from the scram valve operators. This prevents the inadvertent scram of a single drive in the event of a failure of one of the solenoids on the pilot valve.

5.5.2.13 HCU Scram Valves

The scram inlet valve opens to supply pressurized water to the bottom of the CRD piston. This quick opening globe valve is operated by an internal spring and system pressure. It is closed by air pressure applied to the top of its diaphragm operator. A position indicator switch on this valve energizes a light in the Control Room as soon as the valve starts to open.

A scram exhaust valve opens slightly before the scram inlet valve, exhausting water from above the CRD piston. The exhaust valve opens faster than the inlet valve because of greater spring force in the exhaust valve operator. Otherwise the valves are similar.

5.5.2.14 HCU Scram Accumulator

The scram accumulator holds sufficient water volume and stored energy to fully insert a control rod independent of any other source of energy at low reactor vessel pressure. At higher reactor vessel pressures, CRD insertion is assisted on the upper end of the stroke by reactor vessel pressure acting on the drive via a ball check (shuttle) valve (see Section III-5.5.3.3, "Reactor Scram." The accumulator consists of a water volume pressurized by nitrogen. A piston separates the water on top from the nitrogen below. A check valve in the accumulator charging line prevents loss of water pressure in the event supply pressure is lost.

During normal plant operation, the accumulator piston is seated at the bottom of its cylinder. Loss of nitrogen decreases the nitrogen pressure, which actuates a pressure switch and sounds an alarm in the Control Room and illuminates a light on a local accumulator trouble panel.

To ensure that the accumulator is always able to produce a scram, it is continuously monitored for water leakage. A float-type level switch actuates an alarm in the Control Room and illuminates a light on a local accumulator trouble panel if water leaks past the piston barrier and collects in the accumulator instrumentation block.

5.5.2.15 Backup Scram Valves

The two backup scram valves provide a second method of depressurizing the scram air header to all of the scram pilot valves and to the SDV vent and drain valves. Each normally deenergized backup scram valve can vent air from the header to initiate a scram. The backup scram valves are energized from 125 VDC when both RPS trip system A and B are tripped.

5.5.2.16 Alternate Rod Insertion (ARI) Valves

The ARI valves are operated from the ARI ATWS/RPT initiation logic (See Section XIV-5.5.9.3, "Anticipated Transients Without Scram"). The ARI valves provide a means to depressurize the scram air header independently of the scram pilot valves and back-up scram valves. The ARI valves are arranged in an inboard-outboard fashion to allow testing of the ARI logic and valves. The ARI valves are located in the main scram air header, the scram air header branch for the north HCUs and in the scram air header for the south HCUs. Also, a pair of ARI valves (inboard and outboard) are located in the SDV Vent and Drain air header near the north HCUs (see Reactor Controls Drawing CP001, Sheet 1).

5.5.3 Control Rod Drive System Operation

The CRD system performs rod insertion, rod withdrawal, scram, and ARI. These operational functions are described below.

5.5.3.1 Control Rod Insertion

Rod insertion is initiated by a signal from the operator to the insert valve solenoids. This signal causes both insert valves to open. The insert drive valve applies reactor pressure plus approximately 90 psi to the bottom of the drive piston. The insert exhaust valve allows water from above the drive piston to discharge to the exhaust header.

As is illustrated in Figure III-5-3, the locking mechanism is a ratchet type device and does not interfere with rod insertion. The speed at which the drive moves is determined by the pressure drop through the insert speed control valve, which is set for approximately 4 gpm for a shim speed (nonscram operation) or 3 in./sec. During normal insertion, the pressure on the downstream side of the speed control valve is 90 to 100 psi above reactor vessel pressure. However, if the drive slows for any reason, the flow through, and pressure drop across, the insert speed control valve will decrease; the full differential pressure (265 psi) will then be available to cause continued insertion. With 265-psi differential pressure acting on the drive piston, the piston exerts an upward force of 1040 lb.

5.5.3.2 Control Rod Withdrawal

Rod withdrawal is, by design, more involved than insertion. First, the collet fingers (latch) must be raised to reach the unlocked position (see Figure III-5-3). The notches in the index tube and the collet fingers are shaped so that the downward force on the index tube holds the collet fingers in place. The index tube must be lifted before the collet fingers can be released. This is done by opening the drive insert valves (in the manner described in the preceding paragraph) for approximately 1 sec. The withdraw valves are then opened, applying driving pressure above the drive piston and opening the area below the piston to the exhaust header. Pressure is simultaneously applied to the collet piston. As the piston raises, the collet fingers are cammed outward, away from the index tube, by the guide cap.

The pressure required to release the latch is set and maintained at a level high enough to overcome the force of the latch return spring plus the force of reactor pressure opposing movement of the collet piston. When this occurs, the index tube is unlatched and free to move in the withdraw direction. Water displaced by the drive piston flows out through the withdraw speed control valve, which is set to give the control rod a shim speed of 3 in./sec. The entire valving sequence is automatically controlled and is initiated by a single operation of the rod withdraw switch.

5.5.3.3 Reactor Scram

During a scram the scram pilot valves and scram valves are operated as previously described. With the scram valves open, accumulator pressure is admitted under the drive piston, and the area over the drive piston is vented to the SDV.

The large differential pressure (initially approximately 1500 psi and always several hundred psi, depending on reactor vessel pressure) produces a large upward force on the index tube and control rod. This force gives the rod a high initial acceleration and provides a large margin of force to overcome any possible friction. After the initial acceleration is achieved, the drive continues at a nearly constant velocity. This characteristic provides a high initial rod insertion rate. As the drive piston nears the top of its stroke the piston seals close off the large passage (buffer orifices) in the stop piston tube, and the drive slows.

Each drive requires approximately 2.5 gallons of water during the scram stroke. The water capacity in each drive accumulator is adequate to complete a scram in the required time at low reactor vessel pressure. At higher reactor vessel pressures, the accumulator is assisted on the upper end of the stroke by reactor vessel pressure acting on the drive via the ball check (shuttle) valve. As water is forced from the accumulator, the accumulator discharge pressure falls below reactor vessel pressure. This causes a check valve, located in the drive flange, to shift its position to admit reactor pressure under the drive piston. Thus, reactor vessel pressure furnishes the force needed to complete the scram stroke at higher reactor vessel pressures. When the reactor vessel reaches full operating pressure, the accumulator is actually not needed to meet scram time requirements. With the reactor at 1000 psig and the SDV at atmospheric pressure, the scram force without an accumulator exceeds 1000 lb.

The CRD system, with accumulators, provides average scram performances at full power operation, in terms of elapsed time after the opening of the RPS trip actuator (scram signal) for the drives to attain the average scram strokes listed. (See Section 3.1.4 of the CNS Technical Specifications for further information on scram insertion times.)

5.5.3.4 Alternate Rod Insertion[71]

ARI functions as a backup to the RPS scram and will only result in rod motion upon a failure of the RPS initiated scram. The ARI logic will initiate automatically on high reactor vessel pressure, which is set higher than the reactor high pressure scram setting, or low low reactor water level (Level 2), which is lower than the low reactor water level scram setting. ARI can also be initiated using manual pushbuttons. The instruments that initiate ARI also initiate a Reactor Recirculation Pump Trip (RPT) and are described further in Section VII-9.4.4.2. When the ARI logic is initiated, the ARI valves are energized and the two air headers serving the north and south HCUs are depressurized. Upon depressurization of the air header, the scram valves and drives operate as described above during a scram.

5.6 Safety Evaluation

5.6.1 Evaluation of Scram Time

The rod scram function of the CRD system provides the negative reactivity insertion required by safety design basis III-5.2.1.A. The scram time shown in the description is adequate as shown by the transient analyses of Section XIV, "Station Safety Analysis."

5.6.2 Analysis of Malfunctions Relating to Rod Withdrawal

There are no known single malfunctions that cause the unplanned withdrawal of even a single control rod. However, if multiple malfunctions are postulated, the results show that an unplanned rod withdrawal can occur at withdrawal speeds that vary with the combination of malfunctions postulated. In all cases the subsequent withdrawal speeds are less than that assumed in the rod drop accident analysis as discussed in Section XIV, "Station Safety Analysis." Therefore, the physical and radiological consequences of such rod withdrawals are less than those analyzed in the rod drop accident.

5.6.2.1 Drive Housing Fails at Attachment Weld

The bottom head of the reactor vessel has a penetration with an internal nozzle for each CRD location. A drive housing is raised into position inside each penetration and fastened to the top of the internal nozzle with a J-weld. The drive is raised into the drive housing and bolted to a flange at the bottom of the housing. The housing material is seamless, Type-304 stainless steel pipe with a minimum tensile strength of 75,000 psi. The basic failure considered here is a complete circumferential crack through the housing wall at an elevation just below the J-weld.

Static loads on the housing wall include the weight of the drive and the control rod, the weight of the housing below the J-weld, and the reactor pressure acting on the 6-in. diameter cross-sectional area of the housing and the drive. Dynamic loading results from the reaction force during drive operation.

If the housing were to fail as described, the following sequence of events is foreseen. The housing would separate from the vessel. The control rod, drive, and housing would be blown downward against the support structure by reactor pressure acting on the cross-sectional area of the housing and the drive. The downward motion of the drive and associated parts would be determined by the gap between the bottom of the drive and the support structure and by the deflection of the support structure under load. In the current design, the distance that the control rod would travel, gap plus the spring compression deflection, is discussed in Section 8.4. If the collet were to remain latched, no further control rod ejection would occur[40]; the housing would not drop far enough to clear the vessel penetration. Reactor water would leak at a rate of approximately 440 gpm through the 0.06-inch diametral clearance between the housing and the vessel penetration.

If the basic housing failure were to occur while the control rod is being withdrawn (this is a small fraction of the total drive operating time) and if the collet were to stay unlatched, the following sequence of events is foreseen. The housing would separate from the vessel. The drive and housing would be blown downward against the CRD housing support. Calculations indicate that the steady state rod withdrawal velocity would be 0.3 ft/sec. During withdrawal, pressure under the collet piston would be approximately 250 psi greater than the pressure over it. Therefore, the collet would be held in the unlatched position until driving pressure was removed from the pressure-over port.

5.6.2.2 Rupture of Hydraulic Line(s) to Drive Housing Flange

There are three types of possible rupture of hydraulic lines to the drive housing flange: (1) pressure-under line break, (2) pressure-over line break, and (3) coincident breakage of both of these lines.

For the case of a pressure-under line break, a partial or complete circumferential opening is postulated at or near the point where the line enters the housing flange. Failure is more likely to occur after another basic failure wherein the drive housing or housing flange separates from the reactor vessel. Failure of the housing, however, does not necessarily lead directly to failure of the hydraulic lines.

If the pressure-under line were to fail and if the collet were latched, no control rod withdrawal would occur. There would be no pressure differential across the collet piston and, therefore, no tendency to unlatch the collet. Consequently, the associated control rod could not be inserted or withdrawn.

The ball check valve is designed to seal off a broken pressure-under line by using reactor pressure to shift the check ball to its upper seat. If the ball check valve were prevented from seating, reactor water would leak to the atmosphere. Because of the broken line, cooling water could not be supplied to the drive involved. Loss of cooling water would cause no immediate damage to the drive. However, prolonged exposure of the drive to temperatures at or near reactor temperature could lead to deterioration of material in the seals. High temperature would be indicated to the operator by the thermocouple in the position indicator probe. A second indication would be high cooling water flow.

If the basic line failure were to occur while the control rod is being withdrawn and if the collet were to remain open, calculations indicate that the steady state control rod withdrawal velocity would be 2 ft/sec. However, the hydraulic force would not be sufficient to hold the collet open, and spring force normally would cause the collet to latch and stop rod withdrawal.

The case of the pressure-over line breakage considers the complete breakage of the line at or near the point where it enters the housing flange. If the line were to break, pressure over the drive piston would drop from reactor pressure to atmospheric pressure. Any significant reactor pressure (approximately 500 psig or greater) would act on the bottom of the drive piston and fully insert the drive. Insertion would occur regardless of the operational mode at the time of the failure. After full insertion, reactor water would leak past the stop piston seals, the contracting seals on the drive piston, and the collet piston seals. This leakage would exhaust to the atmosphere through the broken pressure-over line. In an experiment to simulate this failure, a leakage rate of less than 2 gpm was measured with reactor pressure at 1000 psi. If the reactor were hot, drive temperature would increase. This situation would be indicated to the reactor operator by the drift alarm, by the fully inserted drive, by a high drive temperature (indicated in the control room), and by operation of the drywell sump pump.

For the simultaneous breakage of the pressures-over and pressure-under lines, pressures above and below the drive piston would drop to zero, and the ball check valve would close the broken pressure-under line. Reactor water would flow from the annulus outside the drive, through the vessel ports, and to the space below the drive piston. As in the case of pressure-over line breakage, the drive would then insert at a speed dependent on reactor pressure. Full insertion would occur regardless of the operational mode at the time of failure. Reactor water would leak past the drive seals and out the broken pressure-over line to the atmosphere, as described above. Drive temperature would increase. Indication in the control room would include the drift alarm, the full-inserted drive, high drive temperature, and operation of the drywell sump pump.

5.6.2.3 All Drive Flange Bolts Fail in Tension

Each CRD is bolted to a flange at the bottom of a drive housing. The flange is welded to the reactor vessel. Bolts are made of High Grade steel, with a minimum tensile strength of 125,000 psi. Each bolt has a minimum allowable load capacity of 15,200 lb. The minimum capacity of the 8 bolts is 121,600 lb. As a result of the reactor design pressure of 1250 psig, the major load on all 8 bolts is 30,400 lb.

If a progressive or simultaneous failure of all bolts were to occur, the drive would separate from the housing. The control rod and the drive would be blown downward against the support structure. Impact velocity and support structure loading would be slightly less than that for drive housing failure, because reactor pressure would act on the drive cross-sectional area only and the housing would remain attached to the reactor vessel. The drive would be isolated from the cooling water supply. Reactor water would flow downward past the velocity limiter piston, through the large drive filter, and into the annular space between the thermal sleeve and the drive. For worst-case leakage calculations, the large filter is assumed to be deformed or swept out of the way so it would offer no significant flow restriction. At a point near the top of the annulus, where pressure would have dropped to 350 psi, the water would flash to steam and cause choke-flow conditions. Steam would flow down the annulus and out the space between the housing and the drive flanges to the atmosphere. Steam formation would limit the leakage rate to approximately 840 gpm.

If the collet were latched, control rod ejection would be limited to the distance the drive can drop before coming to rest on the support structure. There would be no tendency for the collet to unlatch, because pressure below the collet piston would drop to zero. Pressure forces, in fact, exert 1435 lb. to hold the collet in the latched position.

If failure of the drive bolts occurred during control rod withdrawal, pressure below the collet piston would drop to zero. The collet, with 1650 lb. return force, would latch and stop rod withdrawal.

5.6.2.4 Weld Joining Flange to Housing Fails in Tension

The failure considered is a crack in or near the weld that joins the flange to the housing. This weld extends through the wall and completely around the housing. The flange material is forged, Type 304 stainless steel, with a minimum tensile strength of 75,000 psi. The housing material is seamless, Type 304 stainless steel pipe, with a minimum tensile strength of 75,000 psi. The conventional, full-penetration weld of Type 308 stainless steel has a minimum tensile strength approximately the same as that for the parent metal. The design pressure and temperature are 1250 psig and 575°F. Reactor pressure acting on the cross-sectional area of the drive; the weight of the control rod, drive, and flange; and the dynamic reaction force during drive operation result in a maximum tensile stress at the weld of approximately 6000 psi.

In the event that the basic flange-to-housing joint failure occurred, the flange and the attached drive would be blown downward against the support structure. The support structure loading would be slightly less than that for drive housing failure, because reactor pressure would act only on the drive cross-sectional area. Lack of differential pressure across the collet piston would cause the collet to remain latched and limit control rod motion to approximately 3 in. Downward drive movement would be small, therefore, most of the drive would remain inside the housing. The pressure-under and pressure-over lines are flexible enough to withstand the small displacement and remain attached to the flange. Reactor water would follow the same leakage path described above for the flange-bolt failure, except that exit to the atmosphere would be through the gap between the lower end of the housing and the top of the flange. Water would flash to steam in the annulus surrounding the drive. The leakage rate would be approximately 840 gpm.

If the basic failure were to occur during control rod withdrawal (a small fraction of the total operating time) and if the collet were held unlatched, the flange would separate from the housing. The drive and flange would be blown downward against the support structure. The calculated steady state rod withdrawal velocity would be 0.13 ft/sec. Because pressure-under and pressure-over lines remain intact, driving water pressure would continue to the drive, and the normal exhaust line restriction would exist. The pressure below the velocity limiter piston would drop below normal as a result of leakage from the gap between the housing and the flange. This differential pressure across the velocity limiter piston would result in a net downward force of approximately 70 lb. Leakage out of the housing would greatly reduce the pressure in the annulus surrounding the drive. Thus, the net downward force on the drive piston would be less than normal. The overall effect of these events would be to reduce rod withdrawal to approximately one-half of normal speed. With a 560-psi differential across the collet piston, the collet would remain unlatched; however, it should relatch as soon as the drive signal is removed.

5.6.2.5 Housing Wall Ruptures

This failure is a vertical split in the drive housing wall just below the bottom head of the reactor vessel. The flow area of the hole is considered equivalent to the annular area between the drive and the thermal sleeve. Thus, flow through this annular area, rather than flow through the hole in the housing, would govern leakage flow. The housing is made of Type 304 stainless steel seamless pipe, with a minimum tensile strength of 75,000 psi. The maximum hoop stress of 11,900 psi results primarily from the reactor design pressure (1250 psig) acting on the inside of the housing.

If such a rupture were to occur, reactor water would flash to steam and leak through the hole in the housing to the atmosphere at approximately 1030 gpm. Choke-flow conditions would exist, as described above for the flange-bolt failure. However, leakage flow would be greater because flow resistance would be less, that is, the leaking water and steam would not have to flow down the length of the housing to reach the atmosphere. Critical pressure at which the water would flash to steam is 350 psi.

No pressure differential across the collet piston would tend to unlatch the collet; but the drive would insert as a result of loss of pressure in the drive housing causing a pressure drop in the space above the drive piston.

If this failure occurred during control rod withdrawal, drive withdrawal would stop, but the collet would remain unlatched. The drive would be stopped by a reduction of the net downward force acting on the drive line. The net force reduction would occur when the leakage flow of 1030 gpm reduces the pressure in the annulus outside the drive to approximately 540 psig, thereby reducing the pressure acting on top of the drive piston to the same value. A pressure differential of approximately 710 psi would exist across the collet piston and hold the collet unlatched as long as the operator held the withdraw signal.

5.6.2.6 Flange Plug Blows Out

To connect the vessel ports with the bottom of the ball check valve, a hole of 3/4-inch diameter is drilled in the drive flange. The outer end of this hole is sealed with a plug of 0.812-inch diameter and 0.250-inch thickness. A full-penetration, Type 308 stainless steel weld holds the plug in place. The postulated failure is a full circumferential crack in this weld and subsequent blowout of the plug.

If the weld were to fail, the plug were to blow out, and the collet remained latched, there would be no control rod motion. There would be no pressure differential across the collet piston acting to unlatch the collet. Reactor water would leak past the velocity limiter piston, down the annulus between the drive and the thermal sleeve, through the vessel ports and drilled passage, and out the open plug hole to the atmosphere at approximately 320 gpm. Leakage calculations assume liquid flows from the flange. Actually, hot reactor water would flash to steam, and choke-flow conditions would exist. Thus, the expected leakage rate would be lower than the calculated value. Drive temperature would increase and initiate an alarm in the control room.

If this failure were to occur during control rod withdrawal and if the collet were to stay unlatched, calculations indicate that control rod withdrawal speed would be approximately 0.24 ft/sec. Leakage from the open plug hole in the flange would cause reactor water to flow downward past the velocity limiter piston. A small differential pressure across the piston would result in an insignificant driving force of approximately 10 lb. tending to increase withdraw velocity.

A pressure differential of 295 psi across the collet piston would hold the collet unlatched as long as the driving signal was maintained.

Flow resistance of the exhaust path from the drive would be normal because the ball check valve would be seated at the lower end of this travel by pressure under the drive piston.

5.6.2.7 <u>Pressure Regulator and Bypass Valves Fail Closed</u>

By regulating the amount of water from the supply pump, pressure in the drive water header, which is used to supply all drives, is controlled. This control is achieved primarily with the drive water pressure control valves and secondarily with the pressure stabilizing valves. Two drive water control valves are arranged in parallel. One is motor-operated and can be adjusted from the control room. It is normally in service and is partially open to maintain pressure in the header equal to the reactor pressure plus 265 psi. The other valve is hand-operated and normally closed. It can be operated locally when the motor-operated valve is out of service.

The pressure stabilizing valves are solenoid-operated and have built-in needle valves to adjust flow. The two valves are arranged in parallel upstream of the drive water header. One valve is set to bypass 2 gpm; and it closes when any drive is given a withdraw signal and diverts flow to the drive being operated. This serves to maintain a relatively constant header pressure. Similarly, the other valve is set to bypass 4 gpm and closes when an insert signal is given to any drive.

This failure occurs when all these valves are closed and a maximum supply pump head of 1700 psi is in the drive water header. The major portion of the bypass flow normally passes through the motor-operated valve, therefore, closure of this valve is critical.

Because exhaust line pressure is lowest when reactor pressure is zero, this reactor condition is also assumed. If the valve closure failure described above were to occur during control rod withdrawal, calculations indicate that steady state withdrawal speed would be approximately 0.5 ft/sec. or twice normal velocity. A pressure differential of 1670 psi across the collet piston would hold the collet unlatched. Flow would be upward, past the velocity limiter piston, but retarding force would be negligible. Rod movement would stop as soon as the driving signal was removed.

5.6.2.8 Ball Check Valve Fails to Close Passage to Vessel Ports

This failure depends on the ball check valve sealing the passage to the vessel ports during the up-signal portion of the jog withdrawal cycle and the collet remaining unlatched. This is normal withdrawal sequence. Then, if the ball were to move up and foreign material were to jam the ball in the ball cage or prevent it from reseating at the bottom on the seat surface, water below the drive piston would return to the reactor through the vessel ports and the annulus between the drive and the housing. Because the flow resistance of this return path would be lower than normal, the calculated withdrawal speed would be 2 ft/sec. During withdrawal, differential pressure across the collet piston would be approximately 40 psi. Therefore, the collet would tend to latch and would have to stick open before continuous withdrawal, at 2 ft/sec, could occur. Water would flow upward past the velocity limiter piston, generating a small retarding force of approximately 120 lb.

5.6.2.9 Hydraulic Control Unit Valve Failures

Various failures of the valves in the HCU can be postulated, but none could produce differential pressures approaching those described in the preceding paragraphs and none alone could produce a high velocity withdrawal. Leakage through either one or both of the scram valves produces a pressure that tends to insert the control rod rather than to withdraw it. If the pressure in the SDVs should exceed reactor pressure following a scram, a check valve in the line to the scram discharge header prevents this pressure from operating the drive mechanisms.

5.6.2.10 Collet Fingers Fail to Latch

When the drive withdraw signal is removed, the drive continues to withdraw at a fraction of normal speed. Without some initiating signal there is no known means for the collet fingers to become unlocked. If the withdrawal drive valve fails to close following a rod withdrawal, it would have the same effect as failure of the collet fingers to latch in the index tube. Because the collet fingers remain locked until they are unloaded, accidental opening of the withdrawal drive valve normally does not unlock them.

5.6.2.11 Withdrawal Speed Control Valve Failure

Normal withdrawal speed is determined by differential pressures in the drive and is set for a nominal value of 3 in./sec. Withdrawal speed is maintained by the pressure regulating system and is independent of reactor vessel pressure. Tests have shown that accidental opening of the speed control valve to the full-open position produces a velocity of approximately 6 in./sec.

The CRD system prevents rod withdrawal and it has been shown above that only multiple failures in a drive unit and in its HCU could cause an unplanned rod withdrawal.

5.6.2.12 Stuck Control Rod

The CRD is a hydraulically operated unit made up primarily of pistons, cylinders, and a locking mechanism to hold the movable part of the drive at the desired position. The movable part of the drive includes an index tube with circumferential grooves located six inches apart. The collet assembly which serves as the index tube locking mechanism contains fingers which engage a groove in the index tube when the drive is locked in position. In addition to the collet, the collet assembly includes a return spring, a guide cap, a collet retainer tube (collet housing) and collet piston seals. 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. Cracks could occur on the outer surface of the upper thin walled section near the change in wall thickness.

The lower edges of the grooves in the index tube are tapered, allowing index tube insertion without mechanically opening the collet fingers, as they can easily spring outward. If the collet housing were to fail completely at the reported crack location, the coil collet spring could force the upper part of the collet housing and spring retainer upward, to a location where the spring and spring retainer would be adjacent to the collet fingers. The clearance between the collet fingers and the spring when in this location will not permit the collet fingers to spring out of the index tube groove. This would lock the index tube in this position so that the control rod could not be inserted or withdrawn.

One stuck control rod does not create a significant safety concern. However, if a rod cannot be moved and the cause of the failure cannot be determined, the rod could have a failed collet housing. A potentially failed collet housing would be indicative of a problem which could eventually affect the scram capability of more than one control rod. Since the cracks appear to be of a type which propagate slowly, it is highly unlikely that a second control rod would experience a failed collet housing within a short period of time after the first failure. However, since the assumptions utilized in establishing the scram time limits account for only a single stuck control rod, the cause of the inoperability is irrelevant to the Technical Specifications. Technical Specifications define the required actions for inoperable control rods.

If a CRD cannot be moved, the cause of the stuck rod might be a problem affecting other rods. To ensure prompt detection of any additional CRD failures which could prevent movement, the CNS Technical Specification requires surveillance of all partially and fully withdrawn rods when operating above a designated power level if one rod drive cannot be moved.[69][119]

5.6.3 Scram Reliability

High scram reliability is the result of a number of features of the CRD system. These features include:

a. Two sources of scram energy are used to insert each control rod when the reactor is operating--accumulator pressure and reactor vessel pressure.

b. Each drive mechanism has its own scram and pilot valves so only one drive can be affected if a scram valve fails to open. One pilot valve is provided for each drive. This pilot valve is dual solenoid-operated, such that both solenoids must be de-energized to vent the pilot valve to initiate a scram.

c. The RPS and the HCUs are designed so that the scram signal and mode of operation override all others.

d. The collet assembly and index tube are designed so they will not restrain or prevent control rod insertion during scram.

e. Two SDIVs are provided, each with two differential pressure type level transmitters and three float type level switches. This arrangement provides redundancy and diversification of level measurement and will initiate a reactor scram before the SDV accumulates a quantity of water which could interfere with a scram.

f. The two backup scram valves will depressurize the air header to the scram pilot valves if both RPS trip systems trip.

g. An on-line testable ARI system is available in the event that the RPS scram does not occur.

5.6.4 Control Rod Support and Operation

As described above, each control rod is independently supported and controlled as required by safety design bases.

5.7 Inspection and Testing

This USAR section contains historical information as indicated by the italicized text. USAR Section I-3.4 provides a more detailed discussion of historical information. The information being presented in this section as historical has been preserved as it was originally submitted to the NRC in the CNS FSAR.

The control rod drive system is in scope for License Renewal per 10 CFR 54.4(a)(1), (a)(2), and (a)(3) and was subject to aging management review. Aging effects are managed by the following Aging Management Programs: Bolting Integrity (see USAR Section K-2.1.2), External Surfaces Monitoring (see USAR Section K-2.1.14), Oil Analyses (see USAR Section K-2.1.28), and Water Chemistry Control - BWR (see USAR Section K-2.1.39). The following Time-Limited Aging Analyses are applicable: Metal Fatigue (see USAR Section K-2.2.2.1).

5.7.1 Development Tests

The development drive (one prototype) testing to date included more than 5000 scrams and approximately 100,000 latching cycles. One prototype was exposed to simulated conditions for 5000 hours. These tests demonstrated the following:

a. The drive easily withstands the forces, pressures, and temperatures imposed.

b. Wear, abrasion, and corrosion of the nitrided Type 304 stainless parts are negligible. Mechanical performance of the nitrided surface is superior to that of materials used in earlier operating reactors.

c. The basic scram speed of the drive had a satisfactory margin above minimum plant requirements at any reactor vessel pressure.

d. Usable seal lifetimes in excess of 1000 scram cycles can be expected.

5.7.2 Factor Quality Control Tests

Quality control of welding, heat treatment, dimensional tolerances, material verification, and similar factors is maintained throughout the manufacturing process to assure reliable performance of the mechanical reactivity control components. Some of the quality control tests performed on the control rods, CRD mechanisms, and HCUs are listed below.

a. Control blade absorber tube tests:

(1) Material integrity of the tubing and end plug is verified by ultrasonic inspection.

(2) The boron-10 fraction of the boron content of each lot of boron-carbide is verified.

(3) Weld integrity of the finished absorber tubes is verified by helium leak-testing.

b. CRD mechanism tests:

(1) Pressure welds on the drives are hydrostatically tested in accordance with ASME codes.

(2) Electrical components are checked for electrical continuity and resistance to ground.

(3) Drive parts that cannot be visually inspected for dirt are flushed with filtered water at high velocity. No significant foreign material is permitted in effluent water.

(4) Seals are tested for leakage to demonstrate correct seal operation.

(5) Each drive is tested for shim motion, latching, and control rod position indication.

(6) Each drive is subjected to cold scram tests at various reactor pressures to verify correct scram performance.

c. HCU tests:

(1) Hydraulic systems are hydrostatically tested in accordance with USAS B31.1.0.

(2) Electrical components and systems are tested for electrical continuity and resistance to ground.

(3) Correct operation of the accumulator pressure and level switches is verified.

(4) The unit's ability to perform its part of a scram is demonstrated.

(5) Correct operation and adjustment of the insert and withdrawal valves is demonstrated.

5.7.3 Operational Tests

After installation all rods, HCU, and drive mechanisms are tested through their full range for operability.

During normal operation, each time a control rod is withdrawn a notch, the operator can observe in-core monitor indications to verify that the control rod is following the drive mechanism. All control rods that are partially or fully withdrawn from the core are tested for rod-following by inserting the rod at least one notch while the operator observes in-core monitor indications.

To make a positive test of the control-rod-to-control-rod-drive coupling integrity, the operator can withdraw a control rod to the end of its travel and then attempt to withdraw the drive to the overtravel position. Failure of the drive to reach the overtravel position demonstrates rod-to-drive coupling integrity.

Hydraulic supply subsystem pressures can be observed from instrumentation in the control room. Scram accumulator pressures can be observed on the nitrogen pressure gages.

6.0 NUCLEAR DESIGN

6.1 Safety Objective

The safety objective of the nuclear design is to determine reactor-operating limits. Limits on plant operation are established to assure the plant operates under conditions which do not pose an undue risk to the health and safety of the public throughout the life of the core.

6.2 Safety Design Bases

1. Fuel nuclear design shall provide negative reactivity feedback that is sufficient, in combination with other station systems, to prevent fuel damage as a result of any abnormal operational transient (see Section XIV, "Station Safety Analysis").

2. Fuel nuclear design shall exhibit such nuclear characteristics as required to assure that: (1) the nuclear system has no inherent tendency toward divergent or limiting cycle operation and (2) power oscillations which could result in conditions exceeding specified acceptable fuel design limits are not possible or can be readily detected and suppressed.

3. Fuel nuclear design shall limit the excess reactivity of the core sufficiently to assure that reactivity control systems are capable of making the core subcritical at any time with the control rod of highest worth fully withdrawn.

6.3 Power Generation Objective

The objectives of the fuel nuclear design are as follows:

1. To attain rated power generation from the nuclear fuel for a given period of time.

2. To allow normal power operation of the nuclear fuel without sustaining fuel damage.

6.4 Power Generation Design Bases

1. Fuel nuclear design shall provide sufficient excess reactivity so that the desired cycle thermal generation can be achieved without resorting to excessive coastdown.

2. Fuel nuclear design shall provide sufficient negative reactivity feedback to facilitate normal maneuvering and control.

3. Fuel nuclear design shall, in combination with reactivity control systems, allow continuous, stable regulation of core excess reactivity.

6.5 Nuclear Requirements

The following nuclear requirements are established for systems and equipment other than the fuel itself. The fuel nuclear design is compatible with these requirements.

6.5.1 Control Rods

The control rods shall be of such number and reactivity worths that the insertion of all but the control rod of highest worth is sufficient to make the fuel subcritical under the most reactive condition of the nuclear system. The control rod of highest worth shall be considered fully withdrawn.

6.5.2 Reactor Manual Control System

1. Control rod operating patterns and withdrawal sequences shall be specified so that control rod worths are low enough to prevent damage to the nuclear system process barrier as a result of any single control rod dropping from the fully inserted position to the fully withdrawn position.

2. Control rod withdrawal increment (notch) sizes shall be limited so that rod movement of one notch does not result in less than a 20-second reactor period.

6.5.3 Standby Liquid Control System

The standby liquid control system shall have sufficient reactivity characteristics that it is capable of bringing the reactor from full power to a cold shutdown condition at any time in core life. (See Section III-9, "Standby Liquid Control System".)

6.6 Fuel Nuclear Characteristics

The station utilizes a light-water moderated reactor, fueled with slightly enriched uranium dioxide. At operating conditions the moderator boils, producing a spatially variable density of steam voids within the core. The use of a water moderator produces a neutron energy spectrum from which the fissions are produced principally by thermal neutrons.

The presence of uranium-238 in uranium dioxide fuel leads to the production of significant quantities of plutonium during core operation. This plutonium contributes both to fuel reactivity and power production of the reactor. In addition, direct fissioning of uranium-238 by fast neutrons yields approximately 7% of the total power and contributes to an increase of delayed neutrons in the core. The uranium-238 contributes a strong negative Doppler coefficient to reactivity, which improves the inherent response of the reactor and limits the peak power in excursions. The strong negative void reactivity effect contributes to the overall station xenon stability and to the damping of xenon oscillation effects. The reactor is designed to achieve a projected average discharge exposure for each cycle. In reactivity level and reactivity coefficients, the fuel is approximately the same as in other operating General Electric reactors.

6.6.1 Analytical Methods

A lattice physics software code (GESTARII[113])is used to generate few group neutron diffusion constants for use in calculation of relative fuel rod powers within assemblies at various operating conditions and for calculation of three-dimensional reactor power distributions. Local fuel rod powers are calculated for an extensive combination of parameters including fuel and moderator temperatures, burnup, steam voids, and the presence or absence of adjacent control rods. These few group calculations are performed over either single element cells or groups of four assemblies characteristic of repeating arrays in the loaded reactor core.

The nuclear characteristics of the individual lattices within a fuel bundle are based on independent infinite lattice calculations. These calculations fall into three areas: neutron energy spectrum, spatial flux distribution, and isotopic depletion.

The neutron spectrum is evaluated over the complete energy range using many energy groups. In the fast range, the collision probability technique is used. In the resonance range, the B1 method is used with resonance integrals determined by the intermediate resonance method. The thermal spectrum is evaluated using integral transport theory.

The finite multigroup spectrum is then collapsed to three energy groups for calculation of neutron spatial distributions. The spatial distributions are calculated using a two-dimensional transport-corrected diffusion code with averaged three-group cross sections. Important lattice parameters such as K-infinity and peaking factors are also calculated with this code.

The resulting three-group neutron fluxes and the associated nuclear parameters are used, in turn, in an exposure analysis to evaluate isotopic changes due to irradiation. The complete multigroup neutron spectrum calculations are repeated at selected exposure points to generate exposure dependent lattice parameters.

Power and flux distributions, infinite multiplication factors, material and flux weighted cross sections are calculated using integral transport theory on fuel assembly cells and arrays of fuel assembly cells. Burnup calculations are performed by integrating the equations describing the fuel depletion process with spatial neutron flux and energy distributions typical of reactor operating conditions. At selected burnup intervals, the nuclide concentrations are used to recalculate revised flux and material weighted cross sections with the lattice model and these are recycled through two dimensional diffusion theory.

A large capacity, nodal, three dimensional boiling water reactor simulation code which provides for representation and calculation of spatially varying voids, control rods, burnable poisons and other variables is used to compute three-dimensional power distributions, exposure and reactor thermal-hydraulic characteristics at the beginning of core life and as burnup progresses. In addition, it can serve to determine control rod strategy through life, power response to changes in core flow and to calculate assembly as well as core exposure.

6.6.2 Reactivity Control

The excess reactivity designed into the core is controlled by a control rod system supplemented by gadolinia-urania burnable poison rods. The average fuel enrichment value chosen to provide excess reactivity in the fuel assemblies is sufficient to overcome the neutron losses due to core neutron leakage, moderator heating and boiling, fuel temperature rise, equilibrium xenon and samarium poisoning, plus an allowance for fuel depletion. Following each cycle, additional new fuel is added to achieve a planned fuel cycle. During fuel burnup, control rods are used in part to counteract the power distribution effect of steam voids as indicated by the in-core flux monitors.

Taken together, the control rod and void distributions can be used to flatten gross power beyond that possible in a nonboiling core. The design provides considerable flexibility to control the gross power distribution. This permits control of fuel burnup and isotopic composition throughout the core to the extent necessary to counteract the effect of voids on the axial power distribution at the end of a fuel cycle when few control rods remain in the core together with burnable poison. The negative reactivity worth of the gadolinia containing fuel rods decreases with the conversion of the gadolinia high cross section isotopes Gd^{155} and Gd^{157} in a nearly linear manner such that it closely matches the depletion of fissile material. Hence very little rod motion results during full power equilibrium operation until the end of cycle approaches.

The control rods are designed to provide adequate control of anticipated maximum excess reactivity. Since fuel reactivity is a maximum and control strength is a minimum at ambient temperature, the shutdown capability is evaluated assuming a cold, xenon-free core. Safety design basis 3 requires that the core in its maximum reactivity condition be subcritical with the control rod of highest worth fully withdrawn (often referred to as a "stuck rod") and all other rods fully inserted. This basis is established to provide a substantial shutdown margin with all rods in and also to facilitate maintenance and testing of control rod drives.

To assure that the safety design basis 3 is satisfied, an additional design margin is adopted: k_{eff} shall be less than 0.99 with the rod of highest worth fully withdrawn. Thus the safety design bases requires that k_{eff} is equal to or less than 0.99, whereas the limiting condition for operation is that k_{eff} is less than 1.0 with the rod of highest worth fully withdrawn. This limit allows control rod testing at any time in core life and assures that the reactor can be made subcritical by control rods alone. In addition to the control rod shutdown requirements, the Standby Liquid Control (SLC) System provides sufficient reactivity control to shutdown the reactor at any time independent of control rod action.

Analyses of the shutdown margin as a function of fuel exposure indicate that the minimum shutdown margin with one stuck rod satisfies the shutdown margin safety design bases. The variation in k_{eff} is shown in Figure III-6-1.

6.6.3 Refueling Cycle

Each batch size has been varied to maintain a given refueling interval. The flexibility of the boiling water reactor core design permits variation of the intervals between refuelings by varying the refueling batch size.

6.6.4 Control Rod Worth

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

The fractional control rod density versus average moderator density for the beginning-of-life conditions is shown in Figure III-6-2. This density distribution is based on control rod patterns that shape the power distribution and also achieve minimum rod worth. The opposite extreme is to withdraw a cluster of rods in the center of the core. Analyses indicate that criticality is approached after the withdrawal of two adjacent control rods at cold conditions and approximately five adjacent control rods at the hot standby conditions (operating moderator temperature with no voids).

Distributed control rod patterns are used. The operating procedures to accomplish such patterns are implemented by the rod worth minimizer (RWM) program of the process computer, which prevents rod withdrawals yielding a rod worth greater than permitted by the preselected rod withdrawal pattern. The RWM program is described in Section VII-16, "Process Computer System."

Figure III-6-3 shows the maximum of three classes of rod worths as a function of moderator density for fresh fuel. The upper curve of the figure is an envelope of the maximum possible rod worths in extremely abnormal unconstrained control rod patterns. These patterns are not allowed by operating procedures. The lower curve is an envelope of the worth of rods which are scheduled for withdrawal in a normal sequence, with the peak value of less than 0.01 Δk. Only a few rods have worths as high as the envelope; typical values are about half the magnitude of the envelope. The central curve is an envelope of the maximum worths existing for out-of-sequence rods fully inserted in the core during normal withdrawal sequences.

Operating procedures enforced by the RWM prevent the withdrawal of rods with worths higher than the lower curve of Figure III-6-3. Only a few of this class of rods are on the envelope, with typical values less than half the magnitude of the envelope. Figure III-6-4 shows the maximum possible control rod worths, achievable only by very abnormal patterns, as a function of core power at rated temperature.

6.6.5 Reactivity Coefficients

The inherent dynamic behavior of the core is characterized in terms of: (a) fuel temperature or Doppler coefficient, (b) moderator void coefficient, and (c) moderator temperature coefficient.

Doppler reactivity feedback occurs simultaneously with a change in fuel temperature and opposes the power change that caused it; it contributes to system stability. Since the Doppler reactivity opposes load changes, it is desirable to maintain a large ratio of moderator void to Doppler coefficient for optimum load-following capability. The boiling water reactor has an inherently large moderator-to-Doppler coefficient ratio which permits use of coolant flow rate for load following.

In a boiling water reactor, the void coefficient is of primary importance during operation at power. Nuclear design requires the void coefficient inside the fuel channel to be negative. The negative void reactivity coefficient provides an inherent negative feedback during power transients.

Because of the predominant effects of control rods on the void and temperature reactivity coefficients, the coefficients become less negative with fuel exposure and reach their least negative value at the end of each operating cycle when the control rods are fully withdrawn. The void coefficient of reactivity of the critical lattice with the control rods withdrawn is negative. The reactor is designed so that the overall power reactivity coefficient is always negative during the reactor lifetime; however, the moderator temperature coefficient may be small and positive in the cold condition near the middle to end of cycle at the higher exposure levels.

6.6.5.1 Doppler Coefficient

In uranium dioxide fuel, the Doppler coefficient provides instantaneous negative reactivity feedback to any fuel temperature rise, on either a gross or local basis. The magnitude of the Doppler coefficient is inherent in the fuel design and does not vary significantly among various light-water moderated uranium dioxide fuel designs having low fuel enrichment.

The Doppler reactivity effect, which results from the relative increase of the neutron absorption in uranium-238 with an increase of the uranium dioxide temperature, is of considerable importance to plant dynamics and safety. This reactivity effect occurs without significant time lag, while heat conduction to water and the subsequent formation of steam voids must await conduction of heat through the fuel material. During a rapid increase in power, the uranium dioxide temperature increases. The heat conduction mechanism in the fuel has a large time constant which results in a relatively flat temperature distribution across the fuel pellets. Under these conditions, the Doppler effect can remove more than 3 percent reactivity before significant fuel melting is experienced. Consequently, the Doppler effect exerts a strong, prompt, negative shutdown mechanism during rapid transients and contributes considerably to the safety of the plant. The overall Doppler reactivity coefficient is shown in Figure III-6-5. The coefficient varies approximately as the inverse square root of fuel temperature, and becomes more negative as moderator density is decreased by temperature and voids.

$$I = I_{\circ} \exp\left[\alpha \left(T - T_{\circ} \right)\right] + 2.0$$

$$I_{\circ} = 30 \sqrt{S/M + 0.077}$$

$$\alpha = 0.00696 - 0.000262 \ (M/S),$$

where

I = Uranium-238 resonance integral,
T = Effective fuel temperature (°K), and
S/M = Effective surface-to-mass ratio for fuel rod, cm^2/gm.

The net effect is that the Doppler coefficient becomes slightly more negative from the cold to the hot operating condition. Because of the buildup of plutonium-240 the Doppler coefficient is increased by approximately 15 percent as exposure progresses (see Figure III-6-6).

Illustrated in Figure III-6-7 is the total Doppler reactivity defect (the negative integrated Doppler reactivity coefficient) existing in the core under normal steady-state operating conditions up to 110% of rated power. This curve includes the effects on the Doppler reactivity defect of both fuel temperature and steam voids characteristic of normal operation.

The total Doppler availability under normal conditions is included in Figure III-6-8. Doppler defects are shown for adiabatic fuel heating transients under conditions of hot standby, normal operating and maximum void conditions. Fuel temperatures on the abscissa represent effective average fuel temperatures in the core. This shows that substantially more Doppler negative reactivity is available than is required to terminate an excursion caused by removal of any single control rod from a normal pattern.

6.6.5.2 Moderator Temperature and Void Coefficients

Because of the large negative moderator coefficients of reactivity, the BWR has a number of inherent advantages, such as: (1) the use of coolant flow as opposed to control rods for load following, (2) the inherent self-flattening of the radial power distribution, (3) the ease of control, and (4) spatial xenon stability.

These coefficients are partial derivatives of the infinite multiplication factor, neutron leakage, and control system worth with respect to the variables of temperature or void content with the reactor near critical. Mathematically these coefficients are represented as follows:

Temperature Coefficient

$$\frac{1}{k_{eff}}\frac{dk_{eff}}{dT} = \frac{1}{k\infty}\frac{dk\infty}{dT} \quad \frac{C\frac{dL}{dT}}{1-CL} - \left[- \frac{M^2\frac{dB_g^2}{dT} + B_g^2\frac{dM^2}{dT}}{1 + M^2 B_g^2} \right], \text{ and}$$

Void Coefficient

$$\frac{1}{k_{eff}}\frac{dk_{eff}}{dV} = \frac{1}{k\infty}\frac{dk\infty}{dV} - \frac{C\frac{dL}{dV}}{1-CL} - \left[- \frac{M^2\frac{dB_g^2}{dV} + B_g^2}{1 + M^2 B_g^2}\frac{dM^2}{dV} \right]$$

where

T	=	Average moderator temperature,
V	=	In-channel void fraction,
C	=	Lattice constant,
L	=	Diffusion length,
M^2	=	Migration area,
B_g^2	=	Geometric buckling,
$K\infty$	=	Infinite multiplication factor, and
K_{eff}	=	Effective multiplication factor.

Early in core life the control density (or lattice constant C) is highest and the control terms

$$\left[C\frac{dL/dT}{1-CL} \quad \text{and} \quad C\frac{dL/dV}{1-CL} \right]$$

contribute a strong negative effect to the coefficients. Any decrease in moderator density increases neutron leakage from the region of the disturbance and enhances the worth of the control rods. These two contributions are always negative everywhere in the core and monotonically increase in magnitude with moderator density reduction.

For increasing fuel exposures the control density is decreased and the $k\infty$ term has a slight positive trend. The major factor in the reduction of magnitude of the coefficients with fuel exposure is the decrease in the control density. This positive trend causes the moderator temperature coefficient to approach zero and it may become slightly positive at room temperature for the higher exposure levels; however, the void coefficient remains strongly negative throughout core lifetime.

The fuel channel surrounding a fuel bundle can cause a local spatial dependence in the coefficients. Within the fuel assembly a condition of undermoderation exists under all conditions, and the local component is negative. External to the channel, in the surrounding water gaps, the local component of the coefficient is slightly positive from the cold ambient condition through the lower part of the heatup range. In the power range the local components of the coefficients are everywhere negative.

While the $k\infty$ term has a slight positive trend with exposure, the control term is by far the major factor in the reduction of magnitude of the coefficient with fuel exposure. In a gross core power disturbance the $k\infty$ and control terms dominate since leakage is small. In a disturbance such as a control rod withdrawal, the control term in that region is zero but the material buckling and, therefore, leakage from the disturbed zone is large.

Near the middle to end of core life when reactivity control is at a minimum, a gross core transient that is sufficiently slow to allow water gaps to equilibrate with the flowing coolant (tens of seconds) may, below rated operating temperature, result in a slight positive moderator reactivity effect. If the transient accelerates so that steam voids are produced inside the channel, an immediate negative $k\infty$ contribution results such that the total moderator coefficient is negative. Thus, in those regions of the core where rapid moderator density changes can occur, density coefficients are designed to be always negative.

The void coefficient of reactivity for beginning of life is shown in Figure III-6-9. The void coefficient satisfies safety design basis 1 in that it remains strongly negative throughout the core life.

Over a wide range of core designs and core variables it has been found that the moderator temperature coefficient is not significant in terms of the overall plant safety.

Whenever the plant is producing significant power the moderator is at saturated conditions. Therefore, any additional energy input to the moderator during transients will result in the formation of additional voids within the core without any change in moderator temperature. Under such conditions the moderator void coefficient, which is always strongly negative, is controlling and the temperature coefficient is not significant.

During very fast transients as may result from a control rod withdrawal accident from zero power on a control rod drop accident, the Doppler coefficient is controlling because it is effective without significant time delay and limits any excursion before the moderator coefficients which are delayed can be significant. The delay in the effect of the moderator coefficients result from the time required for the transport of the energy released in the fuel to the moderator.

6.6.5.3 Power Coefficient

The power coefficient is a combination of the Doppler and void reactivity coefficients in the power operating range. The power coefficient is not specifically calculated for reload cores. The Doppler and void coefficients are unique for each core and are provided in the Supplemental Reload Licensing Report.

6.6.6 Xenon Transients

The maximum xenon reactivity buildup upon shutdown from full power and the rate of reactivity burnout on return to full power from maximum shutdown xenon buildup are calculated for both the beginning-of-life and the end-of-cycle reactor conditions. It is assumed that the average core exposure at end of cycle is 10,000 MWd/T for this analysis. In the analysis, the maximum rate of change of reactivity is obtained by assuming an instantaneous return to full power. The results of these calculations are shown in Figure III-6-10 for the beginning-of-life condition.

From this analysis it was determined that the maximum rate of reactivity addition due to the burnup of xenon was +0.00010 (Δk/k)/min. Assuming a control rod worth of 0.001 Δk/k with an insertion rate of 3 in./sec., the reactivity addition due to the control rod insertion is 0.00125 (Δk/k)/min. Therefore, even a very weak control rod can more than compensate for the reactivity addition due to xenon burnup. The Standby Liquid Control System which is provided as a backup to the control rods (see Section III-9, "Standby Liquid Control System") is more than adequate to compensate for the reactivity addition due to xenon burnout. With an injection rate for boron of 7 ppm/min., the reactivity insertion rate of the liquid control systems is −0.0015 (Δk/k)/min. Since the design injection rate for the boron is 7 to 50 ppm/min., there is no problem in controlling the reactivity addition due to xenon burnup with the Standby Liquid Control System.

Perturbations of reactor power level can result in shifts of power distribution in the core due to the effects of xenon variations. The inherent nuclear characteristics of the boiling water core lead to strong damping of such disturbances. Such damping is provided by the power coefficient of the operating core. The power coefficient is an inherent reactivity coefficient that opposes changes in power level.

Analytical studies indicate that for large boiling water reactors, underdamped, unacceptable power distribution behavior could occur with power coefficients more positive than about -0.01 Δk/k/ ΔP/P. In this case, the power coefficient at full power is well within the range required for adequate damping of spatial xenon disturbances. Operating experience has shown large boiling water reactors to be inherently stable against xenon-induced power instability. Large load changes have resulted in highly damped changes in the vertical direction only. The vertical power distribution transients are self-damped, with the distribution attaining a steady-state condition about 15 hours after the initiating disturbance.

The model which is used to predict stability against xenon-induced power oscillations has been based on the Randall and St. John model expansion method.[46] This basic model of Randall and St. John has been extended to include the effects of temperature and space dependence of xenon oscillations.

The effects of temperature and space are given in reports by Chernick, Lellouche, and Wollman.[47]

The important assumptions in the development of this model are:

a. One-group diffusion theory is adequate for analyzing xenon oscillations. Since xenon oscillations occur primarily in large thermal reactors, the assumption of monoenergetic diffusion theory is reasonably valid.

b. Delayed neutrons are neglected. Since the period of the xenon oscillations is of the order of hours, the effect of neglecting delayed neutrons is small.

c. Linearized first order perturbation theory is adequate. Any use of linearized equations coupled with first order perturbations implies small perturbations for consistency, and determines the response of the system to small changes. Near the oscillation threshold the assumption of small perturbations is valid to determine the response of the reactor.

d. Mode coupling is negligible. Mode coupling arises from the expansion of the perturbation amplitudes in orthogonal eigen-functions of the unperturbed boundary value problem and the multiplication by the orthogonal eigen-functions followed by integration over the volume. The cross mode terms are of the form

$$\int_{\upsilon \neq j} \varnothing g_{\upsilon}\, g_j dV,$$

and are small because of the orthogonality of the $g\upsilon$'s. They are zero if the flux is flat; furthermore, Canosa[48] has shown that their neglect has little effect in the flux ranges of interest for boiling water reactor design.

e. Neutron absorption in iodine is neglected. The assumption that the absorptions in iodine are negligible corresponds closely to the physical situation; however, it affects the normalization factor to be used in the xenon and iodine governing differential equations. The estimated accuracy in determining threshold fluxes, oscillation periods, and damping ratios is of the order of 10%.

The important parameters used to predict stability against xenon-induced power oscillations are the core length-to-diameter (L/D) ratio, flux or power level, power coefficient, percent of power flattening, and migration area or void transportation in axial direction. The relative stability of any given reactor with respect to power disturbance effects depends on the length-to-diameter (L/D) ratio of the core. Shown in Figure III-6-11 is the relative stability of the radial, azimuthal, and axial power distributions as a function of L/D ratio. Figure III-6-11 shows that axial and azimuthal oscillations have a much higher probability of occurrence than radial oscillations for the reactor. This plot only gives information regarding the relative stability of the various power distributions for a given core. The effect of the flux or power level on the stability of the axial power distribution is shown in Figure III-6-12. These data, which were adopted from Lellouche,[47] include the effect of void transport in the axial direction. Figure III-6-12 shows that the reactor is well damped, and that there is a factor of four between the threshold power coefficient in which xenon oscillations would be induced and the calculated power coefficient at the end of the reactor cycle.

Flux flattening tends to reduce the stability of the various geometrical modes. This is due to the fact that the buckling differences, and consequently the criticality differences between the harmonics, are reduced as the flux shape is flattened. This flux flattening may cause the radial power distribution to be subject to azimuthal oscillations (a rotating edge-to-edge tilt) which are less well damped than the axial oscillations. Shown in Figure III-6-13, which is based on work by Randall and St. John,[46] are the critical power coefficients where the onset of neutral oscillation occurs as a function of flux flattening. From this plot it is seen that the reactor is well damped even when the flat zone is 100% of the radius and demonstrates that the stability requirements for the plant are satisfied.

6.7 Safety Evaluations

The analyses presented in Section III-6.6 show that the safety design bases are satisfied in conjunction with the nuclear design requirements of Section III-6.5. Adequate protection is provided for the cladding and the nuclear system process barrier. The nuclear requirements for reactivity control systems and the settings of the reactor protection system are primarily associated with limitations on the levels and rates of change of reactivity, power, and temperature. Normal plant operation is conducted at rates and values of these parameters such that reactor transients are readily observable and controllable by plant personnel.

The reactor protection system responds to some abnormal operational transients by initiating a scram. The reactor protection system and the control rod drive system act quickly enough to prevent the initiating disturbance from causing fuel damage. The scram reactivity curve used in the reactivity excursion analyses is shown in Figure III-6-14. The scram reactivity for termination of these abnormal operational transients is shown in Figure III-6-15. Abnormal operational transients are evaluated in Section XIV, "Station Safety Analysis." No fuel damage results from any abnormal operational transient.

The specified rod withdrawal sequence programmed into the RWM maintains rod worth at acceptable low values to minimize the consequence of a reactivity accident. At any specified reactor state, peak enthalpies for rod removal accidents vary directly with rod worths; it is the peak enthalpy which is the best measure of the consequences of a reactivity accident. The uranium dioxide vapor pressure data of Ackerman[49] and the interpretation of all the available tests in the TREAT facility of Argonne National Laboratory indicate that the sudden fuel pin rupture threshold is about 425 calories per gram. Analyses indicate that prompt dispersal of finely fragmented fuel into the coolant with subsequent large pressure rise rates does not occur at excursion energy densities below 425 cal/gm. Excursion energies above this level can cause pressure surges which may endanger the nuclear system process barrier.

In order to provide margin below the 425 cal/gm a limit on peak fuel enthalpy of 280 cal/gm is selected. At this point the uranium dioxide vapor pressure is insignificant. This fuel enthalpy limit is supported by a careful study of all available SPERT, TREAT, KIWI, and PULSTAR tests.[50] These tests indicate fairly rapid pressure rise rates above a fuel enthalpy of 400 cal/gm. These pressure rise rates increase with increasing fuel enthalpy. At 280 cal/gm, the pressure rise rates become less than 50 psi/sec. Pressure rise rates of this order of magnitude pose no threat to the nuclear system process barrier. The specified control rod withdrawal sequences to be used are designed to limit rod worth, so that the drop of any control rod from the core to the position of its drive results in a peak fuel enthalpy of not more than 280 cal/gm. A velocity limiter, which is an integral part of the control rod, limits the maximum rod velocity to 5 ft/sec. The velocity limiter is described in Section III-4, "Reactivity Control Mechanical Design."

Control rod removal excursion analysis from the shutdown flux level indicates that peak fuel enthalpies of 280 cal/gm (fully molten uranium dioxide) results from rod worths of 0.029 Δk (cold, critical) or 0.038 Δk (hot, critical) and removal rates of 5 ft/sec. These analyses also show that for excursions initiated from flux levels corresponding to 10% power, the maximum possible control rod worth, 0.038 Δk, is insufficient to cause peak enthalpies of 280 cal/gm. The design basis rod drop accident is evaluated in Section XIV-6, "Analysis of Design Basis Accidents."

Preplanned rod patterns enforced by the rod worth minimizer restrict rod worths to less than 0.01 Δk, although larger values are acceptable within the 280 cal/gm limit. Above 10% power, it is impossible to obtain a rod with worth high enough to produce peak enthalpy of 280 cal/gm if the rod were removed at 5 ft/sec. Planned rod patterns and rod worth minimizer, therefore, are not needed to restrict control rod patterns to limit the consequences of the rod drop accident when the reactor is above 10% power.

6.8 Testing and Verification

The shutdown reactivity requirement is verified anytime core loading changes or control rod replacements are made. Nuclear limitations for components other than the fuel are verified by testing the individual systems periodically. The test capabilities are described in other sections.

7.0 THERMAL AND HYDRAULIC DESIGN

7.1 Safety Objective

The safety objective of the thermal and hydraulic design of the core is to prevent fuel damage at rated power output throughout the life of the fuel.

7.2 Safety Design Bases

1. The thermal hydraulic design of the core shall establish limits for use in setting devices of the nuclear safety systems so that no fuel damage occurs as a result of abnormal operational transients (see Section XIV, "Station Safety Analysis").

2. The thermal hydraulic design of the core shall establish a thermal hydraulic safety limit for use in evaluating the safety margin relating the consequences of fuel barrier failure to public safety.

7.3 Power Generation Objective

The objective of the thermal and hydraulic design of the core is to achieve power operation of the fuel over the life of the core in a safe and efficient manner.[51]

7.4 Power Generation Design Bases

The thermal hydraulic design of the core shall provide the following characteristics:

a. The ability to achieve rated core power output throughout the design lifetime of the fuel without sustaining fuel damage.

b. The flexibility to adjust core power output over the range of plant load and load maneuvering requirements without sustaining fuel damage.

7.5 Thermal and Hydraulic Limits

7.5.1 Steady-State Limits

For purposes of maintaining adequate thermal margin during normal steady-state operation, the minimum critical power ratio (MCPR) shall not be lower than the limiting values in the Technical Specifications, the average planar linear heat generation rate (APLHGR) shall not be greater than the limiting values in the Technical Specifications and the maximum linear heat generation rate (LHGR) shall be maintained below the limits in the Technical Specifications for all bundles.[10] 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 noted previously. The core and fuel design bases for steady-state operation, i.e., MCPR, APLHGR, and LHGR limits have been defined to provide sufficient margin between the steady-state operating condition 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.

7.5.2 Transient Limits

The transient thermal limits are established such that fuel damage is not expected to occur during the most severe abnormal operating transients.

Fuel damage is defined for design purposes as perforation of the cladding which permits release of fission products (see Section III-2, "Fuel Mechanical Design"). The mechanisms which could cause fuel damage in reactor transients are:

a. Severe overheating of the fuel cladding caused by inadequate cooling. Fuel damage due to local overheating of the cladding is conservatively defined as the onset of the transition from nucleate to film boiling, although fuel damage is not expected to occur until well into the film boiling regime. If MCPR remains above limiting values, no fuel damage would be calculated to occur as a result of inadequate cooling.

b. Rupture of the fuel cladding due to strain caused by relative expansion of the uranium dioxide pellet and the fuel cladding. A value of 1% plastic strain of Zircaloy cladding is conservatively defined as the one unit below which fuel damage due to overstraining of the fuel cladding is not expected to occur. Available data indicates that the threshold for damage is in excess of this value. Determination of the limits required to exceed 1% plastic strain is discussed in Section III-2.5.

The mechanical overpower is used to evaluate the potential for overstraining of the cladding. The incremental cladding strain during a transient is proportional to the change in fuel volume average temperature, which is proportional to the change in either the fuel rod linear power or the fuel rod surface heat flux at a particular axial location or cross-section of the fuel rod.

Limits are determined such that the fuel cladding is not overstrained due to mechanical overpower.

c. Fuel rod failure can be caused by fuel centerline melting. Thermal overpower is used to evaluate the potential for the fuel entering the molten state at the fuel centerline. Temperature at the fuel centerline is proportional to either the fuel rod linear power or the fuel rod surface heat flux, so the magnitude of these quantities reached during the transients are the parameters of interest. Determination of the limits required to reach centerline melting is discussed in Section III-2.5.

Limits are determined to protect the fuel centerline temperature from thermal overpower.

The transient limits used in core design require that the peak linear heat generation rate remains below that which will cause fuel damage during the worst anticipated transient, that the resulting MCPR does not decrease below the safety limit,[54] and that the thermal-mechanical limits, as defined above, are met while taking into account special local effects such as control blade history. Demonstration that these limits are not exceeded is sufficient to conclude that no fuel damage occurs. It should be noted also that the steady-state operating limits have been established to assure that sufficient margin exists between the steady-state operating condition and any fuel damage condition to accommodate the worst anticipated transient without experiencing fuel damage at any time in life. APLHGR limitations are contained in the Technical Specifications. LHGR is monitored and its requirements are located in the Technical Specifications.

7.5.3 Fuel Burnup Restrictions

For the Fuel Handling Accident and the Loss of Coolant Accident, Table 3 of RG 1.183 is used for the fission product inventory gap fractions. Footnote 11 to this table states that the values are acceptable for fuels having a peak burnup up to 62,000 MWD/MTU provided that the maximum linear heat generation rate does not exceed 6.3 kw/ft peak rod average power for burnups exceeding 54 GWD/MTU. Core design control measures ensure that the limitations of this footnote are met.

7.6 Thermal and Hydraulic Characteristics

7.6.1 Application of Thermal-Hydraulic Limits to Core Design

The design basis employed for the thermal and hydraulic characteristics incorporated in the core design, in conjunction with the plant equipment characteristics, nuclear instrumentation, and the Reactor Protection System, is to require that no fuel damage occur during normal operation or during abnormal operational transients. Demonstration that the applicable thermal-hydraulic limits are not exceeded is given by analyses.

The plant is designed to operate at an average core power density of approximately 49.1 kW/liter.[55] Refueling is accomplished to achieve a planned fuel cycle. The replacement fuel is designed to achieve an average exposure that is consistent with the desired thermal generation.

7.6.2 Analytical Methods

The analytical methods, thermodynamic data, and hydrodynamic data used in determining the thermal and hydraulic characteristics of the plant are similar to those used throughout the nuclear power industry. Generally, the core power distribution and the required design power are established from design calculations and operating data. The number of fuel assemblies needed for the design power output is first established, and adequate coolant flow is then provided to ensure proper cooling of the fuel assemblies.

The lattice analyses are performed during the bundle design process. The results of these single bundle calculations are reduced to "libraries" of lattice reactivities, relative rod powers, and few group cross sections as functions of instantaneous void, exposure, exposure-void history, control state, and fuel and moderator temperature, for use in the core analysis. These analyses are dependent upon fuel lattice parameters only and are, therefore, valid for plants and cycles to which they are applied.

The core analysis is unique for each cycle. It is performed in the months preceding the cycle loading to demonstrate that the core meets all applicable safety limits. The principal tool used in the core analysis is the three-dimensional Boiling Water Reactor Simulator code, which computes power distributions, exposure, and reactor thermal-hydraulic characteristics, with spatially varying voids, control rods, burnable poisons and other variables.

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 boiling water reactors. The components of bundle pressure drop considered are friction, local, elevation, and acceleration. 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.

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 plate (experiencing a pressure loss) 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 bundles have lower tieplate holes. The majority of the flow continues through the lower tieplate (experiencing a pressure loss) where some flow exits through the flow path defined by the fuel channel and lower tieplate into the bypass region. This bypass flow is generally lower for those fuel assemblies with finger springs. GNF2 assemblies have different sized bypass flow holes in the lower tie plate to ensure that bypass flow in that assembly type is similar to its predecessor, GE14. The bypass flow paths considered in the analysis and typical values of the fraction of bypass flow through each flow path are given in NEDE-21156, February 1976.

Within the fuel assembly, heat balances on the active coolant are performed nodally. Fluid properties are expressed as the bundle average at the particular node of interest and are based on 1967 International Standard Steam-Water Properties. In evaluating fluid properties a constant pressure model is used.

The relative radial and axial power distributions documented in the country-specific supplement are used with the bundle 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.

7.6.3 Performance Range for Normal Operations

A boiling water reactor must operate within certain power and flow restrictions to meet design features, such as Pump Net Positive Suction Head (NPSH) requirements, overall plant control characteristics, and core thermal power limits. A typical power-flow map for the power range of operation is shown in Figure III-7-1a. The nuclear system equipment, nuclear instrumentation, and the Reactor Protection System, in conjunction with operating procedures, maintained operations within acceptable areas of this map for normal operating conditions prior to implementing the modified power/flow line restrictions discussed in Section 7.6.4, below. The boundaries on this map are as follows:

Natural Circulation Line

The operating state of the reactor moves along this line for the normal control rod withdrawal sequence in the absence of recirculation pump operation.

20% Pump Speed Line

Startup operations of the plant are normally carried out with the recirculation pumps operating at approximately 20% speed. The operating state for the reactor follows this line for the normal control rod withdrawal sequence.

Nominal Expected Flow Control Line

The nominal expected flow control line passes through 100% power at 100% flow. The operating state for the reactor follows this line (or one below it) for recirculation flow changes with a fixed control rod pattern. The line is based on constant xenon concentration.

Constant Pump Speed Line

This line shows the change in flow associated with power reduction from 100% power, 100% flow, while maintaining constant recirculation pump speed.

Maximum Expected Flow Control Line

This line represents the flow control line for plant startup in which the recirculation pump speed is increased above 20% speed before control rod withdrawal is continued.

7.6.4 Flow Control

The large negative operating coefficients, which are inherent in the boiling water reactor, provide important advantages as follows:

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.

c. Strong damping of spatial power disturbances.

The reactor power level can be controlled automatically by flow control over approximately a 25% power range (75%-100%). The power range for automatic control is adjusted by moving control rods.

Load following is accomplished by varying the recirculation flow to the reactor. This method of power level control takes advantage of the reactor negative void coefficient. To increase reactor power, it is necessary only to increase the recirculation flow rate which sweeps some of the voids from the moderator, causing an increase in core reactivity. As the reactor power increases, more steam is formed and the reactor stabilizes at a new power level with the transient excess reactivity balanced by the new void formation. No control rods are moved to accomplish this power level change. Conversely, when a power reduction is required, it is necessary only to reduce the recirculation flow rate. When this is done, more voids are formed in the moderator, and the reactor power output automatically decreases to a new power level commensurate with the new recirculation flow rate. No control rods are moved to accomplish the power reduction.

Load following through the use of variations in the recirculation flow rate (flow control) is advantageous relative to load following by control rod positioning. Flow variations perturb the reactor uniformly in the horizontal planes, and thus allow operation with flatter power distribution and reduced transient allowances. As the flow is varied, the power and void distributions remain approximately constant at the steady-state end points for a wide range of flow variations. These constant distributions provide the important advantage that the operator can adjust the power distribution at a reduced power and flow by movement of control rods and then bring the reactor to rated conditions by increasing flow, with the assurance that the power distribution will remain approximately constant. Section VII-9, "Recirculation Flow Control System," also describes the components which are used to vary the recirculation flow.

The following simple description of boiling water reactor operation with recirculation flow control summarizes the principle modes of normal power range operation and the originally imposed restrictions on ascension to full power. Assuming the plant to be initially hot with the reactor critical, full power operation could be approached following the sequence shown as points 1 to 7 in Figure III-7-1a. The first part of the sequence (1 to 3) was achieved with control rod withdrawal and manual, individual recirculation pump control. Individual pump startup procedures were provided which achieved 20% of full pump speed in each loop. The natural circulation characteristics of the boiling water reactor were still influential at this pump speed level as shown in the appropriate curve. Power, steam flow, and feedwater flow reached approximately 20%. An interlock on feedwater flow prevented low power-high recirculation flow combinations which could create recirculation pump NPSH problems.

Once the feedwater interlock was cleared, the operator could manually increase recirculation flow in each loop until the operating state reached point 3, the lower limit of the flow control range.

The recirculation system master controller was limited, and these limits establish the operating state (see Section VII-9, "Recirculation Flow Control System"). The original operating map is shown in Figure III-7-1a with the most restrictive flow control range expected.

Reactor power increased as the operating state moved from point 2 to 3 due to the inherent flow control characteristics of the boiling water reactor. At point 3 the operator could switch to simultaneous recirculation pump control. Thermal output could then be increased by either control rod withdrawal or recirculation flow increase. For example, the operator could reach approximately 50% power in the ways indicated by points 4 and 5. With an increase of recirculation pump speed to rated flow, point 4 could be achieved. If, however, it was desired to maintain lowest recirculation flow, approximately 50% power could be reached by withdrawing control rods until point 5 was reached.

The curves labeled "Minimum Expected Flow Control Line" and "Design Flow Control Line" represent typical steady-state power-flow characteristics for fixed rod patterns. They are slightly affected by xenon, differences in core leakage flow assumptions, and reactor vessel pressure variations. However, for this example, these effects have been neglected.

Normal power range operation was along or below the "Nominal Expected Flow Control Line." If load following response was desired in either direction, plant operation near 90% power provided most capability. If maximum load pickup capability was desired, the nuclear system could be operated near point 6, with fast load response available all the way up to point 7, rated power.

7.6.4.1 Load Line Limit and Extended Load Line Limit Analyses

To provide relief from the operating restrictions inherently imposed during ascension to power by the original power/flow curve of Figure III-7-1a, a modified power/flow curve was used.[56] To further improve capacity factor, operational flexibility, and to reduce operational cost, the power-flow map has been expanded to larger operating boundaries as shown in Figure III-7-1b. In deriving this power-flow map, five design basis objectives were specified:

(1) For those transients and accidents that are analyzed along the power-flow limit line of Figure III-7-1b and are sensitive to variations in power and flow, the license basis point (100% Power, 100% Flow) and/or other operating points shall be shown to be the most limiting.

(2) In no instance shall the permitted power-flow points on the operating map exceed those defined by the modified APRM rod block line.

(3) The slope of the rod block intercept line must be such that recirculation flow increases are capable of compensating for xenon buildup while increasing reactor power.

(4) The consequences of all accidents and transients analyzed in the FSAR and its amendments, the USAR and license submittals must remain within the limits normally specified for such events.

(5) Reactor power ascension from minimum pump speed and zero power to full power shall be directly attainable through combined control rod movement and recirculation flow increases without violation of the modified power-flow limit line.

To meet these objectives, plant unique analyses[79] were performed and conclusions were drawn concerning the safety consequences of operation in the extended operating region (Figure III-7-1b).

7.6.4.2 Maximum Extended Load Line Limit and Increased Core Flow Analyses

To further improve capacity factor, operational flexibility, and to reduce operational cost, the power-flow map has been expanded by Maximum Extended Load Line Limit (MELLL) and Increased Core Flow (ICF) Analyses. The expanded power-flow map is shown in Figure III-7-1b. The MELLL and ICF Analyses are documented in Reference 114 and 118.

The three items specified for plant/cycle-specific investigation which have been verified by the Supplemental Reload Licensing Submittal for CNS cycle 20 are:

(1) Transients - All corewide anticipated operational occurrences were examined and limiting transients were identified and analyzed.[79 & 82] These evaluations and analyses were performed using NRC approved standard reload licensing methodology.[1]

(2) Loss of Coolant Accident (LOCA) - The LOCA analyses covering the expanded power-flow map has been verified to assure that 10CFR50.46 limits will be met.

(3) Stability - The operating procedure of the Cooper Nuclear Station is consistent with the Option 1-D stability solution implemented by CNS documented in Reference 80 and subsequently updated for CNS as reported in Reference 114.

7.6.5 Core Power Distribution

7.6.5.1 Design Power Distribution

Thermal design of the reactor - including the selection of the core size and effective heat transfer area, the design steam quality, the total recirculation flow, the inlet subcooling, and the specification of internal flow distribution - is based on the concept and application of a design power distribution. The reactor design power distribution conservatively represents the worst combination of individual peaking factor magnitudes and shape that can occur in a CNS core.

The design power distribution is based on detailed calculations of the neutron flux distribution. The analyses have been correlated and verified by operating data and experience.

The core power distribution is a function of fuel bundle design, core loading, control rod pattern, core exposure distributions and core coolant flow rate. The thermal performance parameters, MAPLHGR and MCPR, limit unacceptable core power distributions. (GESTARII[113])

7.6.5.2 Local Power Peaking

Core loading consists of fuel assembly types which differ in the following essential respects:

Enrichment

The major difference in the bundle types is that of average enrichment. The fuel enrichments are adjusted to meet the power generation needs of the plant, while keeping thermal limits within approved limits. NEDO-31152P describes the assembly enrichment distribution for all current fuel.

Control Augmentation

Control augmentation is necessary to supplement the control rod system. This additional control is provided in the form of Gd_2O_3 in solid solution with UO_2 in several selected fuel rod positions of the high enrichment bundles. These locations are selected to provide low local as well as axial peaking and low thermal duty on the gadolinia containing fuel rods. The high enrichment bundle types differ in gadolinia concentration and distributions designed to control power distributions in the axial and radial directions.

7.6.5.3 Gross Peaking Factor

Both calculations and observations of operating reactors indicate that the plant operates through the fuel cycle without approaching the design value of gross peaking, 1.40 x 1.40 = 1.96. Results of the analytical evaluations of a typical rod withdrawal program are shown in Figure III-7-3, where gross peaking remains well below the design value of 1.96 (transverse X axial) throughout fuel cycle life. Only at the end of the fuel cycle, when essentially no control rods are left in the core for power shaping, does the power peaking approach the design value.

7.6.5.4 Core Coolant Flow Distribution and Orificing Pattern

Correct distribution of core coolant flow among the fuel assemblies is accomplished by the use of an accurately calibrated fixed orifice at the inlet of each fuel assembly. The orifice is located in the fuel support piece. They serve to control the flow distribution and, hence, the coolant conditions within prescribed bounds throughout the design range of core operation.

The core is divided into two orificed flow zones. The outer zone is a narrow, reduced power region around the periphery of the core; the inner zone consists of the core center region. No other control of flow and steam distribution, other than that incidentally supplied by adjustment of the power distribution with the control rods, is employed or needed. The orifices can be removed for changes during refueling operations if necessary.

The design flow distribution results in an approximately equal critical heat flux ratio in each flow zone at design conditions. The sizing and design of the orifices ensure that the flow in each fuel assembly is stable during all phases of operation at normal operating conditions. The predicted core performance under several conditions is included in Table III-7-1 for the design power and flow distribution.

7.6.6 Incore Vibration[58]

This USAR section contains historical information as indicated by the italicized text. USAR Section I-3.4 provides a more detailed discussion of historical information. The information being presented in this section as historical has been preserved as it was originally submitted to the NRC in the CNS FSAR.

The standard BWR confirmatory test, similar to the tests on Dresden 3, Quad Cities I and II, Pilgrim, Vermont Yankee, Fitzpatrick, and Peach Bottom II, were performed on CNS. This test is described in the FSAR or amendments thereto for these plants (e.g., Quad Cities Amendment 23, Section 3, Vermont Yankee Amendment 26, etc.).

The abnormal operating conditions that are considered during the vibration test program are the transient recirculation pump trip conditions. While operating at power and at 100% core flow, the recirculation pumps are tripped, both individually and simultaneously (three separate trip conditions), and permitted to coast down to minimum speed. Power levels of 50%, 75%, and 100% power are tested in this manner. During each of these pump trip transients, the vibration motions are monitored and recorded to assure that the vibration is well within acceptable levels.

The following description is representative of the different flow modes of operation and transients to which the internals were subjected during vibration testing:

50% Thermal Power Line (Approximately 20% thermal power to 50% thermal power):

1. *Approximately equally spaced flow points from minimum flow to 100% flow.*

2. *With 100% core flow trip pump A.*

3. *With 100% core flow trip pump B.*

4. *With 100% core flow trip both pumps simultaneously.*

TABLE III-7-1

END OF LIFE THERMAL AND HYDRAULIC CONDITIONS*

General Operating Conditions	CASE I	CASE II
Reference Design Thermal Output, MWt	2,381	2,381
Steam Flow Rate, lb/hr	9,810,000	9,810,000
Maximum Core Coolant Flow Rate, lb/hr	73.5×10^6	73.5×10^6
Feedwater Flow Rate, lb/hr	9,810,000	9,810,000
System Pressure, Nominal in Steam Dome, psia	1,020	1,020
Core Pressure for Thermal-Hydraulic Analysis, psia	1,035	1,035
Average Power Density, kW/liter	50.6	50.6
Maximum Thermal Output, kW/ft	17.2	18.5
Average Thermal Output, kW/ft	7.09	7.09
Maximum Heat Flux, Btu/hr-ft^2	399,100	428,300
Average Heat Flux, Btu/hr-ft^2	162,530	162,530
Maximum UO_2 Temperature, °F	4,430	4,430
Average Volumetric Fuel Temperature, °F	1,210	1,210
Average Fuel Rod Surface Temperature, °F	560	560
MCHFR at Reference Design Thermal Output**	1.96	1.90
Coolant Enthalpy at Core Inlet, Btu/lb	520.1	520.1
Core Maximum Exit Voids Within Assemblies, %	75	75

Design Power Peaking Factor	Value	Value
Transverse Peaking Factor	1.50	1.50
Local Peaking Factor	1.17	1.16
Axial Peaking Factor	1.39	1.50
Gross Peaking Factor	2.09	2.24
Total Peaking Factor	2.43	2.61

*(7x7 Fuel Bundles) and updated input information is found in Chapter XIV, Table XIV-5-2.
**MCHFR replaced by MCPR as applicable parameter.

75% Thermal Power Line:

1. *Approximately equally spaced flow points from minimum flow to 100% flow.*

2. *With 100% core flow trip pump A.*

3. *With 100% core flow trip pump B.*

4. *With 100% core flow trip both pumps simultaneously.*

100% Thermal Power Line:

1. *Approximately equally spaced flow points from minimum flow to 100% flow.*

2. *With 100% core flow trip pump A.*

3. *With 100% core flow trip pump B.*

4. *With 100% core flow trip both pumps simultaneously.*

A comparison of the preliminary Fitzpatrick vibration measurement results and the Cooper Reactor Internal vibration measurement results was completed. Fitzpatrick is the designated prototype 218 size BWR/4, and Cooper is a similar plant for which vibration measurements of lesser scope were conducted.[59][115]

The comparable instrumented components were as follows:

	Sensors	
	Cooper	Fitzpatrick
Upper Bolt Guide Ring (Vibration of shroud and shroud head assembly)	A-1, A-2	A-1, A-3
Jet Pump Elbow to Vessel (radial)	D-1, D-2	D-6, 7, 9, 12
Jet Pump Riser Brace	S-1, S-5	S-1 thru S-20

The sensor readings on the upper bolt guide ring and jet pump elbow to vessel can be compared directly, but direct comparison of the riser brace readings is complicated because the Cooper sensors were in half bridge (single sensor) configuration and the Fitzpatrick sensors were in full bridge (two sensors) configuration. Thus in the case of the riser brace data, comparison between amplitudes expressed as percentages of the steady state criteria is used.

For this comparison a list of test results for the 100% rated core flow on each power line is provided.

A direct comparison of the amplitudes and frequencies for the jet pump elbow to vessel sensors and the upper bolt guide ring show that for significant measurements (greater than one mil) Cooper and Fitzpatrick have similar vibration characteristics. Comparing riser brace strain gage amplitudes, stated as percentages of the criteria, the Cooper data lies within the range of values for the Fitzpatrick data. Maximum amplitudes at Cooper are generally lower than at Fitzpatrick because a smaller number of riser braces were instrumented at Cooper.

These results also show that vibration levels of major core structures and internals in both of these plants are within acceptable limits during normal plant operation. No significant unexplained or anomalous vibration behavior was observed.

JP Elbow to Vessel D-1, D-2 Cooper
D-6, 7, 9, 12 Fitzpatrick

	Cooper		Fitzpatrick	
	Amplitude (mils)	Frequency (Hz)	Amplitude (mils)	Frequency (Hz)
50% P.L.	4.0	24.1	3.0	23.8
	1.0	34		
75% P.L.	3.2	24.5	6.0	23
	1.4	35	1.25	31.5
	.4	41	2.5	33
			1.0	38
100% P.L.	4.0	24.7	6.*	23.8
	1.2	26		

**6 mils is 7.7% of the allowable amplitude at this frequency.*

Upper Bolt Guide Ring (A-1, A-2 Cooper)
(A-1, A-3 Fitzpatrick)

	Cooper		Fitzpatrick	
	Amplitude (mils)	Frequency (Hz)	Amplitude (mils)	Frequency (Hz)
50% P.L.	3	5	3.0	4.7
	4	9.5	4.5	5.5
			10	9.0
75% P.L.	6.7	4.0	9.5	5.0
	5.5	9.0	4.5	10.25
	2.3	18	3.0	12
	1.5	22	.75	31
			.5	34.5
100% P.L.	10	5	11.*	4.8
	5	9	5	9
	5	10	3.5	13.4
	1.5	23.5	.5	22.5
	1	33.5	1.1	33.5
	1	37.5		

**11 mils is 14% of the allowable amplitude at this frequency.*

Jet Pump Riser Brace (Cooper S-2, S-5)
(Fitzpatrick S-1 thru S-20)

	Cooper					
	Fitzpatrick					
	Amplitude □□	*Frequency* Hz	*%* Criteria	*Amplitude* □□	*Frequency* Hz	*%* Criteria
50% P.L.	*20*	*25*	*9.3*	*40*	*26*	*20.7*
	15	*26*	*7.0*	*25*	*29.5*	*8*
	24	*34*	*11.8*	*30*	*31*	*45*
	10	*36*	*7.2*	*27.5*	*35*	*8*
				37.5	*36*	*10.6*
				37.5	*39*	*4*
75% P.L.	*15*	*22*	*10*	*20*	*24*	*10.4*
	18	*23*	*8.4*	*20*	*29*	*7.6*
	9	*33*	*2.1*	*45*	*33.5*	*10.6*
	17	*37*	*6.8*	*45*	*35*	*12.7*
	5.8	*116*	*5.5*	*32*	*37*	*9.0*
				10	*108*	*3.0*
				5	*121*	*1.5*
				3	*242*	*1.2*
100% P.L.	*20*	*25*	*20.6*	*25*	*32.5*	*38*
	10	*31*	*4.3*	*35*	*35*	*9.8*
	20	*37*	*9.8*	*30*	*37.5*	*8.4*

7.6.6.1 Evaluation of Incore Vibration for Increased Core Flow

The flow-induced vibration responses of the reactor internals at Cooper were originally licensed based on the results of the Regulatory Guide 1.20 prototype test performed at the Fitzpatrick Nuclear Power Station. The major components instrumented at Fitzpatrick include the shroud, head, shroud-separator assembly, jet pumps, and fuel channels. The results of these tests were reviewed to determine the reactor internals most likely to have significant vibration in the ICF and MELLL domains, and vibration stresses for these components were extrapolated and compared with the acceptance criteria. An evaluation was also performed to determine if the natural frequencies of these components were at or near the new recirculation pump vane passing frequencies when operating in the ICF domain.

In similar analyses, the control rod guide tubes and in-core guide tubes and the steam dryer were evaluated using test results from other BWR plants. In addition to these analyses, the jet pump sensing lines (JPSLs) were evaluated using a detailed modal analysis coupled with test results from the GE High Flow Hydraulic Facility.

All safety-related reactor internal components, except for two JPSLs, that were evaluated had stresses less than the acceptance criteria at the increased core flow rate condition. Two JPSLs, one per recirculation loop, were found to have a remote possibility for a second natural frequency that could be near the recirculation pump vane passing frequency if five as-built lengths were at their extreme design value tolerances. Therefore, it is highly improbable that these two sensing lines actually have natural frequencies near the maximum pump speed. In any case, a single JPSL is sufficient to detect any jet pump anomaly in a jet pump pair. Thus, failure of one JPSL is not a safety issue.

7.7 Safety Evaluation

7.7.1 MCPR Limit[10]

Three different types of boiling heat transfer from the fuel rods to the coolant can occur in a forced convection system; nucleate, transition, and film boiling. Nucleate boiling, which occurs at lower heat transfer rates, is an extremely efficient mode of heat transfer, allowing large quantities of heat to be transferred with a very small temperature rise at the cladding surface. As heat flux is increased, the boiling heat transfer surface alternates between nucleate and film boiling (termed transition boiling), leading to fluctuations in the cladding surface temperature. With a continued increase in heat flux, transition boiling ends and film boiling begins. Film boiling heat transfer is characterized by stable, but much higher, cladding surface temperatures than those experienced during nucleate boiling.

The fuel rod cladding employed for the nuclear fuel is Zircaloy-2. This material is selected primarily for its nuclear properties. Zircaloy also has good corrosion and strength properties at normal operation conditions (nucleate boiling region). Extended operation at the elevated cladding temperatures possible in the transition and film boiling regimes, could cause gradual reduction in the fuel rod cladding strength and could accelerate its corrosion, possibly resulting in damage to the cladding. The objective for normal operation and abnormal operational transients is to maintain nucleate boiling and thus avoid a transition to film boiling.

The conditions which produce a boiling transition are determined experimentally. The original analysis for Cooper Nuclear Station presented in the SAR characterized the conditions producing transition boiling in terms of critical heat flux as a function of critical quality. The required margin between steady-state conditions and those which would produce a boiling transition was termed the minimum critical heat flux ratio (MCHFR). If MCHFR was maintained above 1.0 during normal operation and abnormal operational transients, no fuel damage would be calculated to occur as a result of inadequate cooling.

Continued experimentation resulted in development of a new correlation for the conditions producing the onset of transition boiling. This was the General Electric Critical Quality (X_c) - Boiling Length Correlation (GEXL). The GEXL quality-boiling length correlation was developed in the 1970's. As new fuel designs were developed and tested, the information in the database for boiling transition in 8x8 fuel assemblies was expanded. A comparison of the GEXL correlation to the expanded database demonstrated that a refinement was necessary to improve the accuracy of the boiling transition prediction for current fuel designs. The result of this refinement is a modified form of GEXL which is referred to as GEXL-Plus. (A description of the GEXL-Plus correlation and its application to the derivation of safety limit MCPR and operating limit MCPR are documented in References 83 and 84. The NRC approval of the correlation and application are documented in References 85 and 86.)

In both the GEXL and GEXL-Plus correlations, the margin between steady-state conditions and those which would produce a boiling transition is expressed as a critical power ratio. This is defined as the ratio of the critical power (fuel bundle power at which some point within the bundle experiences onset of transition boiling) to the operating bundle power.

Thermal margin is stated in terms of the minimum value of the critical power ratio, MCPR. To ensure that adequate margin is maintained, a design requirement based on a statistical analysis was selected as follows:

> Moderate frequency transients caused by a single operator error or equipment malfunction shall be limited such that, considering uncertainties in manufacturing and monitoring the core operating state, at least 99.9% of the fuel rods would be expected to avoid boiling transition.

The lowest allowable transient MCPR limit which meets the above design requirement is termed the fuel cladding integrity safety limit MCPR. Statistical analyses have been performed which provide fuel cladding integrity safety limit MCPRs applicable to all GE fuel designs in BWR reload and initial core cycles. The statistical analysis procedure, including codes, correlations and analytical procedures is referred to as the General Electric Thermal Analysis Basis (GETAB) uncertainties with revised methodology (Reference 121).

A plant-unique operating limit MCPR is established to provide adequate assurance that the fuel cladding integrity safety limit is not exceeded for any anticipated operational transients. The operating limit MCPR is obtained by addition of the maximum ΔCPR value for the most limiting transient postulated to occur at the plant to the fuel cladding integrity safety limit. In general, the analysis basis for most of the transient analyses is the full power, full flow, EOC, all-rods out condition. Cycle specific ΔCPR values are determined as part of the reload analysis and are reported in the Supplemental Reload Licensing Report (Reference 88).

CNS is licensed for the Average Power Range Monitor, Rod Block Monitor and Technical Specification (ARTS) Improvement Program and has both power- and flow-dependent limits imposed on the operating limit MCPR to ensure that the safety limit MCPR is not exceeded for all power/flow conditions. The flow-dependent limit provides the thermal margin required to protect the fuel from transients resulting from inadvertent core flow increases. The power-dependent limit protects the fuel from the other limiting abnormal operating transients, including localized events such as a rod withdrawal error.

In addition to core thermal power, core flow, fuel type, and fuel exposure; the operating limit MCPR value depends on the scram time. As identified in Reference 88, core-wide rapid pressurization events (e.g., load reject without bypass, turbine trip without bypass, feedwater controller failure) are analyzed using the GEMINI method of the system model ODYN. The ΔCPR which results must be adjusted such that there is a 95% probability with 95% confidence that the safety limit will not be violated. Each plant has the choice of operating under either Option A or Option B to make this adjustment.

With the GEMINI set of methods, the MCPR for each event is determined using statistically evaluated scram times. Plants that do not demonstrate compliance with the statistically evaluated scram times must operate using a higher limit that does not take credit for these scram times. The higher limit is referred to as Option A. Plants using Option B must demonstrate that their scram speed distribution is consistent with that used in the statistical analysis.

The Supplemental Reload Licensing Report will contain operating limit MCPR results for both the case where scram speed compliance is demonstrated (Option B) and the case where it is not (Option A). The actual operating limit will be a straight-line interpolation between these two values dependent on the results of scram speed testing. Reference 87 provides the curves and instructions for determining the required Operating Limit MCPR.

7.7.2 Fuel Damage Analysis

Fuel damage is defined as perforation of the fuel cladding, refer to Section III-2.6 for fuel damage analysis. Defects in the fuel cladding should be minimized for two reasons:

a. Defects permit the release of fission products to the reactor coolant. This release involves a portion of those fission products that have diffused out of uranium dioxide matrix.

b. Water which enters the fuel rods through defects can cause progressive clad corrosion and further deterioration of the cladding in the fuel rod leading eventually to water and steam leaching of fission products and uranium from the fuel pellets. If this progressive failure persists, the reactor coolant activity level increases, and it becomes necessary to remove the fuel assembly from the core.

Predictions of the amount of fuel damage associated with a specific operation involves complex functions interrelating design methods and material properties, manufacturing methods and assembly tolerances, material specifications and quality control, operating variables and the effectiveness of reactor protection equipment, the initial starting conditions, and the amount of change in these conditions that constitute fuel damage. The interrelationships between these variables are continually evaluated by design, production, operating, and safety engineers. The only practical method of interrelating these variables is through use of probability functions and uncertainty analyses. Quality control procedures, including 100% ultrasonic inspection of Zircaloy tubing and 100% helium leak check of fuel rods assure an extremely low probability that fuel rods have leaks prior to operation.

7.7.3 Fuel Damage Experience

Although the incidence of failure in General Electric Zircaloy-clad UO_2 fuel has been quite low (data available prior to the filing of the original FSAR indicated 0.5% out of more than 218,000 Zircaloy-clad UO_2 fuel segments had experienced failure), fuel has been operated at Dresden Unit 1 and elsewhere with perforated cladding. Dresden Unit 1 has been operated with some failures in Type I Zircaloy-clad UO_2 fuel and in the Type II stainless steel-clad UO_2 fuel. The Humboldt Bay and Big Rock Point reactors have also been operated with failures in stainless steel fuel (Humboldt Type I and Big Rock Type A). The Big Rock reactor has operated with some fuel failures in both the Type B and Type E Zircaloy-clad UO_2 fuel designs as well as a number of failed high power (22 to 27 kW/ft) fuel rods in the center-melt development fuel assemblies. KRB has also operated with some perforated fuel rods during its initial operation. Failures have occurred due to manufacturing defects, incompatibility of stainless steel as cladding material in the BWR core steam/water environment, inadequate volume for accommodation of fuel expansion and/or fission gas pressure for fuel operated beyond design exposures, cladding overtemperature caused by excessive deposits of crud on fuel rod surfaces resulting from materials in the feedwater system and fretting wear caused by foreign debris trapped in fuel rod spacers. In essentially all cases, the mechanisms causing the fuel rod failures to develop in service have been carefully identified and appropriate corrections have been made to the manufacturing process, the fuel design or the system design and operation to preclude the recurrence of such failures in the current generation of reactors.

Operation with failed fuel rods has shown that the fission product release rate from defective fuel rods can be controlled by regulation of power level. The rate of increase in released activity apparently associated with progressive deterioration of failed rods has been deduced from chronological plots of the off-gas activity measurements in operating plants. These data indicate that the activity release level can be lowered by lowering the local power density in the vicinity of the fuel rod failure. These measured data also indicate that sudden or catastrophic failure of the fuel assembly does not occur with continued operation, and that the presence of a failed rod in a fuel assembly does not result in propagation of failure to neighboring rods. Shutdown for removal of fuel assemblies with large defects can be scheduled as required.

Evaluation of the fission product release rate for failed fuel rods shows a wide variation in the activity release levels. Attempts to relate the release rates to defect type, size, and specific power level have been made. These data support the qualitative observations that fission product release rates are functions of power density and that progressive deterioration is a function of time. However, insufficient failure data are available to quantify the detailed correlation between these variables.

A more detailed summary of General Electric experience with BWR Zircaloy-clad UO_2 pellet fuel, including production and development data obtained since the original application for a construction permit, is contained in Section III-2, "Fuel Mechanical Design."

7.8 Testing and Verification

The detailed core power and bundle power distribution are calculated periodically. The plant is operated as necessary to maintain MCPR, APLHGR, and the linear heat generation rates within the design values.[10]

8.0 CONTROL ROD DRIVE HOUSING SUPPORTS

8.1 Safety Objective

The safety objective of the control rod drive (CRD) housing supports is to prevent any significant nuclear transient in the event a drive housing breaks or separates from the bottom of the reactor vessel.

8.2 Safety Design Bases

The CRD housing supports shall meet the following safety design bases:

a. Following a postulated CRD housing failure, control rod downward motion shall be limited so that any resulting nuclear transient could not be sufficient to cause fuel damage.

b. The clearance between the CRD housings and the supports shall be sufficient to prevent vertical contact stresses caused by thermal expansion during plant operation.

8.3 Description

The CRD housing supports are shown in Figure III-8-1. Horizontal beams are installed immediately below the bottom head of the reactor vessel, between the rows of CRD housings. The beams are bolted to brackets welded to the steel form liner of the drive room in the reactor support pedestal.

Hanger rods, approximately 10 feet long and 1-3/4 inches in diameter, are supported from the beams on stacks of disc springs. These springs compress approximately two inches under the design load.

The support bars are bolted between the bottom ends of the hanger rods. The spring pivots at the top, and the beveled, loose-fitting ends on the support bars prevent substantial bending moment in the hanger rods if the support bars are overloaded.

Individual grids rest on the support bars between adjacent beams. Because a single-piece grid would be difficult to handle in the limited work space and because it is necessary that control rod drives, position indicators, and in-core instrumentation components be accessible for inspection and maintenance, each grid is designed for in-place assembly or disassembly. Each grid assembly is made from two grid plates, a clamp, and a bolt. The top part of the clamp guides the grid to its correct position directly below the respective CRD housing that it would support in the postulated accident.

When the support bars and grids are installed, a gap of approximately 1.25 inches at room temperature (approximately 70°F) is provided between the grid and the bottom contact surface of the control rod drive flange. During system heatup, this gap is reduced by a net downward expansion of the housings with respect to the supports. In the hot operating condition, the gap is approximately .25 inch.

In the postulated CRD housing failure, the CRD housing supports are loaded when the lower contact surface of the CRD flange contacts the grid. The resulting load is then carried by two grid plates, two support bars, four hanger rods, their disc springs, and two adjacent beams.

The American Institute of Steel Construction (AISC) Manual of Steel Construction, "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," was used in designing the CRD housing support system. To provide a structure that absorbs as much energy as practical without yielding, the assumed allowable tension and bending stresses are 90 percent of yield and the assumed shear stress is 60 percent of yield. These assumed bases are 1.5 times the AISC allowable stresses (60 percent and 40 percent of yield, respectively).

For purposes of mechanical design, the postulated failure resulting in the highest forces is an instantaneous circumferential separation of the CRD housing from the reactor vessel while the reactor is at operating conditions of pressure and temperature, with an internal pressure of 1250 psig (reactor vessel design pressure) acting on the area of the separated housing. The weight of the separated housing, control rod drive, and blade, plus the pressure of 1250 psig acting on the area of the separated housing, gives a force of approximately 35,000 lbs. This force is multiplied by a factor of three for impact, conservatively assuming that the housing travels through a one-inch gap before it contacts the supports. The total force (10^5 lb) is then treated as a static load in design. The CRD housing supports are designed as Class I (seismic) equipment in accordance with Appendix C of the USAR.

All CRD housing support subassemblies are fabricated of ASTM-A-36 structural steel, except for the following items:

	Material
Grid	ASTM-A-441
Disc springs	Schnorr, Type BS-125-71-8
Hex bolts and nuts	ASTM-A-307

8.4 Safety Evaluation

Downward travel of the CRD housing and its control rod following the postulated housing failure equals the sum of these distances: (1) the compression of the disc springs under dynamic loading and (2) the initial gap between the grid and the bottom contact surface of the CRD flange. With the reactor hot and pressurized, the downward motion of the control rod would be limited to the spring compression (approximately two inches) plus a gap of approximately 0.25 inches. The maximum gap allowed by procedure in the cold condition is 1.25" but with the reactor hot and pressurized the maximum gap decreases, to approximately 0.25", due to thermal and pressure effects. Thus, the control rod movement following a housing failure is substantially limited below one drive "notch" movement (six inches). The abnormal operational transient from sudden withdrawal of any control rod through a distance of one drive notch at any position in the core does not produce a transient sufficient to damage any radioactive material barrier.

The CRD housing supports are in place during power operation and when the nuclear system is pressurized. If a control rod is ejected during shutdown, the reactor remains subcritical because it is designed to remain subcritical with any one control rod fully withdrawn at any time.

8.5 Inspection and Testing

CRD housing supports are removed for inspection and maintenance of the control rod drives. The supports can be removed during reactor shutdown, even when the reactor is pressurized, because all control rods are then inserted. When the support structure is reinstalled, it is inspected for correct assembly, with particular attention to maintaining the correct gap between the CRD flange lower contact surface and the grid.

9.0 STANDBY LIQUID CONTROL SYSTEM

9.1 Safety Objective

a. The safety objective of the standby liquid control (SLC) system is to provide a backup method, independent of the control rods, to maintain the reactor subcritical as the nuclear system cools. Maintaining subcriticality thus assures that the fuel barrier is not threatened by overheating in the special event that not enough of the control rods can be inserted to counteract the positive reactivity effects of a colder moderator. The SLC system also meets the requirements of the final NRC rule on Anticipated Transients Without Scram (ATWS)[70].

b. The SLC system has a safety objective for accident mitigation based on the implementation of the Alternate Source Term (AST) methodology. SLC is used to control the pH of the water in the Suppression Pool, Reactor Vessel, and Core Cooling systems following a Design Basis LOCA.

9.2 Safety Design Bases

The standby liquid control system shall meet the following safety design bases:

a. Backup capability for reactivity control shall be provided, independent of normal reactivity control provisions in the nuclear reactor, to be able to shut down the reactor if the normal control ever becomes inoperative.

b. The backup system shall have the capacity for controlling the reactivity difference between the steady state rated operating condition of the reactor with voids and the cold shutdown condition, including shutdown margin, to assure complete shutdown from the most reactive condition at any time in core life.

c. The time required for actuation and effectiveness of the backup control shall be consistent with the nuclear reactivity rate of change predicted between rated operating and cold shutdown conditions. A fast scram of the reactor or operational control of fast reactivity transients is not specified to be accomplished by this system.

d. Means shall be provided by which the functional performance capability of the backup control system components can be verified periodically under conditions approaching actual use requirements. A substitute solution, rather than the actual neutron absorber solution, can be injected into the reactor to test the operation of components of the redundant control system.

e. The neutron absorber shall be dispersed within the reactor core in sufficient quantity to provide a reasonable margin for leakage or imperfect mixing.

f. The system shall be reliable to a degree consistent with its role as a backup to a safety system; the possibility of unintentional or accidental shutdown of the reactor by this system shall be minimized.

g. The system shall be designed for 86 gpm/13 boron percent weight equivalency[70].

h. The injection of the Sodium Pentaborate Decahydrate ($Na_2B_{10}0_{16}*10H_2O$) from the SLC system will raise the Suppression Pool pH enough to keep the pH above 7 after a Design Basis LOCA.

i. The SLC system is required to complete injection of Sodium Pentaborate Decahydrate within the first 8 hours following a Design Basis LOCA.

9.3 Description

The standby liquid control system (see Burns and Roe Drawing 2045, Sheet 2 and General Electric Drawing 920D225BB, Sheet 1) is manually initiated from the main control room to pump a boron neutron absorber solution into the reactor if the reactor cannot be shut down or kept shut down with the control rods. However, insertion of control rods is expected to assure prompt shutdown of the reactor should it be required.

The SLC system is required to shut down the reactor at a steady rate and keep the reactor from going critical again as it cools.

The SLC system is needed only in the special event that not enough control rods can be inserted in the reactor core to accomplish shutdown and cooldown in the normal manner.

The boron solution tank, the test water tank, the two positive-displacement pumps, the two explosive valves, and associated local valves and controls are mounted in the reactor building outside the primary containment. The liquid is piped into the reactor vessel and discharged near the bottom of the core shroud so it mixes with the cooling water rising through the core (see USAR Sections IV-2, "Reactor Vessel and Appurtenances Mechanical Design," and III-3, "Reactor Vessel Internals Mechanical Design").

The boron absorbs thermal neutrons and thereby terminates the nuclear fission chain reaction in the uranium fuel.

The specified neutron absorber solution is sodium pentaborate decahydrate ($Na_2B_{10}0_{16}.\ 10H_2O$). It is prepared by dissolving stoichiometric (enriched) quantities of borax and boric acid in demineralized water. This is accomplished by placing sodium pentaborate in the SLC storage tank and filling with demineralized water to at least the low level alarm volume. The solution is at design concentration at the low level alarm point and can be diluted with water up to the high level alarm to allow for evaporation losses, or to the overflow level volume to lower the saturation temperature. An air sparger is provided in the tank for mixing. To prevent system plugging, the tank outlets are located on the side of the tank above the bottom.

In MODES 1 and 2, the SLC system shall be able to deliver a sodium pentaborate solution or equivalent into the reactor as specified in the Technical Specifications.

The saturation temperature of the specified solution is 63°F at the low level alarm volume and approximately 51°F at the tank overflow volume (see Technical Specification Figures 3.1.7-1 and 3.1.7-2). A heater system maintains the solution temperature to prevent precipitation of the sodium pentaborate from the solution during storage and to prevent cavitation at temperatures above 140°F. High or low temperature causes an alarm in the control room. The high and low temperature alarms are set at 110° and 85°F, respectively. Tank level is also monitored in the control room. The high and low level alarms are set to indicate a change in solution volume, which might indicate a solution concentration change. Tank level is controlled within the volume and concentration limits specified by the Tech. Specs. Solution Volume vs. Concentration requirements. The tank overflow volume is 4565 gallons.

The equipment containing the solution is installed in a room in which the air temperature is normally maintained within the range of 50° to 100°F. In the event the heater system fails and the room temperature is below the minimum tank temperature of 85°F, the solution concentration in the tank can be reduced by adding water to the tank until an acceptable concentration level is reached (see Technical Specification Figures 3.1.7-1 and 3.1.7-2). By adding water to the storage tank, the solution concentration in the suction line can be reduced to the allowable 11.5% (see Technical Specification Figure 3.1.7-1). This corresponds to an adjusted saturation temperature* of 62°F (see Technical Specification Figure 3.1.7-2).).

At the minimum pump flow rate of 38.2 gpm (as specified in Technical Specification, SR 3.1.7.7), each positive displacement pump is capable of injecting the tank contents into the reactor within 82 to 116 minutes, independent of the solution concentration in the tank (within the required volume range of 3132 gal@ 16% and 4414 gal@ 11.5%, Technical Specification Figure 3.1.7-1). This range of injection time ensures that the boron gets into the reactor core quicker than the cooldown rate and that there is sufficient mixing so the boron does not recirculate through the core in uneven concentrations that could possibly cause nuclear power to rise and fall cyclically.

In the unlikely event the suction piping temperature could not be maintained above the adjusted solution saturation temperature*, due to complete failure of the heating system and excessively cold weather conditions outside the reactor building, the following operating procedures will be followed. The tank outlet valve would be temporarily closed and the solution drained from the pump suction line and replaced with demineralized water. This represents no new operating procedures as the system is designed to operate in this manner when it is being tested using the test tank, and no credit is taken for the storage of sodium pentaborate in the suction lines.[63] The SLC System does not rely on the heating equipment to meet its safety design basis.

The pump and system design pressure between the explosive valves and the pump discharge is 1460 psig. A relief valve is configured at the discharge of each pump which prevents exceeding USAS B31.1 Code allowable values by lifting and recirculating flow back to the pump suction during over-pressure conditions. The two relief valves are set to a nominal 1540 psig ±1% setting. To prevent bypass flow from one pump in case of relief valve failure in the line from the other pump, a check valve is installed downstream of each relief valve line in the pump discharge pipe.

The two explosive-actuated injection valves provide high assurance of opening when needed and ensure that boron will not leak into the reactor even when the pumps are being tested.

Each explosive valve is closed by a plug in the inlet chamber. The plug is circumscribed with a deep groove so the end will readily shear off when pushed with the valve plunger. This opens the inlet hole through the plug. The sheared end is pushed out of the way in the chamber; it is shaped so it will not block the ports after release.

The shearing plunger is actuated by an explosive charge with dual ignition primers, inserted in the side chamber of the valve. Ignition circuit continuity is monitored by a trickle current, and an alarm occurs in the control room if either circuit opens. Indicator lights show which primary circuit opened. To service a valve after firing, a six-inch length of pipe

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* The adjusted saturation temperature is equal to the actual saturation temperature plus 10°F.

(spool piece) must be removed immediately upstream of the valve to gain access to the shear plug.

The SLC system is actuated by separate, keylock on-off switches for each pump on the control room console. The use of keylock switches assures that switching from the "off" position is a deliberate act. Starting Pump A will open explosive valve 14A and close the inboard isolation valve of the Reactor Water Clean Up (RWCU) system to prevent loss or dilution of the boron. Similarly, starting Pump B will open explosive valve 14B and close the outboard isolation valve of the RWCU system. Separate switches for each pump allows both pumps to operate simultaneously to inject the boron. A green light in the control room indicates that power is available to the pump motor contactor and that the contactor is open (pump not running). A red light indicates that the contactor is closed (pump running).

If the pump lights, storage tank level, pump discharge pressure, or explosive valve light indicate that the liquid may not be flowing, the operator can immediately actuate the alternate pump. Cross piping and check valves assure a flow path through either pump and either explosive valve. The chosen pump(s) will start even though the local switch at the pump is in the "stop" position for test or maintenance. The pressure in the common discharge header is also indicated in the control room.

Equipment drains and tank overflow are not piped to the waste system but to separate containers (such as 55-gal. drums) that can be removed and disposed of independently to prevent any trace of boron from inadvertently reaching the reactor.

Instrumentation consisting of solution temperature indication and control, tank level, and heater system status is provided locally at the SLC storage tank.

9.4 Safety Evaluation

The standby liquid control (SLC) system is a backup to a safety system and it is required to be operable when the reactor is in MODES 1, 2, and 3. The system is never expected to be needed to maintain the unit subcritical because of the large number of independent control rods available to shut down the reactor.

The SLC system has a function in accident mitigation based on the implementation of the AST methodology as the radiological source term for the design basis Loss of Coolant Accident analysis (in accordance with RG 1.183). SLC will be used to control the pH of the water in the Suppression Pool, Reactor Vessel, and Core Cooling systems following a Design Basis LOCA. This function is required because the water in the Suppression Pool is expected to become acidic (pH<7) following a LOCA, unless chemicals are added to the Suppression Pool to raise the pH. The injection of the Sodium Pentaborate Decahydrate ($Na_2B_{10}0_{16*}10H_2O$) from the SLC system will raise the Pool pH enough to keep the pH>7.

Injection of Sodium Pentaborate using the SLC system within approximately 12 hours after a LOCA is sufficient to maintain pH above 7.0 for the 30-day duration of the accident. The SLC system is required to complete injection of Sodium Pentaborate Decahydrate within the first 8 hours following a Design Basis LOCA. This requirement supports the assumptions made in the AST LOCA Analysis, and it will ensure that the safety function is completed before the area where the SLC system is located becomes a harsh environment.

The SLC system is not a CNS Essential system. However, the non-Essential SLC system meets the design and performance requirements necessary to credit it during post-LOCA operation. The portions of the system that interface directly with essential systems are classified essential.

However, to assure the availability of the SLC system to meet its safety objective, two sets of the components (e.g. pumps and explosive valves) are provided in parallel redundancy. Redundancy is not required for the tank heater or heating cable.

The system is designed to bring the reactor from rated power to MODE 4 at any time in core life. The reactivity compensation provided will reduce reactor power from rated to zero level to allow cooling the nuclear system to normal room temperature, with the control rods remaining withdrawn in the rated power pattern. It includes the reactivity gains that result from complete decay of the rated power xenon inventory. It also includes the positive reactivity effects from eliminating steam voids, changing water density from hot to cold, reducing Doppler effect in uranium, reducing neutron leakage from boiling to cold, and decreasing control rod worth as the moderator cools. The specified minimum final concentration of boron in the reactor core includes a negative reactivity margin for calculational uncertainties and, therefore, assures a substantial shutdown margin.

The specified minimum average concentration of natural boron in the system to provide the specified shutdown margin, after operation of the SLC system, is 660 ppm (see Cycle Specific Supplemental Reload Licensing Report). The calculation of the minimum weight of sodium pentaborate (containing the boron) to be injected into the reactor is based on the required 660 ppm average concentration in the reactor coolant and the quantity of reactor coolant in the reactor vessel at Normal Water Level, recirculation loops, and the RHR system in the shutdown cooling mode at 70°F. The required concentration of 660 ppm is increased by 25 percent to allow for imperfect mixing, leakage, and volume in other small piping connected to the reactor. This minimum concentration will be achieved if the solution is prepared as defined in USAR Section III-9.3 and maintained in accordance with the volume - concentration requirements in the Technical Specifications.

The low-level setpoint for the SLC storage tank, is based on the minimum weight of sodium pentaborate in solution at the design solution concentration of 15.0 weight percent. The high level setpoint is selected to provide an adequate operating region between the high and low alarm setpoints to minimize the frequency of tank servicing. In regard to the mixing of boron solution with the SLC Storage Tank, GE Topical Report NEDE-31096-P[110] discussed three alternatives for meeting the ATWS rule. For CNS, the ATWS Rule is met by the 2-pump operation[111] alternative, and the concentration of sodium pentaborate was not increased, therefore, continuous mixing is not a requirement of the GE Topical Report[110] alternative chosen by CNS.

Cooldown of the nuclear system will require a minimum of several hours to remove the thermal energy stored in the reactor, cooling water, and associated equipment and to remove most of the radioactive decay heat. The controlled limit for the reactor vessel cooldown is 100°F per hour, and normal operating temperature is approximately 550°F. Usually, using the main condenser and various shutdown cooling systems to shut down the plant will require 10 to 24 hours before the reactor vessel is opened and much longer to reach room temperature (70°F); this is the condition of maximum reactivity and, therefore, the condition that requires the maximum concentration of boron.

The SLC system equipment necessary for injection of neutron absorber solution into the reactor is designed as Class I (seismic) for withstanding the specified earthquake loadings (see USAR Section XII and Appendix C). Nonprocess equipment such as the test tank is designed as

Class II (seismic). The system piping and equipment are designed, installed, and tested in accordance with requirements stated in Appendix C.

The SLC system is required to be operable in the event of a loss of normal station power. Therefore the pumps, valves, and controls are powered from the standby AC power supply in the absence of normal power. The pumps and valves are powered and controlled from separate buses and circuits so that a single electrical failure will not prevent system operation.

The SLC system and pumps have sufficient pressure margin to assure solution injection into the reactor at anticipated ATWS pressures (near 1100 psig), which are above the normal pressure of approximately 1030 psig in the bottom of the reactor. The SLC relief valve setting of 1540 psig ±1% ensures that when assumed inaccuracies of 3% are applied, the minimum and maximum lift pressures will be 1478 and 1602 psig respectively. Since the maximum expected system pressure for two-pump operation will not exceed 1433 psig (which includes allowances for positive displacement pump ripple effects) the valves will not lift and cause recirculation flow during the operating interval between setpoint verification tests.[104] The SLC system positive displacement pumps cannot overpressurize the nuclear system because the nuclear system relief and safety valves begin to relieve pressure above approximately 1080 to 1100 psig.

While both pumps may be operated simultaneously, only one of the two standby liquid control pumps is needed for system operation pursuant to the original design basis for the system. If one pump is found to be inoperable, there is no immediate threat to shutdown capability, and reactor operation can continue during repairs. The period while one redundant component upstream of the explosive valves may be out of operation should be consistent with the very small probability of failure of both the control rod shutdown capability and the alternate component in the SLC system, together with the fact that nuclear system cooldown takes several hours while liquid control solution injection takes approximately two hours[73]. This indicates the considerable time available for testing and restoring the SLC system to operable condition after testing, while reactor operation continues.

Compliance with the NRC ATWS Rule

Compliance with the NRC ATWS Rule 10CFR50.62 has been demonstrated by means of the equivalent control capacity concept, using the plant-specific minimum parameters. This concept requires that each boiling water reactor must have a standby liquid control system with a minimum flow capacity and boron content equivalent in control capacity of 86 gallons per minute of 13 weight percent sodium pentaborate solution (natural boron enrichment) used for the 251-inch diameter vessel studied[67]. The described minimum system parameters (76.4 gpm and 11.5 percent concentration by weight) will ensure an equivalent boron injection capacity that exceeds the ATWS Rule requirements. To achieve the required flow rate of 76.4 gpm, the system permits the simultaneous operation of both pumps (the minimum permitted flow rate of each pump is 38.2 gpm)[71,72,111]. The volume-concentration curve for the solution tank (reference Technical Specification Figure 3.1.7-1), was revised to reflect the required minimum concentration of 11.5 percent[81]. With the simultaneous operation of both positive displacement pumps, the solution can be injected into the reactor in 41 to 58 minutes, independent of the amount of solution in the tank (within the required volume range of 3132 gal.@ 16% and 4414 gal.@ 11.5%), and the pump rates (within the specified Technical Specification limits of a minimum of 38.2 gpm for each pump for a total minimum flow rate of 76.4 gpm).

The changes made to the SLC system as a result of the ATWS Rule do not invalidate the original SLC system design basis. That is, only one pump is required in order to shutdown the reactor within the specified time. The modifications, however give the operator the capability to use both SLC pumps pursuant to the ATWS Rule in the unlikely event they are needed.

Compliance with Alternate Source Term

The development of the AST methodology resulted in a change of the assumption for the distribution of the chemical forms of iodine. The AST methodology postulates that the chemical form of iodine releases from the fuel would be approximately 95% particulate cesium iodide (CsI) and 5% elemental. Following a LOCA, the water in the Suppression Pool is expected to become acidic (pH>7) unless a chemical is added to the Suppression Pool within the first 24 hours following the initiation of the LOCA. If the Suppression Pool water remains acidic, the CsI in the Suppression Pool will dissociate and the resulting elemental iodine will be released from the Suppression Pool. This would result in airborne concentrations of radioactive iodine in the containment atmosphere. It is preferable to maintain the Suppression Pool basic (pH>7) so that the iodine remains in a more soluble form as CsI. The volume of the water in the Suppression Pool post LOCA also includes the volume of water in the reactor vessel, recirculation piping and emergency core cooling systems.

The SLC system is used to maintain the Suppression Pool basic. This new function for SLC is an accident mitigation function. The AST amendment request credited the SLC system for pH control even though the SLC system is not a safety-related system and could be considered susceptible to a single active failure because the SLC system meets the NRC guidelines for the control of Suppression Pool pH following a LOCA. The guidelines are directed at those applications: (1) where the SLC system is determined to be susceptible to single failure, or (2) where the system may not be safety-related, or (3) any other case where the SLC system is missing the quality commonly required for an engineered safety function (ESF) grade system.

9.5 Inspection and Testing

Operational testing of the SLC system is performed in at least two parts to avoid inadvertently injecting boron into the reactor.

With the valves to and from the solution tank closed and the three valves to and from the test tank opened, demineralized water in the test tank can be recirculated by locally starting either pump.

The injection portion of the system can be functionally tested by valving the injection lines to the test tank and actuating the system from the control room. The selected pump's explosive injection valve opens on actuation. System operation is indicated in the control room.

After functional tests, the injection valves and explosive charges must be replaced and all the valves returned to their normal positions as indicated in Burns and Roe Drawing 2045, Sheet 2.

The circuits on the two explosive valves (one per loop) are continuously monitored and are provided with visual indication in the event of a break in continuity. One explosive valve is destructively tested at each refueling outage and a replacement valve installed. On the following refueling outage the other valve is tested in a like manner.[68]

By closing a local open valve to the reactor, the containment isolation check valves can be tested for leakage in accordance with the Technical Specifications. Position indicator lights in the control room indicate that the local valve is closed for tests or open and ready for operation.

Demineralized water is available for refilling, testing, or flushing the system.

Should the boron solution ever be injected into the reactor, either intentionally or inadvertently, then after making certain that the normal reactivity controls will keep the reactor subcritical, the boron is removed from the reactor coolant system by flushing for gross dilution followed by operating the RWCU system. There is practically no effect on reactor operations when the boron concentration has been reduced below approximately 50 ppm.

The concentration of the sodium pentaborate in the solution tank and the Suppression Pool pH are determined periodically by chemical analysis.

The standby liquid control system is in scope for License Renewal per 10 CFR 54.4(a)(1), (a)(2), and (a)(3) and was subject to aging management review. Aging effects are managed by the following Aging Management Programs: Bolting Integrity (see USAR Section K-2.1.2), External Surfaces Monitoring (see USAR Section K-2.1.14), Oil Analysis (see USAR Section K-2.1.28), Periodic Surveillance and Preventive Maintenance (see (USAR Section K-2.1.31), and Water Chemistry Control – BWR (see USAR Section K-2.1.39). There are no Time-Limited Aging Analyses that are applicable.

10.0 THERMAL-HYDRAULIC STABILITY ANALYSIS

10.1 Safety Objective

The objective of the thermal-hydraulic stability evaluation is to analytically demonstrate that thermal-hydraulic instabilities are readily detectable and able to be suppressed without compromising the integrity of the fuel or nuclear process barriers to a release of radioactivity to the environment.

10.2 Safety Design Basis

The nuclear system shall exhibit no inherent tendency toward limiting cycle oscillations or that thermal-hydraulic instabilities are able to be readily detected and suppressed.

10.3 Description and Performance Analysis

10.3.1 Introduction

The stability licensing basis is set forth in 10CFR50 Appendix A, General Design Criteria (GDC). Specifically, GDC-12 requires assurance that power oscillations (thermal-hydraulic instabilities) which can result in conditions exceeding specified acceptable fuel design limits are either not possible or can be reliably and readily detected and suppressed. The physics of Boiling Water Reactors (BWRs) make thermal-hydraulic instabilities possible, therefore to ensure compliance with GDC-12, thermal-hydraulic instabilities must be reliably and readily detected and suppressed. Thermal-hydraulic instabilities occur in the low flow and high power region of the power/flow map. Two types of thermal-hydraulic instabilities are possible in BWRs: 1) Regional (or channel) Instability and 2) Core Wide Instability. Regional Instabilities are strongly dependent on void reactivity feedback whereas Core Wide instabilities are more a function of Thermal-Hydraulic/Reactivity feedback mechanisms. Regardless of the mechanism, a thermal hydraulic instability is caused by an increase in the coolant void fraction which increases the two phase friction for flow through the affected fuel bundles. This increased friction (or resistance to flow) causes a reduction in coolant flow and results in affected fuel rods becoming steam blanketed. The steam blanket results in an increase in fuel cladding temperature and a reduction in the heat transfer from the affected fuel rods into the coolant flow. This occurs because the heat transfer coefficient for steam is less than that for water. The reduced heat transfer into the coolant allows the steam to condense back into water and re-wet the fuel cladding. This re-wetting of the fuel cladding results in a reduction in the fuel cladding temperature and an increase in heat transfer into the coolant. This process repeats itself on a characteristic frequency of 2 to 3 seconds and the cyclic thermal stress imposed on the fuel cladding can result in fuel cladding damage.

10.3.2 Regional Exclusion Methodology Description

The ODYSY methodology is used for stability licensing frequency domain calculations. This methodology was approved by the NRC as documented in Reference 116. ODYSY replaces the FABLE/BYPSS methodology originally used for stability licensing at CNS.

10.4 Ultimate Performance Limit Criteria and Conformance

10.4.1 Criteria Definition

The nuclear system shall exhibit no inherent tendency toward limiting cycle oscillations or that thermal-hydraulic instabilities are able to be readily detected and suppressed.

Acceptability of the regional exclusion methodology solution is based on the ability of the Reactor Protection System (RPS), in this case the Flow Biased Average Power Range Monitor (APRM) Neutron Flux Scram, to detect and suppress any instability prior to exceeding the Safety Limit Minimum Critical Power Ratio (SLMCPR). Inherent in this solution methodology is that regional mode instabilities are unlikely to occur. This is because the APRM System may not be able to detect a regional mode instability prior to exceeding the SLMCPR. Similarly, the regional exclusion methodology is dependent upon the ability of the RPS to detect and suppress a core wide instability before the SLMCPR is exceeded.

Cooper Nuclear Station utilizes "Regional Exclusion with Flow-Biased APRM Neutron Flux Scram" to ensure compliance to the performance limit criteria. Specifically, CNS uses an unfiltered APRM Flow Biased Neutron Flux Scram signal (as opposed to a Simulated Thermal Power Scram signal). The neutron flux scram signal is very sensitive to power oscillations resulting from core wide thermal-hydraulic instabilities and provides an instantaneous response (reactor scram) to core wide power oscillations. This APRM Flow Biased scram will suppress the core wide thermal-hydraulic instability before the SLMCPR (or any thermal limit) is exceeded.

10.4.2 Channel Hydrodynamic (Regional Mode Instability) Conformance to the Ultimate Performance Criteria

10.4.2.1 Current Analysis

The stability regional exclusion analysis was updated for introduction of Maximum Extended Load Line Limit (MELLL) Analysis and Increased Core Flow (ICF). The effect of ICF on stability is negligible. Since only limiting portions of the stability analysis were performed for the current analysis, the previous stability analysis is retained in Section 10.4.2.2. The methodology used in the current analysis is identical to the previous analysis. The current analysis shows that the channel decay ratio is below 0.56 when the core decay ratio is greater than 0.80 for all cases including at the intersection of the MELLL region and the natural circulation line where channel decay ratio is 0.473 and core decay ratio is 1.484. This confirms that core wide instabilities are much more likely to occur than regional mode instabilities for CNS.

Core and channel decay ratios were calculated for several power/flow combinations on the power to flow operating map using the same method as described for the historical analysis. These calculations resulted in a small change to the exclusion region boundary. Changes to the exclusion region are reported in the cycle-specific Core Operating Limits Report (COLR) whenever there is a change to the exclusion region boundary definition.

10.4.2.2 Previous Analysis

The previous analysis provides the initial application exclusion region and demonstrates that it is acceptable to apply the stability solution methodology described to CNS.

Integral to the regional exclusion methodology is the assertion that regional mode instabilities have a low probability of occurrence. The Stability Solutions Licensing Methodology report[107, 108] shows that the probability of regional mode instabilities becomes progressively smaller as the channel hydraulic decay ratio is decreased. Regional mode instabilities have not been observed for channel hydraulic decay ratios of less than 0.6. Under these conditions, core wide mode instabilities can be assumed to be the predominant mode as long as the channel decay ratio is below 0.56 when the core decay ratio is greater than 0.8. Analysis performed by General Electric for CNS[80] shows that the channel decay ratio is below 0.56 when the core decay ratio is greater than 0.8 for all cases except at the intersection of the Extended Load Line Limit Analysis (ELLLA) region and natural circulation line where the channel decay ratio is 0.65. However, in this case, because of the very high core decay ratio of 1.46, core wide instabilities are much more likely to occur.

A second integral part of the regional exclusion methodology is the use of an unfiltered APRM Flow Biased Neutron Flux Scram as opposed to a Simulated Thermal Power scram. The APRM neutron flux signal provides an instantaneous response to an oscillation rather than the slower fuel thermal response associated with a Simulated Thermal Power scram. Therefore, a core wide mode instability will be excited long before a regional (azimuthal) mode instability and the APRM flow biased flux scram will suppress the instability before any fuel thermal limit is reached.

Core and channel decay ratios were calculated for several power/flow combinations on the power to flow operating map (see USAR Figure III-10-1) using the method described in USAR section 10.3.2. This analysis, using these point combinations (see USAR Table III-10-2), determines the exclusion region boundary on the power/flow map and establishes core wide instabilities as the predominant instability mode. The calculations yield decay ratios as presented below (USAR Table III-10-1). Points 1 through 4 are along the ELLLA Rod Line and points 5 through 8 are along the natural circulation line.

10.4.3 Reactor Core (Core Wide Mode) Conformance to Ultimate Performance Criteria

10.4.3.1 Current Analysis

The Detect and Suppress analysis was updated for introduction of Maximum Extended Load Line Limit (MELLL) Analysis and Increased Core Flow (ICF). Since the Detect and Suppress methodology is unchanged, the previous Detect and Suppress analysis description is retained in Section 10.4.3.2. The APRM flow-biased scram was changed as a result of MELLL. The current analysis performed by General Electric for CNS shows that CNS meets all of the necessary criteria and that a core wide mode instability will be detected and terminated by the APRM flow-biased scram prior to exceeding the SLMCPR. Confirmation that SLMCPR protection is provided by the flow-biased scram for core wide mode instability is reported in the cycle-specific Supplemental Reload Licensing Report (SRLR).

10.4.3.2 Previous Analysis

The previous analysis describes the initial application Detect and Suppress analysis and demonstrates that it is acceptable to apply the stability solution methodology described to CNS.

Table III-10-1 Core and Channel Decay Ratio Results

Point Number	Power (%)	Flow (%)	Channel Hydraulic Decay Ratio	Core Decay Ratio
1	76.1	45.0	0.39	0.74
2	74.3	42.0	0.45	0.89
3	73.2	40.0	0.51	1.01
4	50.0	30.0	0.65	1.46
5	45.0	30.0	0.52	1.31
6	40.0	30.0	0.42	0.94
7	37.0	30.0	0.37	0.82
8	35.0	30.0	0.34	0.76

Table III-10-2 Coordinates of Exclusion Region Boundary

Point Number	Power (%)	Flow (%)
A	75.4	43.8
B	36.3	30.0

The points identified in USAR Table III-10-2 were used to determine the location of the Exclusion Region boundary, which is shown in USAR Figure III-10-1. Note that entry into the exclusion region above the ELLLA line is operation in a non-licensed part of the power/flow map.

The application of the Regional Exclusion Methodology is dependent upon the ability to satisfy two criteria: 1) the ability to show that regional mode instabilities are unlikely and 2) the ability to show that the RPS can readily and reliably detect and suppress core wide instabilities before the SLMCPR is exceeded. Section 10.4.2 shows that Regional Mode Instabilities are unlikely to occur at Cooper Nuclear Station. The purpose of this section is to show that the APRM system at Cooper Nuclear Station can readily and reliably detect and suppress Core Wide Instabilities before the SLMCPR is exceeded.

The Detect and Suppress licensing methodology for application to the Regional Exclusion Solution is documented in BWROG Licensing Topical Report[109]. Consistent with the qualification of Cooper Nuclear Station as a Regional Exclusion Solution plant, the Regional Exclusion methodology demonstrates that core wide instabilities are the predominant instability mode. Therefore, the Detect and Suppress calculation must only be performed for core wide mode instabilities.

The Detect and Suppress methodology assumes that a core wide instability occurs, and is terminated by automatic reactor scram when the APRM oscillations magnitude reaches the flow biased APRM flux trip setpoint. This methodology applies a statistical method, using a combination of statistical and deterministic inputs, to determine the final MCPR (FMCPR) with a high statistical confidence when control rod insertion disrupts the oscillation. The flow biased APRM flux trip provides adequate protection as long as the FMCPR is greater than the SLMCPR.

The TRACG GS3 methodology[120] replaces the Delta CPR/Initial CPR Vs. Oscillation Magnitude (DIVOM) methodology portion of the BWROG Licensing Topical Report[109]. The GS3 methodology was used to statistically determine the required MCPR margin to the Safety Limit on a generic basis; using an approved best estimate plus uncertainty evaluation. The reduction of the large conservatisms, inherent in the DIVOM methodology, with the use of GS3 causes the cycle specific stability based operating limit MCPR to be bounded by the transient analysis operating limit MCPR.

Analyses[80] performed by General Electric show that Cooper Nuclear Station meets all of the necessary criteria and that a core wide mode instability will be detected and terminated by the APRM flow biased scram prior to exceeding SLMCPR.

Also, thermal-hydraulic instabilities only occur at high power and low flow conditions. This is the region where conservative decay ratio calculations indicate that thermal-hydraulic instabilities are possible. Intentional entry into this region of the power/flow map (see USAR Figure III-10-1) is administratively prohibited and an immediate exit is required for unintentional entries. Additionally, Cooper Nuclear Station uses an on-line stability monitor and predictor that provides an analytical determination of the core decay ratio for any operating condition.

10.5 Conclusion

Analysis of the Stability of the Nuclear System demonstrates that it can be operated safely, even in highly improbable and unusual modes, without danger of compromising any radioactive material barriers or fuel safety limits because of instability.

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