DUKE POWER COMPANY
POWER BUILDING
422 SOUTH CHURCH STREET, CHARLOTTE, N. C. 28242

WILLIAM O. PARKER, JR.
VICE PRESIDENT
STEAM PRODUCTION

December 21, 1978

TELEPHONE: AREA 704
373-4083

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. Robert L. Baer, Chief
Light-Water Reactor Project Branch #2

Reference: McGuire Nuclear Station
Units 1 and 2
Docket Nos. 50-369, 50-370

Dear Mr. Denton:

Duke Power Company is filing herewith Amendment No. 59 to its Application for Licenses for the McGuire Nuclear Station which is under construction pursuant to Construction Permits CPPR-83 and -84. This filing includes three signed original copies of the amendment with an attachment entitled, "Revision 35" to the Final Safety Analysis Report." This filing also includes 15 additional conformed copies of the amendment with an additional 45 copies of Revision 35 to the Final Safety Analysis Report.

The purpose of Amendment 59 is to revise the FSAR to include the response to Mr. Robert L. Baer's letter of November 6, 1978 concerning the fuel building ventilation system. Also included in this amendment are descriptions of several design changes that have been made including:

a) Deletion of part length control rods
b) Revised instrumentation for main steam line break protection
c) Addition of load follow capability (modified D bank configuration)

Other miscellaneous revisions to the FSAR are also included.

Copies of this Amendment 59 have been distributed in accordance with the requirements of Part 2.101 of the Commission's regulations and the instructions contained in Mr. Roger S. Boyd's letter of August 6, 1976.

Respectfully submitted,

William O. Parker, Jr.

William O. Parker, Jr.

GAC/sch

Attachments

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WILLIAM O. PARKER, JR., being duly sworn, states that he is Vice President of Duke Power Company; that he is authorized on the part of said Company to sign and file with the Nuclear Regulatory Commission this Amendment 59 to its application and documents appended thereto; and that all statements and matters set forth therein are true and correct to the best of his knowledge.

William O. Parker, Jr.

William O. Parker, Jr., Vice President

Subscribed and sworn to before me this 21st day of December, 1978.

Vivian B. Robbins

Vivian B. Robbins, Notary Public
(Notarial Seal)

My Commission Expires:

February 15. 1982

DUKE POWER COMPANY
AMENDMENT NO. 59
APPLICATION FOR LICENSES
DOCKET NO. 50-369, -370

MCGUIRE NUCLEAR STATION
FINAL SAFETY ANALYSIS REPORT

Revision 35
December 21, 1978

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4.0 REACTOR

4.1 SUMMARY DESCRIPTION

This chapter describes 1) the mechanical components of the reactor and reactor core including the fuel rods and fuel assemblies, reactor internals, and the control rod drive mechanisms, 2) the nuclear design, and 3) the thermal-hydraulic design.

The reactor core is comprised of an array of fuel assemblies which are identical in mechanical design, but different in fuel enrichment. Three enrichments are employed in the initial core.

The core is cooled and moderated by light water at a pressure of 2250 psia in the Reactor Coolant System. The moderator coolant contains boron as a neutron poison. The concentration of boron in the coolant is varied as required to control relatively slow reactivity changes including the effects of fuel burnup. Additional boron, in the form of burnable poison rods, is employed in the first core to establish the desired initial reactivity.

Two hundred and sixty-four fuel rods are mechanically joined in a square array to form a fuel assembly. The fuel rods are supported in intervals along their length by grid assemblies which maintain the lateral spacing between the rods throughout the design life of the assembly. The grid assembly consists of an
3 "egg-crate" arrangement of interlocked straps. The straps contain spring fingers and dimples for fuel rod support as well as coolant mixing vanes. The fuel rods consist of slightly enriched uranium dioxide ceramic cylindrical pellets contained in slightly cold worked Zircaloy-4 tubing which is plugged and seal welded at the ends to encapsulate the fuel. All fuel rods are pressurized with helium during fabrication to reduce stresses and strains to increase fatigue life.

The center position in the assembly is reserved for the in-core instrumentation, while the remaining 24 positions in the array are equipped with guide thimbles joined to the grids and the top and bottom nozzles. Depending upon the position of the assembly in the core, the guide thimbles are used as core locations for rod cluster control assemblies, neutron source assemblies, and burnable poison rods. Otherwise, the guide thimbles are fitted with plugging devices to limit bypass flow.

The bottom nozzle is a box-like structure which serves as a bottom structural element of the fuel assembly and directs the coolant flow distribution to the assembly.

The top nozzle assembly functions as the upper structural element of the fuel assembly in addition to providing a partial protective housing for the rod cluster control assembly or other components.

The rod cluster control assemblies each consist of a group of individual absorber rods fastened at the top end to a common hub or spider assembly. The rods in these assemblies contain absorber material to control the
35 reactivity of the core under operating conditions to control axial power distribution.

The control rod drive mechanisms for the full length rod cluster control assemblies are of the magnetic latch type. The latches are controlled by
3 three magnetic coils. They are so designed that upon a loss of power to the coils, the rod cluster control assembly is released and falls by gravity to shutdown the reactor.

35 The control rod drive mechanisms for the part length control rods are no longer used since the part length rods have been removed.

The components of the reactor internals are divided into three parts consisting of the lower core support structure (including the entire core barrel and neutron shield pad assembly), the upper core support structure and the in-core instrumentation support structure. The reactor internals support the core, maintain fuel alignment, limit fuel assembly movement, maintain alignment between fuel assemblies and control rod drive mechanisms, direct coolant flow past the fuel elements and to the pressure vessel head, provide gamma and neutron shielding, and provide guides for the in-core instrumentation.

3 The nuclear design analyses and evaluation establish physical locations for control rods and burnable poisons and physical parameters such as fuel enrichments and boron concentration in the coolant such that the reactor core has inherent characteristics which together with corrective actions of the reactor control, protective and emergency cooling systems provide adequate reactivity control even if the highest reactivity worth rod cluster control assembly is stuck in the fully withdrawn position.

The design also provides for inherent stability against diametral and azimuthal power oscillations and for control of induced axial power oscillations through
35 the use of the full length control rods.

The thermal-hydraulic design analyses and evaluation establish coolant flow parameters which assure that adequate heat transfer is assured between the fuel cladding and the reactor coolant. The thermal design takes into account local variations in dimensions, power generation, flow distribution and mixing. The mixing vanes incorporated in the fuel assembly spacer grid design induces additional flow mixing between the various flow channels within a fuel assembly as well as between adjacent assemblies.

Instrumentation is provided in and out of the core to monitor the nuclear,
3 thermal-hydraulic, and mechanical performance of the reactor and to provide inputs to automatic control functions.

The reactor core design together with corrective actions of the reactor control protection and emergency cooling systems can meet the reactor performance and safety criteria specified in Section 4.2.

To illustrate the effects of the change in fuel design, Table 4.1-1 presents a comparison of the principal nuclear, thermal-hydraulic and mechanical design parameters between McGuire Units 1 and 2 and the reference plant, both with 17 x 17 fuel assemblies including fuel densification effect.

The effects of fuel densification for standard Westinghouse fuel were evaluated
with the methods described in Reference 1. If fuel is produced for the McGuire
units by means other than those which formed the basis for Reference 1, specifi-
cations are followed to ensure that the effects of densification would be no
16 greater than has been allowed for in the design. The specifications for quanti-
fying the extent of densification are based on the NRC draft regulatory guide
on fuel densification, Reference 2.

The analysis techniques employed in the core design are tabulated in Table 4.1-2.
The loading conditions considered in general for the core internals and components
are tabulated in Table 4.1-3. Specific or limiting loads considered for design
purposes of the various components are listed as follows: fuel assemblies in Sub-
3 division 4.2.1.1.2; reactor internals in Subdivision 4.2.2.3 and Subsection 5.2.1;
neutron absorber rods, burnable poison rods, neutron source rods and thimble
35 plug assemblies in Subdivision 4.2.3.1.3; full-length control rod drive
mechanisms in Subdivision 4.2.3.1.4. The dynamic analyses, input forcing func-
tions, and response loadings are presented in Section 3.9.

3 4.1.1 REFERENCES

1. Hellman, J. M., (Ed.), Fuel Densification Experimental Results and Model for Reactor Application, WCAP-8219, October 1973.

16 2. Core Performance Branch, U. S. Nuclear Regulatory Commission, "The Analysis of Fuel Densification, Draft, November 1975.

4.2 MECHANICAL DESIGN

The unit conditions for design are divided into four categories in accordance with their anticipated frequency of occurrence and risk to the public: Condition I - Normal Operation; Condition II - Incidents of Moderate Frequency; Condition III - Infrequent Incidents; Condition IV - Limiting Faults.

The reactor is designed so that its components meet the following performance and safety criteria:

1. The mechanical design of the reactor core components and their physical arrangement, together with corrective actions of the reactor control, protection, and emergency cooling systems (when applicable) assure that:

   1. Fuel damage* is not expected during Condition I and Condition II events. It is not possible, however, to preclude a very small number of rod failures. These are within the capability of the unit cleanup system and are consistent with the unit design bases.

   b. The reactor can be brought to a safe state following a Condition III event with only a small fraction of fuel rods damaged* although sufficient fuel damage might occur to preclude resumption of operation without considerable outage time.

   c. The reactor can be brought to a safe state and the core can be kept subcritical with acceptable heat transfer geometry following transients arising from Condition IV events.

2. The fuel assemblies are designed to accommodate expected conditions for design for handling during assembly inspection and refueling operations and shipping loads.

3. The fuel assemblies are designed to accept control rod insertions in order to provide the required reactivity control for power operations and reactivity shutdown conditions.

4. All fuel assemblies have provisions for the insertion of in-core instrumentation necessary for unit operation.

5. The reactor internals in conjunction with the fuel assemblies direct reactor coolant through the core to achieve acceptable flow distribution and to restrict bypass flow so that the heat transfer performance requirements can be met for all modes of operation. In addition, the internals provide core support and distribute coolant flow to the pressure vessel head so that the temperature differences between the vessel flange and head do not result in leakage from the flange during the Condition I and II modes of operation. Required in-service inspection can be carried out as the internals are removable and provide access to the inside of the pressure vessel.

*Fuel Damage as used here is defined as penetration of the fission product barrier (i.e. the fuel rod clad).

Allowable Deflections

For normal operating conditions, downward vertical deflection of the lower core support plate is negligible.

For the loss of coolant accident plus the 1/2 safe shutdown earthquake condition, the deflection criteria for critical internal structures are the limiting values given in Table 4.2.2-1. The corresponding no loss of function limits are included in Table 4.2.2-1 for comparison purposes with the allowed criteria.

The criteria for the core drop accident are based upon analyses which have been performed to determine the total downward displacement of the internal structures following a hypothesized core drop resulting from loss of the normal core barrel supports. The initial clearance between the secondary core support structures and the reactor vessel lower head in the hot condition is approximately one half inch. An additional displacement of approximately 3/4 inch would occur due to strain of the energy absorbing devices of the secondary core support; thus the total drop distance is about 1-1/4 inches which is insufficient to permit the grips of the rod cluster control assembly to come out of the guide thimble in the fuel assemblies.

Specifically, the secondary core support is a device which will never be used, except during a hypothetical accident of the core support (core barrel, barrel flange, etc.). There are 4 supports in each reactor. This device limits the fall of the core and absorbs the energy of the fall which otherwise would be imparted to the vessel. The energy of the fall is calculated assuming a complete and instantaneous failure of the primary core support and is absorbed during the plastic deformation of the controlled volume of stainless steel, loaded in tension. The maximum deformation of this austenitic stainless piece is limited to approximately 15 percent, after which a positive step is provided to insure support.

For additional information on design loading categories, see Section 3.9.

4.2.2.5 Design Criteria Basis

The basis for the design stress and deflection criteria is identified below:

Allowable Stress

For normal operating conditions, Section III of the ASME Nuclear Power Plant Component Code is used as a basis for evaluating acceptability of calculated stresses. Both static and alternating stress intensities are considered. Under code case 1618 bolt material type 316 Stainless Steel is now covered in ASME Section III and is so treated. It should be noted that the allowable stresses in Section III of the ASME Code are based on unirradiated material properties. In view of the fact that irradiation increases the strength of the 304 stainless steel used for the internals, although decreasing its elongation, it is considered that use of the allowable stresses in Section III is appropriate and conservative for irradiated internal structures.

The allowable stress limits during the design basis accident used for the core support structures are based on the January 1971 draft of the ASME Code for Core Support Structures, Subsection NG, and the Criteria for Faulted Conditions.

4.2.3 REACTIVITY CONTROL SYSTEM

4.2.3.1 Design Bases

Bases for temperature, stress on structural members, and material compatibility are imposed on the design of the reactivity control components.

4.2.3.1.1 Design Stresses

The reactivity control system is designed to withstand stresses originating from various operating conditions as summarized in Table 5.2.1-2.

Allowable Stresses: For normal operating conditions Section III of the ASME Boiler and Pressure Vessel Code is used. All components are analyzed as
3 Class I components under Article NB-3000.

Dynamic Analysis: The cylic stress due to dynamic loads and deflections are combined with the stresses imposed by loads from component weights, hydraulic forces and thermal gradients for the determination of the total stresses of the reactivity control system.

4.2.3.1.2 Material Compatibility

Materials are selected for compatibility in a Pressurized Water Reactor environment, for adequate mechanical properties at room and operating temperature, for resistance to adverse property changes in a radioactive environment, and for compatibility with interfacing components.

4.2.3.1.3 Reactivity Control Components

The reactivity control components are subdivided into two categories:

1. Permanent devices used to control or monitor the core and,

2. Temporary devices used to control or monitor the core.

35 The permanent type components are the full-length rod cluster control assemblies, control rod drive assemblies, neutron source assemblies, and thimble plug assemblies. Although the thimble plug assembly does not directly contribute to the reactivity control of the reactor, it is presented
3 as a reactivity control system component in this document because it is needed to restrict bypass flow through those thimbles not occupied by absorber, source or burnable poison rods.

The temporary component is the burnable poison assembly which is normally used only in the initial core. The design bases for each of the mentioned components are in the following paragraphs.

Absorber Rods

The following are considered design conditions under Article NB-3000 of the ASME Boiler and Pressure Vessel Code Section III. The control rod which is cold rolled 304 stainless is the only non code material used in the control rod assembly. The stress intensity limit S_m for this material is defined at 2/3 of the 0.2% offset yield stress.

1. The external pressure equal to the Reactor Coolant System operating pressure.

2. The wear allowance equivalent to 1,000 reactor trips.

3. Bending of the rod due to a misalignment in the guide tube.

4. Forces imposed on the rods during rod drop.

5. Loads caused by accelerations imposed by the control rod drive mechanism.

6. Radiation exposure for maximum core life.

The absorber material temperature shall not exceed its melting temperature (1470 F for Ag-In-Cd absorber material) (Reference 13).

Burnable Poison Rods

The burnable poison rod clad is designed as a class 1 component under Article NB-3000 of the ASME Boiler and Pressure Vessel Code, Section III, 1973 for Conditions I and II. For abnormal loads during Conditions III and IV code stresses are not considered limiting. Failures of the burnable poison rods during these conditions must not interfere with reactor shutdown or emergency cooling of the fuel rods.

The burnable poison absorber material is non-structural. The structural elements of the burnable poison rod are designed to maintain the absorber geometry even if the absorber material is fractured. The rods are designed so that the absorber material is below its softening temperature (1492 F* for reference 12.5 w/o boron rods). In addition, the structural elements are designed to prevent excessive slumping.

Neutron Source Rods

The neutron source rods are designed to withstand the following:

* Borosilicate glass is accepted for use in burnable poison rods if the softening temperature is 1510 + 18 F. The softening temperature is defined in ASTM C338.

1. The external pressure equal to the Reactor Coolant System operating pressure and

2. An internal pressure equal to the pressure generated by released gases over the source rod life.

Thimble Plug Assembly

The thimble plug assemblies satisfy the following:

1. Accommodate the differential thermal expansion between the fuel assembly and the core internals,

2. Maintain positive contact with the fuel assembly and the core internals.

3 3. Be inserted into or withdrawn from the fuel assembly by a force not exceeding 25 pounds.

4. Provide a flow path from the bottom of the UHI to the fuel assemblies during a postulated LOCA.

4.2.3.1.4 Control Rod Drive Mechanisms

The mechanisms are Class I components designed to meet the stress requirements for normal operating conditions of Section III of the ASME Boiler and Pressure Vessel Code. Both static and alternating stress intensities are considered. The stresses originating from the required design transients are included in
6 the analysis. See Subdivision 5.2.1.5 for transient details. Q 110.15

A dynamic seismic analysis is required on the full length control rod drive mechanism when a seismic disturbance has been postulated to confirm the ability of the mechanism to meet ASME Code, Section III allowable stresses and to confirm its ability to trip when subjected to the seismic disturbance.

3

The control rod drive mechanism (CRDM) design used for the 17X17 fuel assembly control rod is identical to the 15X15 control rod drive mechanism. The seismic analysis and response of the 17X17 control rod drive mechanism will be identical to those of the 15X15 mechanism.

35

Full Length Control Rod Drive Mechanism Operational Requirements

The basic operational requirements for the full length control rod drive mechanisms are:

1. 5/8 inch step,

3 2. 150 inch travel,

3. 360 pound maximum load,

4. Step in or out at 45 inches/min (72 steps/min),

5. Power interruption shall initiate release of drive rod assembly,

6. Trip delay of less than]50 ms - Free fall of drive rod assembly shall begin less than]50 ms after power interruption no matter what holding or stepping action is being executed with any load and coolant temperatures of 100°F to 550°F.

3

7. 40 year design life with normal refurbishment,

8. 28,000 complete travel excursions which is]3 x 10 steps with normal refurbishment.

35

4.2.3.2 Design Description

Reactivity control is provided by neutron absorbing rods and a soluble chemical neutron absorber (boric acid). The boric acid concentration is varied to control long-term reactivity changes such as:

3 1. Fuel depletion and fission product buildup.

2. Cold to hot, zero power reactivity change.

3. Reactivity change produced by intermediate-term fission products such as xenon and samarium.

4. Burnable poison depletion.

Chemical and Volume Control is covered in Subsection 9.3.4.

The rod cluster control assemblies provide reactivity control for:

1. Shutdown.

2. Reactivity changes due to coolant temperature changes in the power range.

3 3. Reactivity changes associated with the power coefficient of reactivity.

4. Reactivity changes due to void formation.

The first fuel cycle contains more excess reactivity than subsequent cycles due to the loading of all fresh (unburned) fuel. If soluble boron were the sole means of control, the moderator temperature coefficient would be positive. It is desirable to have a negative moderator temperature coefficient throughout the entire cycle in order to reduce possible deleterious effects caused by a positive coefficient during loss of coolant or loss of flow accidents. This is accomplished by installation of burnable poison assemblies.

The neutron source assemblies provide a means of monitoring the core during periods of low neutron activity.

35 The most effective reactivity control components are the full length rod cluster control assemblies and their corresponding drive rod assemblies which are the only kinetic parts in the reactor. Figure 4.2.3-1 identifies the full length rod cluster control and drive rod assembly, in
3 addition to the arrangement of these components in the reactor relative to the interfacing fuel assembly, guide tubes, and control rod drive mechanism. In the following paragraphs, each reactivity control component is described in detail.

35 The guidance system for the full-length control rod cluster is provided by the guide tube as shown in Figure 4.2.3-1. The guide tube provides two regimes of guidance. 1) In the lower section a continuous guidance system provides support immediately above the core. This system protects the rod against excessive deformation and wear due to hydraulic loading. 2) The region above the continuous section provides support and guidance at uniformly spaced intervals.

3 The envelope of support is determined by the pattern of the control rod cluster as shown in Figure 4.2.3-2. The guide tube assures alignment and support of the control rods, spider body, and drive rod while maintaining trip times at or below required limits.

4.2.3.2.1 Reactivity Control Components

<u>Full Length Rod Cluster Control Assembly</u>

The full length rod cluster control assemblies are divided into two categories:

control and shutdown. The control groups compensates for reactivity changes due to variations in operating conditions of the reactor, i.e., power and temperature variations. Two criteria have been employed for selection of the control groups. First the total reactivity worth must be adequate to meet the nuclear requirements of the reactor. Second, in view of the fact that some of these rods may be partially inserted at power operation, the total power peaking factor should be low enough to ensure that the power capability is met. The control and shutdown groups provide adequate shutdown margin which is defined as the amount by which the core would be subcritical at hot shutdown if all rod cluster control assemblies are tripped assuming that the highest worth assembly remains fully withdrawn and assuming no changes in xenon or boron concentration or part length rod cluster control position.

A rod cluster control assembly comprises a group of individual neutron absorber rods fastened at the top end to a common spider assembly, as illustrated in Figure 4.2.3-2.

The absorber materials used in the control rods is silver-indium-cadmium alloy which is essentially "black" to thermal neutrons and has sufficient additional resonance absorption to significantly increase its worth. The alloy is in the form of extruded rods which are sealed in stainless steel tubes to prevent the rods from coming in direct contact with the coolant. In construction, the silver-indium-cadmium rods are inserted into cold-worked stainless steel tubing which is then sealed at the bottom and the top by welded end plugs as shown in Figure 4.2.3-3. Sufficient diametral and end clearance is provided to accommodate relative thermal expansions.

The bottom plugs are made bullet-nosed to reduce the hydraulic drag during reactor trip and to guide smoothly into the dashpot section of the fuel assembly guide thimbles. The upper plug is threaded for assembly to the spider and has a reduced end section to make the joint more flexible.

The spider assembly is in the form of a central hub with radial vanes containing cylindrical fingers from which the absorber rods are suspended. Handling detents and detents for connection to the drive rod assembly are machined into the upper end of the hub. A coil spring inside the spider boyd absorbs the impact energy at the end of a trip insertion. The radial vanes are joined to the hub and the fingers are joined to the vanes by furnace brazing. A centerpost which holds the spring and its retainer is threaded into the hub within the skirt and welded to prevent loosening in service. All components of the spider assembly are made from type 304 and 308 stainless steel except for the retainer which is of 17-4 PH material and the springs which are Inconel 718 alloy or oil tempered carbon steel where the springs do not contact the coolant.

The absorber rods are fastened securely to the spider to assure trouble free service. The rods are first threaded into the spider fingers and then pinned to maintain joint tightness, after which the pins are welded in place. The end plug below the pin position is designed with a reduced section to permit flexing of the rods to correct for small operating or assembly misalignments.

The overall length is such that when the assembly is withdrawn through its full travel the tips of the absorber rods remain engaged in the guide thimbles

so that alignment between rods and thimbles is always maintained. Since
3 the rods are long and slender, they are relatively free to conform to any
small misalignments with the guide thimble.

35

Burnable Poison Assembly

Each burnable poison assembly consists of burnable poison rods attached to a hold down assembly. Burnable poison assemblies are shown in Figure 4.2.3-5.

The poison rods consist of borosilicate glass tubes contained within Type
304 stainless steel tubular cladding which is plugged and seal welded at
3 the ends to encapsulate the glass. The glass is also supported along the
length of its inside diameter by a thin wall tubular inner liner of Type
304 stainless steel. The top end of the liner is open to permit the diffused
helium to pass into the void volume and the liner overhangs the glass. The
liner has an outward flange at the bottom end to maintain the position of
the liner with the glass. A typical burnable poison rod is shown in
longitudinal and transverse cross-sections in Figure 4.2.3-6.

35

Full Length Control Rod Drive Mechanism

Control rod drive mechanisms are located on the dome of the reactor vessel. They are coupled to rod control clusters which have absorber material over the entire length of the control rods and derive their name from this feature. The full length control rod drive mechanism is shown in Figure 4.2.3-10 and schematically in Figure 4.2.3-11.

The primary function of the full length control rod drive mechanism is to insert or withdraw rod control clusters within the core to control average core temperature and to shut down the reactor.

The full length control rod drive mechanism is a magnetically operated jack. A magnetic jack is an arrangement of three electro-magnets which are energized in a controlled sequence by a power cycler to insert or withdraw rod control clusters in the reactor core in discrete steps.

The control rod drive mechanism consists of four separate subassemblies. They are the pressure vessel, coil stack assembly, the latch assembly, and the drive rod assembly.

1. The pressure vessel includes a latch housing a rod travel housings which are connected by a threaded, seal welded, maintenance joint which facilitates replacement of the latch assembly. The closure at the top of the rod travel housing is a threaded plug with a canopy seal weld for pressure integrity.

   The latch housing is the lower portion of the vessel and contains the latch assembly. The rod travel housing is the upper portion of the vessel and provides space for the drive rod during its upward movement
3 as the control rods are withdrawn from the core.

2. The coil stack assembly includes the coil housing, an electrical condiut and connector, and three operating coils; 1) the stationary gripper coil, 2) the moveable gripper coil, and 3) the lift coil.

   The coil stack assembly is a separate unit which is installed on the drive mechanism by sliding it over the outside of the latch housing. It rests on the base of the latch housing without mechanical attachment.

   Energizing of the operation coils causes movement of the pole pieces and latches in the latch assembly.

3. The latch assembly includes the guide tube, stationary pole pieces, moveable pole pieces, and two sets of latches; 1) the moveable gripper latch, and 2) the stationary gripper latch.

   The latches engage grooves in the drive rod assembly. The moveable gripper latches are moved up of down in 5/8 inch steps by the lift pole to raise or lower the drive rod. The stationary gripper latches hold the drive rod assembly while the moveable gripper latches are repositioned for the next 5/8 inch step.

4. The drive rod assembly includes a flexible coupling, a drive rod, a disconnect button, a disconnect rod, and a locking button.

The drive rod has 5/8 inch grooves which receive the latches during holding or moving of the drive rod. The flexible coupling is attached to the drive rod and produces the means for coupling to the rod control cluster assembly.

The disconnect button, disconnect rod, and locking button provide positive locking of the coupling to the rod control cluster assembly and permits remote disconnection of the drive rod.

The control rod drive mechanism is a trip design. Tripping can occur during any part of the power cycler sequencing if power to the coils is interrupted.

The control rod drive mechanism is threaded and seal welded on an adaptor on top of the reactor vessel and is coupled to the rod control cluster assembly directly below.

The mechanism is capable of handling a 360 pound load, including the drive rod weight, at a rate of 45 inches/minute. Withdrawal of the rod control cluster is accomplished by magnetic forces while insertion is by gravity.

The mechanism internals are designed to operate in 650°F reactor coolant. The pressure vessel is designed to contain reactor coolant at 650°F and 2500 psia. The three operating coils are designed to operate at 392°F with forced air cooling required to maintain that temperature.

The full length control rod drive mechanism shown schematically in Figure 4.2.3-11 withdraws and inserts its control rod as electrical pulses are received by the operator coils. An ON or OFF sequence, repeated by silicon controlled rectifiers in the power programmer, causes either withdrawal or insertion of the control rod. Position of the control rod is measured by 42 discrete coils mounted on the position indicator assembly surrounding the rod travel housing. Each coil magnetically senses the entry and presence of the top of the ferro-magnetic drive rod assembly as it moves through the coil center line.

During unit operation the stationary gripper coil of the drive mechanism holds the control rod withdrawn from the core in a static position until the moveble gripper coil is energized.

Rod Cluster Control Assembly Withdrawal

The control rod is withdrawn by repetition of the following sequence of events:

1. Movable Gripper Coil (B) - ON

    The latch locking plunger raises and swings the movable gripper latches into the drive rod assembly groove. A 1/16 inch axial clearance exists between the latch teeth and the drive rod.

2. Stationary Gripper Coil (A) - OFF

    The force of gravity, acting upon the drive rod assembly and attached control rod, causes the stationary gripper latches and plunger to move

If power to the stationary gripper coil is cut off, the combined weight of
the drive rod assembly and the rod cluster control assembly is sufficient
to move latches out of the drive rod assembly groove. The control rod falls
by gravity into the core. The trip occurs as the magnetic field, holding
the stationary gripper plunger half against the stationary gripper pole,
4 collapses and the stationary gripper plunger half is forced down by the
weight acting upon the latches. After the drive rod assembly is released
by the mechanism, it falls freely until the control rods enter the buffer
section of their thimble tubes.

35

(Deleted)

(Deleted)

The rotational energy is supplied in sequential pulses to the armature which
35 rotates directionally 15 degrees per pulse as controlled by the power supply.

4.2.3.3 Design Evaluation

4.2.3.3.1 Reactivity Control Components

The components are analyzed for loads corresponding to normal, upset, emergency and faulted conditions. The analysis performed depends on the mode of operation under consideration.

The scope of the analysis requires many different techniques and methods, both static and dynamic.

Some of the loads that are considered on each component where applicable are as follows:

1. Control Rod Scram (equivalent static load)
2. Differential Pressure
3. Spring Preloads
4. Coolant Flow Forces (static)
5. Temperature Gradients
6. 3 Differences in thermal expansion
    a. Due to temperature differences
    b. Due to expansion of different materials
7. Interference between components
8. Vibration (mechanically or hydraulically induced)
9. All operational transients listed in Table 5.2.1-2
10. Pump Overspeed
11. Seismic Loads (Safe shutdown earthquake and 1/2 safe shutdown earthquake).

The main objective of the analysis is to satisfy allowable stress limits,
to assure an adequate design margin, and to establish deformation limits which
are concerned primarily with the functioning of the components. The stress
limits are established not only to assure that peak stresses will not reach
unacceptable values, but also limit the amplitude of the oscillatory stress
component in consideration of fatigue characteristics of the materials.
Standard methods of strength of meterials are used to establish the stresses
and deflections of these components for the control rod drive mechanism.
These stress limits are established by their manufacturer and translated into
allowable bending moments which may result from a seismic disturbance. Verifi-
cation is then provided that earthquake induced bending moments are below the
6 maximum tolerable umbrella described above. The analytical procedure used for Q
this verification is a normal mode/seismic response spectrum linear analysis 110.15
of a lumped parameter finite element model of the CRDM system. A typical
comparison for the system is given in Figure 4.2.3-15. The dynamic behavior
of the reactivity control components has been studied using experimental test
data (D-loop, Section 1.5) and experience from operating reactors.

3

The design of reactivity component rods provides a sufficient cold void volume within the burnable poison and source rods to limit the internal pressures to a value which satisfies the criteria in Subdivision 4.2.3.1.

The void volume for the helium in the burnable poison rods is obtained through the use of glass in tubular form which provides a central void along the length of the rods. Helium gas is not released by the neutron absorber rod material thus the absorber rod only sustains an external pressure during operating conditions. The internal pressure of source rods continues to increase from ambient until end of life at which time the internal pressure never exceeds that allowed by the criteria in Subdivision 4.2.3.1. The stress analysis of reactivity component rods assumed 100% gas release to the rod void volume, considers the initial pressure within the rod, and assumes the pressure external to the component rod is zero.

Based on available data for properties of the borosilicate glass and on nuclear and thermal calculations for the burnable poison rods, gross swelling or cracking of the glass tubing is not expected duringoperation. Some minor creep of the glass at the hot spot on the inner surface of the tube could occur but would continue only until the glass came in contact with the inner liner. The wall thickness of the inner liner is sized to provide adequate support in the event of slumping and to collapse locally before rupture of the exterior cladding if unexpected large volume changes due to swelling or cracking should occur. The top of the inner liner is open to allow communication to the central void by the helium which diffuses out of the glass.

Sufficient diametral and end clearances have been provided in the neutron absorber, burnable poison, and source rods to accommodate the relative thermal expansions between the enclosed material and the surrounding clad and end plugs. There is no bending or warping induced in the rods although the clearance offered by the guide thimble would permit a postulated warpage to occur without restraint on the rods. Bending, therefore, is not considered in the analysis of the rods. The radial and axial temperature profiles have been determined by considering gap conductance, thermal expansion, and
3 neutron and/or gamma heating of the contained material as well as gamme heating of the clad. The maximum neutron absorber material temperature was found to be less then 850°F which occurs axially at only the highest flux region. The maximum borosilicate glass temperature was calculated to be about 1200°F and takes place following the initial rise to power. The glass temperature then decreases rapidly for the following reasons: 1) reduction in power generation due to B depletion; 2) better gap conductance as the helium produced diffuses to the gap; and 3) external gap reduction due to borosilicate glass creep. Rod, guide thimble, and dashpot flow analysis performed indicates that the flow is sufficient to prevent coolant boiling and maintain clad temperatures at which the clad material has adequate strength to resist coolant operating pressure and rod internal pressures.

Analysis on the full length rod cluster control spider indicates the spider is structurally adequate to withstand the various operating loads including the higher loads which occur during hte drive mechanism stepping action and rod drop. Experimental verification of the spider structural capability is planned (see Section 1.5).

35

The materials selected are considered to be the best available from the stand point of resistance to irradiation damage and compatibility to the reactor environment. The materials selected partially dictate the reactor environment (e.g., Cl control in the coolant). The current design type reactivity controls have been inservice for as much as six years with no apparent degradation of
3 construction materials.

With regard to the material of construction exhibiting satisfactory resistance to adverse property changes in a radioactive environment, it should be noted

That work or breeder reactors in current design utilize similar materials. At high fluences the austenitic materials increase in strength with a corresponding decreased ductility (as measured by tensile tests) but energy absorption (as measured by impact tests) remain quite high. Corrosion of the materials exposed to the coolant is quite low and proper control of Cl and O_2 in the coolant will prevent the occurrence of stress corrosion. All of the
3 austenitic stainless steel base materials used are processed and fabricated to preclude sensitization. Although the control rod spiders are fabricated by furnace brazing, the procedure used requires that the pieces be rapidly cooled so that the time-at-temperature is minimized. The time that is spent by the control rod spiders in the sensitization range, 800 - 1500°F, is not more than 0.2 hours, as a maximum, during fabrication to preclude sensitization. The 17-4 PH parts are all aged at the highest standard aging temperature of 1100°F to avoid stress corrosion problems exhibited by aging at lower temperatures.

35 Analysis of the full length rod cluster control assemblies show that if the drive mechanism housing ruptures the rod cluster control assembly will be ejected from the core by the pressure differential of the operating pressure and ambient pressure across the drive rod assmeblv. The ejection is also predicted on the failure of the drive mechanism to retain the drive rod/rod cluster control assembly position. It should be pointed out that a drive mechanism housing rupture will cause the ejection of only one rod cluster control assembly with the other assemblies remaining in the core. Analysis also showed that a pressure drop in excess of 4000 psi must occur across a two-fingered vane to break the vane/spider body joint causing ejection of two neutron absorber rods from the core. Since the greatest pressure of the primary system coolant is only 2250 psi, a pressure drop in excess of 4000 psi could not be expected to occur. Thus, the ejection of the neutron absorber rods is not possible.

Ejection of a burnable poison or thimble plug assembly is conceivable based on the postulation that the hold down bar fails and that the base plate and burnable poison rods are severely deformed. In the unlikely event that failure of the hold down bar occurs, the upward displacement of the burnable poison assembly only permits the base plate to contact the upper core plate. Since this displacement is small, the major portion of the borosilicate glass
3 tubing remains positioned within the core. In the case of the thimble plug assembly, the thimble plugs will partially remain in the fuel assembly guide thimbles thus maintaining a majority of the desired flow impedance. Further displacement or complete ejection would necessitate the square base plate and burnable poison rods be forced, thus plastically deformed, to fit up through a small diameter hole. It is expected that this condition requires a substantially higher force or pressure drop than that of the hold down bar failure.

Experience with control rods, burnable poison rods, and source rods is discussed in Reference 3.

The mechanical design of the reactivity control components provides for the protection of the active elements to prevent the loss of control capability and functional failure of critical components. The components have been reviewed for potential failure and consequences of a functional failure of critical parts. The results of the review are summarized below.

35 Full Length Rod Cluster Control Assembly

1. The basic absorbing material is sealed from contact with the primary coolant and the fuel assembly and guidance surfaces by a high quality stainless steel clad. Potential loss of absorber mass or reduction in reactivity control material due to mechanical or chemical erosion or wear is therefore reliably prevented.

2. A breach of the cladding for any postulated reason does not result in serious consequences. The absorber material silver-indium-cadium is relatively inert and would still remain remote from high coolant velocity regions. Rapid loss of material resulting in significant loss of reactivity control material would not occur.

3. The individually clad absorber rods are doubly secured to the retaining spider vane by a threaded joint and a welded lock pin. No failure of this joint has ever been experienced in functional testing or in years of actual service in operating plants such as San Onofre, Connecticut Yankee, Zorita, Beznau No. 1, Robert Emmett Ginna, etc.

   It should also be noted that in several instances of control rod jamming caused by foreign particles, the individual rods at the site of the jam have borne the full capacity of the control rod drive mechanism and higher impact loads to dislodge the jam without failure. The guide tube card/thimble arrangement is such that large loads are required to buckle
3 individual control rods. The conclusion to be drawn from this experience is that this joint is extremely insensitive to potential mechanical damage. A failure of the joint would result in the insertion of the individual rod into the core. This results in reduced reactivity which is a fail safe condition.

4. The spider finger braze joint by wich the individual rods are fastened to the vanes has also experienced the service described above and been subjected to the same jam freeing procedures also without failure. A failure of this joint would also result in insertion of the individual rod into the core.

5. The radial vanes are attached to the spider body, again by a brazed joint. The joints are designed to a theoretical strength in excess of that of the components joined.

   It is a feature of the design that the guidance of the rod cluster control is accomplished by the inner fingers of these vanes. They are therefore the most susceptible to mechanical damage. Since these vanes carry two rods, failure of the vane-to-hub joint such as the isolated incidents at Connecticut-Yankee does not prevent the free insertion of the rod pair (Reference 3). Neigher does such a failure interfere with the continuous free operation of the drive line, also as experienced at Connecticut-Yankee (Reference3).

Failure of the vane-to-hub joint of a single rod vane could potentially result in failure of the separated vane and rod to insert. This could occur only at withdrawal elevations where the spider is above the continuous guidance section of the guide tube (in the upper internals). A rotation of the disconnected vane could cause it to hang on one of the guide cards in the intermediate guide tube. Such an occurrence would be evident from the failure of the rod cluster control to insert below a certain elevation but with free motion above this point.

This possibility is considered extremely remote because the single rod vanes are subjected to only vertical loads and very light lateral reactions from the rods. The lateral loads are light even during a seismic event because the guide tube/guide thimble arrangement allows very limited lateral motion. The consequences of such a failure are not considered critical since only one drive line of the reactivity control system would be involved. This condition is readily observed and can be cleared at shutdown. Furthermore, this has never occurred.

6. The spider hub being of single unit cylindrical construction is very rugged and of extremely low potential for damage. It is difficult to postulate any condition to cause failure. Should some unforeseen event cause fracture of the hub above the vanes, the lower portion with the vanes and rods attached would insert by gravity into the core causing reactivity decrease. The rod could then not be removed by the drive line, again a fail safe condition. Fracture below the vanes cannot be postulated since all loads, including scram impact, are taken above the vane elevation.

7. The rod cluster control rods are provided a clear channel for insertion by the guide thimbles of the fuel assemblies. All fuel rod failures are protected against by providing this physical barrier between the fuel rod and the intended insertion channel. Distortion of the fuel rods by bending cannot apply sufficient force to damage or significantly distort the guide thimble. Fuel rod distortion by swelling, though precluded by design, would be terminated by fracture before contact with the guide thimble occurs. If such were not the case, it would be expected that a force reaction at the point of contact would cause a slight deflection of the guide thimble. The radius of curvature of the deflected shape of the guide thimbles would be sufficiently large to have a negligible influence on rod cluster control insertion.

Burnable Poison Assemblies

The burnable poison assemblies are static temporary reactivity control elements. The axial position is assured by the hold down assembly which bears against the upper core plate. Their lateral position is maintained by the guide thimbles of the fuel assemblies.

The individual rods are shouldered against the underside of the retainer plate and securely fastened at the top by a threaded nut which is then locked in place by a welded pin. The square dimension of the retainer plate is larger than the diameter of the flow holes through the core plate. Failure of the hold down bar or spring pack therefore does not result in ejection of the burnable poison rods from the core.

The only incident that couldpotentially result in ejection of the burnable poison rods is a multiple fracture of the reta.ner plate. This is not considered credible because of the ight loads borne by this component. During normal operation the loads borne by the plate are approximately 5 lb/rod or a total of 100 lb. distributed at the points of attachment. Even a multiple fracture of the retainer plate would result in jamming of the plate segments agianst the upper core plate, again preventing ejection. Excessive reactivity increase due to burnable poison ejection is therefore prevented.

The same type of stainless steel clad used on rod cluster controls is also used on the burnable poison rods. In this application there is even less susceptibility to mechanical damage since these are static assemblies. The guide thimbles of the fuel assembly afford the same protection from damage due to fuel rod failures as that described for the rod cluster control rods.

3 The consequences of clad breach are also similarly small. The poison material is borosilicate glass which is maintained in position by a central hollow tube. In the event of a hole developing in the clad for any postulated reason the expected consequence is only the loss of the helium produced by the abosrption process into the primary coolant. The glass is chemically inert and remains remote from high coolant velocities, therefore significant loss of poison material resulting in reactivity increase is not expected.

Rods of this de e performed very well in actual service with no failures observed throu .ife of one fuel cycle.

Drive Rod Assemblies

All postulated failures of the drive rod assemblies either by fracture or uncoupling lead to the fail safe condition. If the drive rod assembly fractures at any elevation, that portion remaining coupled falls with, and is guided by the rod cluster control assembly. This always results in reactivity decrease for full length control rods.

35

4.2.3.3.2 Control Rod Drive Mechanism

Material Selection

All pressure-containing materials comply with Section III of the ASME pressure vessel code, and with the exception of the needle vent valve, will be fabricated from austenitic (304) stainless steel or CF-8 stainless steel. The vent valve is a modified austenitic stainless steel cap screw.

3 Magnetic pole pieces are fabricated from 410 stainless steel. All non magnetic parts, except pins and springs, are fabricated from 304 stainless steel. Haynes 25 is used to fabricate link pins. Springs are made from Inconel-X. Latch arm tips are clad with Stellite 6 to provide improved wearability. Hard chrome plate and Stellite 6 are used selectively for bearing and wear surfaces.

4.3.1.1 Fuel Burnup

Basis

The fuel rod design basis is described in Section 4.2. The nuclear design basis is to install sufficient reactivity in the fuel to attain a region discharge burnup of 33,000 MWD/MTU. The above along with the Design Basis 4.3.1.3, Control of Power Distribution, satisfies GDC 10.

Discussion

Fuel burnup is a measure of fuel depletion which represents the integrated
3 energy output of the fuel (MWD/MTU) and is a convenient means for quantifying fuel exposure criteria.

The core design lifetime or design discharge burn-up is achieved by installing sufficient initial excess reactivity in each fuel region and by following a fuel replacement program (such as that described in Subsection 4.3.2) that meets all safety related criteria in each cycle of operation.

Initial excess reactivity installed in the fuel, although not a design basis, must be sufficient to maintain core criticality at full power operating conditions throughout cycle life with equilibrium xenon, samarium, and other fission products present. The end of design cycle life is defined to occur when the chemical shim concentration is essentially zero with control rods present to the degree necessary for operational requirements (e.g. the controlling band at the
35 "bite" position). In terms of chemical shim boron concentration this represents approximately 10 ppm with no control rod insertion.

A limitation on initial installed excess reactivity is not required other than as is quantified in terms of other design bases such as core negative reactivity feedback and shutdown margin discussed below.

4.3.1.2 Negative Reactivity Feedbacks (Reactivity Coefficient)

Basis

The fuel temperature coefficient will be negative and the moderator temperature
3 coefficient of reactivity will be non-positive for power operating conditions, thereby providing negative reactivity feedback characteristics. The design basis meets GDC 11.

Discussion

When compensation for a rapid increase in reactivity is considered, there are two major effects. These are the resonance absorption effects (Doppler) associated with changing fuel temperature and the spectrum effect resulting from changing moderator density. These basic physics characteristics are often identified by reactivity coefficients. The use of slightly enriched uranium insures that the Doppler coefficient of reactivity is negative. This coefficient provides the most rapid reactivity compensation. The core is also designed to have an overall negative moderator temperature coefficient of reactivity so that average coolant temperature or void content provides another, slower compensatory effect. Nominal power operation is permitted only in a range of overall negative moderator temperature coefficient. The negative moderator temperature coefficient can be achieved through use of fixed burnable poison and/or control rods by limiting the reactivity held down by soluble boron.

Burnable poison content (quantity and distribution) is not stated as a design basis other than as it relates to accomplishment of a non-positive moderator temperature coefficient at power operating conditions discussed above.

4.3.1.3 Control of Power Distribution

Basis

The nuclear design basis is that, with at least a 95% confidence level:

1. The fuel will not be operated at greater than 12.9 KW/ft under normal operating conditions including an allowance of 2% for calorimetric error and including densification effects.

2. Under abnormal conditions including the maximum overpower condition, the fuel peak power will not cause melting as defined in Subdivision 4.4.1.2.

3. The fuel will not operate with a power distribution that violates the departure from nucleate boiling (DNB) design basis (i.e., the DNBR shall not be less than 1.30, as discussed in Subsection 4.4.1) under Condition I and II events including the maximum overpower condition.

4. Fuel management will be such as to produce rod powers and burnups consistent with the assumptions in the fuel rod mechanical integrity analysis of Section 4.2.

The above basis meets GDC-10.

Discussion

Calculation of extreme power shapes which affect fuel design limits is performed with proven methods and verified frequently with measurements from operating reactors. The conditions under which limiting power shapes are assumed to occur are chosen conservatively with regard to any permissible operating state.

Even though there is good agreement between measured peak power calculations and measurements, a nuclear uncertainty margin is applied to calculated peak local power. Such a margin is provided both for the analysis of normal operating states and for anticipated transients.

are readily observable and alarmed, using the excore long ion chambers. Indications are also continuously available from incore thermocouples and
3 loop temperature measurements. Moveable incore detectors can be activated to provide more detailed information. In all presently proposed cores these horizontal plane oscillations are self-damping by virtue of reactivity feedback effects designed into the core.

However, axial xenon spatial power oscillations may occur late in core life.
35 The control bank and ex-core detectors are provided for control and monitoring of axial power distributions. Assurance that fuel design limits are not exceeded is provided by reactor overpower ΔT and overtemperature ΔT trip functions which use the measured axial power imbalance as an input.

4.3.1.7 Anticipated Transients Without Trip

The effects of anticipated transients with failure to trip are not considered in the Design Bases of the plant. Analysis has shown that the likelihood of such a hypothetical event is negligibly small.[1] Furthermore, analysis of the consequences of a hypothetical failure to trip following anticipated transients will be performed to show that no significant core damage would result and system peak pressures would be limited to acceptable values and no failure of the Reactor Coolant System would result. These analyses were documented in September, 1974 in accordance with the AEC policy outlined in WASH 1270 "Technical Report on Anticipated Transients Without Scram for Water-Cooled Power Reactors," September, 1973.

4.3.2 DESCRIPTION

3 4.3.2.1 Nuclear Design Description

The reactor core consists of a specified number of fuel rods which are held in bundles by spacer grids and top and bottom fittings. The fuel rods are constructed of Zircaloy cylindrical tubes containing UO_2 fuel pellets. The bundles, known as fuel assemblies, are arranged in a pattern which approximates a right circular cylinder.

Each fuel assembly contains a 17 x 17 rod array composed of 264 fuel rods, 24 rod cluster control (RCC) thimbles and an in-core instrumentation thimble. Figure 4.2.1-1 shows a cross sectional view of a 17 x 17 fuel assembly and the related RCC locations. Further details of the fuel assembly are given in Subsection 4.2.1.

The fuel rods within a given assembly have the same uranium enrichment in both the radial and axial planes. Fuel assemblies of three different enrichments are used in the initial core loading to establish a favorable radial power distribution. Figure 4.3.2-1 shows the fuel loading pattern to be used in the first core. Two regions consisting of the two lower enrichments are interspersed so as to form a checkerboard pattern in the central portion of the core. The third region is arranged around the periphery of the core and contains the highest enrichment. The reference reloading pattern is placement of new fuel on the core periphery, with depleted fuel moved inward. The core will normally operate approximately one year between refueling, accumulating approximately 11,000 MWD/MTU per year. The enrichments for the first core are shown in Table 4.3.2-1.

The core average enrichment is determined by the amount of fissionable material required to provide the desired core lifetime and energy requirements, namely a region average discharge burnup of 33,000 MWD/MTU. The physics of the burnout process is such that operation of the reactor depletes the amount of fuel available due to the absorption of neutrons by the U-235 atoms and their subsequent fission. The rate of U-235 depletion is directly proportional to the power level at which the reactor is operated. In addition, the fission process results in the formation of fission products, some of which readily absorb neutrons. These effects, depletion and the buildup of fission products, are partially offset by the buildup of plutonium, shown in Figure 4.3.2-2 for the 17 x 17 fuel assembly, which occurs due to the non-fission absorption of neutrons in U-238. Therefore, at the beginning of any cycle a reactivity reserve equal to the depletion of the fissionable fuel and the buildup of fission product poisons over the specified cycle life must be "built" into the reactor. This excess reactivity is controlled by removable neutron absorbing material in the form of boron dissolved in the primary coolant and burnable poison rods.

The concentration of boric acid in the primary coolant is varied to provide control and to compensate for long-term reactivity requirements. The concentration of the soluble neutron absorber is varied to compensate for reactivity changes due to fuel burnup, fission product poisoning including xenon and samarium, burnable poison depletion, and the cold-to-operating moderator temperature change. Using its normal makeup path, the Chemical and Volume Control System (CVCS) is capable of inserting negative reactivity at a rate of approximately 30 pcm/min when the reactor coolant boron concentration is 1000 ppm and approximately 35 pcm/min when the reactor coolant boron concentration is 100 ppm. If the emergency boration path is used, the CVCS is capable of inserting negative reactivity at a rate of approximately 65 pcm/min when the reactor coolant concentration is 1000 ppm and approximately 75 pcm/min when the reactor coolant boron concentration is 100 ppm. The peak burnout rate for xenon is 25 pcm/min (Subdivision 9.3.4.3 discusses the capability of the CVCS to counteract xenon decay). Rapid transient reactivity requirements and safety shutdown requirements are met with control rods.

As the boron concentration is increased, the moderator temperature coefficient becomes less negative. The use of a soluble poison alone would result in a positive moderator coefficient at BOL for the first cycle. Therefore, burnable poison rods are used in the first core to reduce the soluble boron concentration sufficiently to insure that the moderator temperature coefficient is negative for power operating conditions. During operation the poison content in these rods is depleted thus adding positive reactivity to offset some of the negative reactivity from fuel depletion and fission product buildup. The depletion rate of the burnable poison rods is not critical since chemical shim is always available and flexible enough to cover any possible deviations in the expected burnable poison depletion rate. Figure 4.3.2-3 is a graph of a typical core depletion with and without burnable poison rods. Note that even at end-of-life conditions some residual poison remains in the burnable poison rods resulting in a net decrease in the first cycle lifetime. Upon completion of the first cycle all the burnable poison rods are normally removed because the moderator temperature coefficient in reload cores is sufficiently negative.

To include the allowances made for densification effects, which are height dependent, the following quantities are defined.

$S(Z)$ = the allowance made for densification effects at height Z in the core. See Subdivision 4.3.2.2.5.

$P(Z)$ = ratio of the power per unit core height in the horizontal plane at height Z to the average value of power per unit core height.

3 Then

$$F_Q = \text{Total peaking factor}$$
$$= \frac{\text{Maximum kW/ft}}{\text{Average kW/ft}}$$

Including densification allowance

$$F_Q = \max F^N_{XY}(Z) \times P(Z) \times S(Z) \times F^N_U \times F^E_Q$$

4.3.2.2.2 Radial Power Distributions

The power shape in horizontal sections of the core at full power is a function of the fuel and burnable poison loading patterns and the presence or absence of a single bank of full length control rods. Thus, at any time
35 in the cycle any horizontal section of the core can be characterized as either unrodded or having Group D control rods present. These two situations combined with burnup effects determine the radial power shapes which can exist in the core at full power. The effect on radial power shapes of power level, xenon, samarium and moderator density effects are considered also but these are quite small. The effect of non-uniform flow distribution is
3 negligible. While radial power distributions in various planes of the core are often illustrated, the core radial enthalpy rise distribution as determined by the integral of power up each channel is of greater interest. Figures 4.3.2-6 thru 4.3.2-11 show representative radial power distributions for one eighth of the core for representative operating conditions. These conditions are (1) Hot Full Power (HFP) at Beginning of Life (BOL) - unrodded - no xenon, (2) HFP at BOL - unrodded - equilibrium xenon, (3) HFP at BOL -
35 Bank D in - equilibrium xenon, (4) HFP at Middle of Life - unrodded - equilibrium xenon, and (5) HFP at End of Life - unrodded - equilibrium xenon.

Since the position of the hot channel varies from time to time a single reference radial design power distribution is selected for DNB calculations.
3 This reference power distribution is chosen conservatively to concentrate power in one area of the core, minimizing the benefits of flow redistribution. Assembly powers are normalized to core average power.

4.3.2.2.3 Assembly Power Distributions

For the purpose of illustration, assembly power distributions from the
BOL and EOL conditions corresponding to Figures 4.3.2-7 and 4.3.2-10,
respectively, are given for the same assembly in Figures 4.3.2-12 and
3 4.3.2-13, respectively.

Since the detailed power distribution surrounding the hot channel varies from time to time, a conservatively flat assembly power distribution is assumed in the DNB analysis, described in Section 4.4, with the rod of maximum integrated power artifically raised to the design value of $F^N_{\Delta H}$. Care is taken in the nuclear design of all fuel cycles and all operating conditions to ensure that a flatter assembly power distribution does not occur with limiting values of $F^N_{\Delta H}$.

4.3.2.2.4 Axial Power Distributions

The shape of the power profile in the axial or vertical direction is largely
under the control of the operator either through the manual operation of the full
35 length control rods or automatic motion of full length rods responding to
manual operation of the Chemical and Volume Control System. Nuclear effects
which cause variations in the axial power shape include moderator density,
Doppler effect on resonance absorption, spatial xenon and burnup. Automatically
controlled variations in total power output and full length rod motion are
also important in determining the axial power shape at any time. Signals
are available to the operator from the excore ion chambers which are long ion
chambers outside the reactor vessel running parallel to the axis of the core.
3 Separate signals are taken from the top and bottom halves of the chambers.
The difference between top and bottom signals from each of four pairs of
detectors is displayed on the control panel and called the Flux Difference, ΔI.
Calculations of core average peaking factor for many plants and measurements
from operating plants under many operating situations are associated with either
ΔI or axial offset in such a way that an upper bound can be placed on the peaking
factor. For these correlations axial offset is defined as

$$\text{axial offset} = \frac{\phi_t - \phi_b}{\phi_t + \phi_b}$$

and ϕ_t and ϕ_b are the top and bottom detector readings.

Representative axial power shapes for BOL, MOL, and EOL conditions are shown
35 in Figures 4.3.2-14 through 4.3.2-17. These figures cover a wide range of
axial offset including values not permitted at full power.

The radial power distributions shown in Figures 4.3.2-8 and 4.3.2-9 involving the
partial insertion of control rods represent a synthesis of power shapes from
3 the rodded and unrodded planes. The applicability of the separability assumption
upon which this procedure is based is assured through extensive three-dimensional

control and load following procedures (Reference 6). The following paragraphs summarize these reports and describe the calculations used to establish the upper bound on peaking factors.

The calculations used to establish the upper bound on peaking factors, F_Q and $F_{\Delta H}$, include all of the nuclear effects which influence the radial and/or axial power distributions throughout core life for various modes of operation including load follow, reduced power operation, and axial xenon transients.

Radial power distributions are calculated for the full power condition and fuel and moderator temperature feedback effects are included for the average enthalpy plane of the reactor. The steady state nuclear design calculations are done for normal flow with the same mass flow in each channel and flow redistribution effects neglected. The effect of flow redistribution is calculated explicitly where it is important in the DNB analysis of accidents. The effect of xenon on radial power
3 distribution is small (compare Figures 4.3.2-6 and 4.3.2-7) but is included as part of the normal design process. Radial power distributions are relatively fixed and easily bounded with upper limits.

The core average axial profile, however, can experience significant changes which can occur rapidly as a result of rod motion and load changes and more slowly due to xenon distribution. For the study of points of closest approach to axial power distribution limits, several thousand cases are examined. Since the properties of the nuclear design dictate what axial shapes can occur, boundaries on the limits of interest can be set in terms of the parameters which are readily observed on the unit. Specifically, the nuclear design parameters which are significant to the axial power distribution analysis are:

(a) core power level
(b) core height
(c) coolant temperature and flow
(d) coolant temperature program as a function of reactor power
(e) fuel cycle lifetimes
(f) rod bank worths
(g) rod bank overlaps

35

Normal operation of the unit assumes compliance with the following conditions:

3 (1) Control rods in a single bank move together with no individual rod insertion differing by more than 13 steps (indicated) from the bank demand position;

(2) Control banks are sequenced with overlapping banks;

35 (3) The control bank insertion limits are not violated;

(4) Axial power distribution procedures, which are given in terms of flux difference control and control bank position, are observed.

3 The axial power distribution procedures referred to above are part of the required operating procedures which are followed in normal operation. Briefly they require control of the axial offset (flux difference % fractional power) at all power levels within a permissible operating band of a target value corresponding to the equilibrium full power value. In the first cycle, the target value changes from about -10% to 0% linearly through the life of the cycle. This minimizes xenon transient effects on the axial power distribution, since the procedures essentially keep the xenon distribution in phase with the power distribution.

35

Calculations are performed for normal operation of the reactor including load following maneuvers. Beginning, middle and end of cycle conditions
3 are included in the calculations. Different histories of operation are assumed prior to calculating the effect of load follow transients on the axial power distribution. These different histories assume base loaded
35 operation and extensive load following. The calculated points have been synthesized from axial calculations combined with radial factors appropriate for rodded and unrodded planes in the first cycle. The calculated values have been increased by a factor of 1.05 for conservatism and a factor of 1.03 for the engineering factor, F_Q^E.

The envelope drawn over the calculated (F_Q · Power) points in Figure 4.3.2-21 represents these results as an upper bound envelope on local power density
3 versus elevation in the core. It should be emphasized that this envelope is a conservative representation of the bounding values of local power density. Expected values are considerably smaller and, in fact, less conservative bounding values may be justified with additional analysis or surveillance requirements. For example, Figure 4.3.2-21 bounds both BOL and EOL conditions but without consideration of radial power distribution flattening with burnup, i.e. both BOL and EOL points presume the same radial peaking factor. Inclusion of the burnup flattening effect would reduce the local power densities corresponding to EOL conditions which may be limiting at the higher core elevations.

35

Finally, as previously discussed, this upper bound envelope is based on
3 procedures of load follow which require the operator to operate within
an allowed deviation from a target equilibrium value of axial flux
35 difference, observing certain Bank D insertion limits. These procedures
are detailed in Technical Specifications and are predicated only upon
excore surveillance supplemented by the normal monthly full core map
requirement and by computer based alarms on deviation and time of deviation
from the allowed flux difference band.

Allowing for fuel densification effects the average kw/ft at 3411 MWt
is 5.44 kw/ft. From Figure 4.3.2-21, the conservative upper bound value
of normalized local power density, including uncertainty allowances, is
3 2.32 corresponding to a peak local power density of 12.9 kw/ft at 102%
power.

To determine Reactor Protection System set points, with respect to power
distributions, three categories of events are considered, namely rod
control equipment malfunctions, operator errors of commission and operator
errors of omission.

The first category comprises uncontrolled rod withdrawal (with rods
35 moving in the normal bank sequence) for full length rod banks.
Also included are motions of the full length rod banks below
their insertion limits, which could be caused, for example, by uncontrolled
dilution or primary coolant cooldown. Power distributions were calculated
throughout these occurrences assuming short term corrective action, that
is no transient xenon effects were considered to result from the malfunction.
The event was assumed to occur from typical normal operating situations
which did include normal operating situations which did include normal
xenon transients. It was further assumed in determining the power distributions that total power level would be limited by reactor trip to
3 below 118%. Since the study is to determine protection limits with respect
to power and axial offset, no credit was taken for trip set point reduction
due to flux difference. Results are given in Figure 4.3.2-22 in units of
kw/ft. The peak power density which can occur in such events, assuming
reactor trip at or below 118%, is thus limited to 18.0 kw/ft including uncertainties and densification effects. The second category, also appearing
in Figure 4.3.2-22, assumes that the operator mis-positions the full
35 length rod bank in violation of the insertion limits and creates short
term conditions not included in normal operating conditions.

The third category assumes that the operator fails to take action to correct
a flux difference violation. The results shown on Figure 4.3.2-24 are F_Q
multiplied by 102% power including an allowance for calorimetric error.
3 The figure shows that provided the assumed error in operation does not
continue for a period which is long compared to the xenon time constant,
the maximum local power does not exceed 22.8 kW/ft including the above
factors. However, the technical specifications restrict ΔI at 102% power
such that the peak linear power density is less than 18 kW/ft. These events
are considered Condition II events.

It should be noted that a reactor overpower accident is not assumed to occur coincident with an independent operator error.

Analyses of possible operating power shapes for the reactor described herein show that the appropriate hot channel factors F_Q and $F^N_{\Delta H}$ for peak local power density and for DNB analysis at full power are the values given in Table 4.3.2-2 and addressed in Technical Specifications.

F_Q can be increased with decreasing power as shown in the Technical Specifications. Increasing $F^N_{\Delta H}$ with decreasing power is permitted by the DNB protection setpoints and allows radial power shape changes with rod insertion to the insertion limits as described in Subdivision 4.4.3.2. It has been determined that provided the above conditions 1 through 4 are observed, the Technical Specification limits are met.

When a situation is possible in normal operation which could result in local power densities in excess of those assumed as the pre-condition for a subsequent hypothetical accident, but which would not itself cause fuel failure, administrative controls and alarms are provided for returning the core to a safe condition. These alarms are described in detail in Chapters 7 and 16.

4.3.2.2.7 Experimental Verification of Power Distribution Analysis

This subject is discussed in depth in Reference 2. A summary of this report is given here.

In a measurement of peak local power density, F_Q, with the movable detector system described in Subsections 7.7.1 and 4.4.5, the following uncertainties have to be considered:

(a) reproducibility of the measured signal

(b) errors in the calculated relationship between detector current and local flux

(c) errors in the calculated relationship between detector flux and peak rod power some distance from the measurement thimble.

The appropriate allowance for (a) above has been quantified by repetitive measurements made with several inter-calibrated detectors by using the common thimble features of the incore detector system. This system allows more than one detector to access any thimble. Errors in category (b) above are quantified to the extent possible, by using the fluxes measures at one thimble location to predict fluxes at another location which is also measured. Local power distribution predictions are verified in critical experiments on arrays of rods with simulated guide thimbles, control rods, burnable poisons, etc. These critical experiments provide quantification of errors of types (b) and (c) above.

Reference 2 describes critical experiments performed at the Westinghouse Reactor Evaluation Center and measurement taken on two Westinghouse units terms of reactivity change per percent power change. The power coefficient at BOL and EOL conditions is given in Figure 4.3.2-35.

It becomes more negative with burnup reflecting the combined effect of moderator and fuel temperature coefficients with burnup. The power defect (integral reactivity effect) at BOL and EOL is given in Figure 4.3.2-36.

4.3.2.3.4 Comparison of Calculated and Experimental Reactivity Coefficients

Subsection 4.3.3 describes the comparison of calculated and experimental reactivity coefficients in detail. Based on the data presented there, the accuracy of the current analytical model is:

- $\pm$ .2% $\Delta\rho$ for Doppler and power defect
- $\pm$ 2 pcm/°F for the moderator coefficient

Experimental evaluation of the calculated coefficients will be done during the physics startup tests described in Chapter 14.

4.3.2.3.5 Reactivity Coefficients Used in Transient Analysis

Table 4.3.2-2 gives the representative ranges for the reactivity coefficients used in transient analysis. The exact values of the coefficient used in the analysis depend on whether the transient of interest is examined at the beginning or end of life, whether most negative or the most positive (least negative) coefficients are appropriate, and whether spatial nonuniformity must be considered in the analysis. Conservative values of coefficients, considering various aspects of analysis are used in the transient analysis. This is completely described in Chapter 15.

The values listed in Table 4.3.2-2 and illustrated in Figures 4.3.2-28 thru 4.3.2-36 apply to the core described in Table 4.3.2-1. The coefficients appropriate for use in subsequent cycles depends on the core's operating history, the number and enrichment of fresh fuel assemblies, the loading pattern of burned and fresh fuel, the number and location of any burnable poison rods, etc. The need for a reevaluation of any accident in a subsequent cycle is contingent upon whether or not the coefficients for that cycle fall within the identified range used in the analysis presented in Chapter 15. Control rod requirements are given in Table 4.3.2-3 for the core described and for a hypothetical equilibrium cycle since these are markedly different. These latter numbers are provided for information only and their validity in a particular cycle would be an unexpected coincidence.

4.3.2.4 Control Requirements

To insure the shutdown margin stated in the Technical Specifications under conditions where a cooldown to ambient temperature is required, concentrated soluble boron is added to the coolant. Boron concentrations for several core conditions are listed in Table 4.3.2-2. For all core conditions including refueling, the boron concentration is well below the solubility limit. The rod cluster control assemblies are employed to bring the reactor to the hot standby condition. The minimum required shutdown margin is given in Technical Specifications.

The ability to accomplish the shutdown for hot conditions is demonstrated in Table 4.3.2-3 by comparing the difference between the rod cluster control assembly reactivity available with an allowance for the worst stuck rod with that required for control and protection purposes. The shutdown margin includes an allowance of 10 percent for analytic uncertainties (see Subdivision 4.3.2.4.9). The largest reactivity control requirement appears at the end-of-life (EOL) when the moderator temperature coefficient reaches its peak negative value as reflected in the larger power defect.

The control rods are required to provide sufficient reactivity to account for the power defect from full power to zero power and to provide the required shutdown margin. The reactivity addition resulting from power reduction consists of contributions from Doppler, variable average moderator temperature, flux redistribution, and reduction in void content as discussed below.

4.3.2.4.1 Doppler

The Doppler effect arises from the broadening of U-238 and Pu-240 resonance peaks with an increase in effective pellet temperature. This effect is most noticeable over the range of zero power to full power due to the large pellet temperature increase with power generation.

3 4.3.2.4.2 Variable Average Moderator Temperature

When the core is shutdown to the hot, zero power condition, the average moderator temperature changes from the equilibrium full load value determined by the steam generator and turbine characteristics (steam pressure, heat transfer, tube fouling, etc.) to the equilibrium no load value, which is based on the steam generator shell side design pressure. The design change in temperature is conservatively increased by 4°F to account for the control dead band and measurement errors.

Since the moderator coefficient is negative, there is a reactivity addition with power reduction. The moderator coefficient becomes more negative as the fuel depletes because the boron concentration is reduced. This effect is the major contributor to the increased requirement at end of life.

4.3.2.4.3 Redistribution

During full power operation the coolant density decreases with core height, and this, together with partial insertion of control rods, results in less fuel depletion near the top of the core. Under steady state conditions, the relative power distribution will be slightly asymmetric towards the bottom of the core. On the other hand, at hot zero power conditions, the coolant density is uniform up the core, and there is no flattening due to Doppler. The result will be a flux distribution which at zero power can be skewed toward the top of the core. The reactivity insertion due to the skewed distribution is calculated with an
35 allowance for the most adverse effects of xenon distribution.

4.3.2.4.4 Void Content

A small void content in the core is due to nucleate boiling at full power. The void collapse coincident with power reduction makes a small reactivity contribution.

4.3.2.4.5 Rod Insertion Allowance

At full power, the control bank is operated within a prescribed band of travel to compensate for small periodic changes in boron concentration, changes in temperature and very small changes in the xenon concentration not compensated for by a change in boron concentration. When the control bank reaches either limit of this band, a change in boron concentration is required to compensate for additional reactivity changes. Since the insertion limit is set by a rod travel limit, a conservatively high calculation of the inserted worth is made which exceeds the normally inserted reactivity.

4.3.2.4.6 Burnup

Excess reactivity of 10% $\Delta\rho$ (hot) is installed at the beginning of each cycle to provide sufficient reactivity to compensate for fuel depletion and fission products throughout the cycle. This reactivity is controlled by the addition of soluble boron to the coolant and by burnable poison. The soluble boron concentration for several core configurations, the unit boron worth, and burnable poison worth are given in Tables 4.3.2-1 and 4.3.2-2. Since the excess reactivity for burnup is controlled by soluble boron and/or burnable poison, it is not included in control rod requirements.

4.3.2.4.7 Xenon and Samarium Poisoning

Changes in xenon and samarium concentrations in the core occur at a sufficiently slow rate, even following rapid power level changes, that the resulting reactivity change is controlled by changing the soluble boron concentration.

4.3.2.4.8 pH Effects

Changes in reactivity due to a change in coolant pH, if any, are sufficiently small in magnitude and occur slowly enough to be controlled by the boron system. Further details are available in Reference 10.

4.3.2.4.9 Experimental Confirmation

Following a normal shutdown, the total core reactivity change during cooldown with a stuck rod has been measured on a 121 assembly, 10 ft. high core and 121 assembly, 12 ft. high core. In each case, the core was allowed to cooldown until it reaches criticality simulating the steamline break accident. For the ten foot core, the total reactivity change associated with the cooldown is overpredicted by about 0.3% $\Delta\rho$ with respect to the measured result. This represents an error of about 5% in the total reactivity change and is about half the uncertainty allowance for this quantity. For the 12 foot core, the difference between the measured and predicted reactivity change was an even smaller 0.2% $\Delta\rho$. These measurements and others demonstrate the ability of the methods described in Subsection 4.3.3 to accurately predict the total shutdown reactivity of the core.

4.3.2.5 Control

Core reactivity is controlled by means of a chemical poison dissolved in the coolant, rod cluster control assemblies, and burnable poison rods as described below.

4.3.2.5.1 Chemical Poison

Boron in solution as boric acid is used to control relatively slow reactivity
3 changes associated with:

1. The moderator temperature defect in going from cold shutdown at ambient temperature to the hot operating temperature at zero power,

2. The transient xenon and samarium poisoning, such as that following power changes or changes in rod cluster control position,

3. The excess reactivity required to compensate for the effects of fissile inventory depletion and buildup of long-life fission products,

4. The burnable poison depletion.

The boron concentrations for various core conditions are presented in Table 4.3.2-2.

4.3.2.5.2 Rod Cluster Control Assemblies

Fifty-three full length rod cluster control assemblies are employed.
35 These are used for shutdown and control purposes to offset fast reactivity changes associated with:

1. The required shutdown margin in the hot zero power, stuck rods condition,

2. The reactivity compensation as a result of an increase in power above hot zero power (power defect including Doppler, and moderator reactivity changes),

3 3. Unprogrammed fluctuations in boron concentration, coolant temperature, or xenon concentration (with rods not exceeding the allowable rod insertion limits),

4. Reactivity ramp rates resulting from load changes.

The allowed full length control bank reactivity insertion is limited at full power to maintain shutdown capability. As the power level is reduced, control rod reactivity requirements are also reduced and more rod insertion is allowed. The control bank position is monitored and the operator is notified by an alarm if the limit is approached. The determination of the insertion limit uses
35 conservative xenon distributions and axial power shapes. In addition,
3 the rod cluster control assembly withdrawal pattern determined from these

analyses is used in determining power distribution factors and in determining the maximum worth of an inserted rod cluster control assembly ejection accident. For further discussion, refer to the Technical Specifications on Rod Insertion Limits.

3 Power distribution, rod ejection and rod misalignment analyses are based on the arrangement of the shutdown and control groups of the rod cluster control assemblies shown in Figure 4.3.2-37. All shutdown rod cluster control assemblies are withdrawn before withdrawal of the control banks is initiated. In going from zero to 100 percent power, control banks A, B, C and D are withdrawn sequentially. The limits of rod positions and further discussion on the basis for rod insertion limits are provided in the Technical Specifications.

35

4.3.2.5.3 Burnable Poison Rods

The burnable poison rods provide partial control of the excess reactivity available during the first fuel cycle. In doing so, these rods prevent the moderator temperature coefficient from being positive at normal operating conditions. They perform this function by reducing the requirement for
3 soluble poison in the moderator at the beginning of the first fuel cycle as described previously. The burnable poison rod pattern in the core together with the number of rods per assembly is shown in Figure 4.3.2-5, while the arrangements within an assembly are displayed in Figure 4.3.2-4. The reactivity worth of these rods is shown in Table 4.3.2-1. The boron in the rods is depleted with burnup but at a sufficiently slow rate so that the resulting critical concentration of soluble boron concentration is such that the moderator temperature coefficient remains negative at all times for power operating conditions.

4.3.2.5.4 Peak Xenon Startup

Compensation for the peak xenon buildup is accomplished using the boron control system. Startup from the peak xenon condition is accomplished with a combination of rod motion and boron dilution. The boron dilution may be made at any time, including during the shutdown period, provided the shutdown margin
3 is maintained.

4.3.2.5.5 Load Follow Control and Xenon Control

During load follow maneuvers, power changes are accomplished using control rod motion and dilution or boration by the boron system as required. Control rod motion is limited by the control rod insertion limits on full length rods as
35 provided in the Technical Specifications and discussed in Subdivision 4.3.2.5.2. The power distribution is maintained within acceptable limits through the location of the full-length rod bank. Reactivity changes due to the changing xenon concentration can be controlled by rod motion and/or changes in the soluble boron
3 concentration.

Late in cycle life, extended load follow capability is obtained by augmenting the limited boron dilution capability at low soluble boron concentrations by temporary moderator temperature reductions.

Rapid power increases (5%/min) from part power during load follow operation are accomplished with a combination of rod motion, moderator temperature reduc-
35 tion, and boron dilution. Compensation for the rapid power increase is accomplished initially by a combination of rod withdrawal and moderator temperature reduction. As the slower boron dilution takes effect after the initial rapid power increase, the moderator temperature returns to the programmed value.

4.3.2.5.6 Burnup

Control of the excess reactivity for burnup is accomplished using soluble boron and/or burnable poison. The boron concentration must be limited during operating conditions to insure the moderator temperature coefficient is negative. Sufficient burnable poison is installed at the beginning of a cycle to give the desired cycle lifetime without exceeding the boron concentration limit. The practical minimum boron concentration is 10ppm.

4.3.2.7 <u>Control Rod Pattersn and Reactivity Worth</u>

The full-length rod cluster control assemblies are designated by function as the control groups and the shutdown groups. The terms "group" and "bank" are used synonymously throughout this report to describe a particular grouping of control assemblies. The rod cluster assembly pattern is displayed in Figure
3 4.3.2-37 which is not expected to change during the life of the unit. The control banks are labeled A, B, C and D and the shutdown banks are labeled SA, SB, etc., as applicable. Each bank, although operated and controlled as a unit, is comprised of two subgroups. The axial position of the full-length rod cluster control assemblies may be controlled manually or automatically. These rod cluster control assemblies are all dropped into the core following actuation of a reactor trip signals.

Two criteria have been employed for selection of the control groups. First the total reactivity worth must be adequate to meet the requirements specified in Table 4.3.2-3. Second, in view of the fact that these rods may be partially inserted at power operation, the total power peaking factor should be low enough to ensure that the power capability requirements are met. Analyses indicate that
3 the first requirement can be met either by a single group or by two or more banks whose total worth equals at least the required amount. The axial power shape would be more peaked following movement of a single group of rods worth three to four percent $\Delta\rho$; therefore, four banks (described as A, B, C, and D in Figure 4.3.2-37) each worth approximately one percent $\Delta\rho$ have been selected.

The position of control banks for criticality under any reactor condition is determined by the concentration of boron in the coolant. On an approach to criticality boron is adjusted to ensure that criticality will be achieved with control rods above the insertion limit set by shutdown and other considerations (See Technical Specifications). Early in the cycle there may also be a withdrawal limit at low power to maintain a negative moderator temperature coefficient. Usual practice is to adjust boron to ensure that the rod position lies within the so-called maneuvering band, that is such that an escalation from zero power to full power does not require further adjustment of boron concentration.

Ejected rod worths are given in Section 15.4.6 for several different conditions. Experimental confirmation of these worths can be found by reference to start-up test reports such as Reference 11.

Allowable deviations due to misaligned control rods are discussed in Technical Specifications.

3 A representative calculation for two banks of control rods withdrawn simultaneously (rod withdrawal accident) is given in Figure 4.3.2-38.

Calculation of control rod reactivity worth versus time following reactor trip involves both control rod velocity and differential reactivity worth. The rod position versus time of travel after rod release assumed is given in Figure 4.3.2-39. For nuclear design purposes, the reactivity worth versus rod position is calculated by a series of steady state calculations at various control rod positions assuming all rods out of the core as the initial position in order to minimize the initial reactivity insertion rate. Also, to be conservative, the rod of highest worth is assumed stuck out of the core and the flux distribution (and thus reactivity importance) is assumed to be skewed to the bottom of the core. The result of these calculations is shown on Figure 4.3.2-40.

The shutdown groups provide additional negative reactivity to assure an adequate shutdown margin. Shutdown margin is defined as the amount by which the core would be subcritical at hot shutdown if all rod cluster control assemblies are tripped, but assuming that the highest worth assembly remains fully withdrawn and no changes in xenon or boron concentration take place. The loss of control
35 rod worth due to the material irradiation is negligible since only bank D may be in the core under normal operating conditions.

The values given in Table 4.3.2-3 show that the available reactivity in withdrawn rod cluster control assemblies provides the design bases minimum shutdown margin allowing for the highest worth cluster to be at its fully withdrawn position.
3 An allowance for uncertainty in the calculated worth of N-1 rods is made before determination of the shutdown margin.

4.3.2.7 Criticality of Fuel Assemblies

Criticality of fuel assemblies outside of the reactor is precluded by adequate design of fuel transfer and fuel storage facilities and by administrative control procedures. This section identifies those criteria important to criticality safety analyses.

New fuel is generally stored in fuel storage facilities with no water present but which are designed so as to prevent accidental criticality even if unborated water is present.

In the analysis for the storage facilities, the fuel assemblies are assumed to be in their most reactive condition, namely fresh or undepleted and with no control rods or removable neutron absorbers present. Assemblies can not be closer together than the design separation provided by the storage facility except in special cases such as in fuel shipping containers where analyses are carried out to establish the acceptability of the design. The mechanical integrity of the fuel assembly is assumed.

For full flooding with unborated water, the fuel assembly spacing of the facility provides essentially full nuclear isolation and K_{eff} for the array is no greater than K_{eff} for the single most reactive fuel assembly.

The design basis for wet fuel storage criticality analyses is that, considering possible variations, there is a 95% confidence level for the effective multiplication factor (K_{eff}) of the fuel storage array being less than 0.95 per ANSI standard N18.2-1973. The possible variations in the criticality analyses are in two categories: (1) calculational uncertainties, and (2) fuel rack fabrication uncertainties.

The results of comparing standard calculations with 101 critical experiments as summarized in Table 4.3.2-6 indicate that:

1. The average difference between the calculations and experimental results, or bias in the computations, was 0.1%Δk, and

2. The standard deviation in the difference between the calculations and experimental results was 0.86%Δk.

Typical fuel rack fabrication uncertainties are as follows:

1. The clearance to permit insertion of the fuel assemblies into the rack is 0.64 inch, and

2. The tolerance on the center-to-center spacing between fuel racks is 0.19 inch.

The fuel assembly (17 x 17 fuel rods) of standard design and 3.5 w/o enriched uranium oxide, without a control rod or burnable poison rods, fully flooded and reflected with cold clean water, has a k_{eff} of about 0.85. Two such fuel assemblies spaced one inch apart with parallel axes 9.5 inches apart have a k_{eff} of about 0.99. Three such fuel assemblies spaced one inch apart with parallel axes would be supercritical.

An infinite number of dry fuel assemblies of this design would have a $k_{eff} < 0.80$.

3 a. The core was stable against induced axial xenon transients both at the core average burnups of 1550 MWD/MTU and 7700 MWD/MTU. The measured stability indices are $-0.041\ hr^{-1}$ for the first test (Curve 1 of Figure 4.3.2-41) and $-0.014\ hr^{-1}$ for the second test (Curve 2 of Figure 4.3.2-41). The corresponding oscillation periods are 32.4 hrs. and 27.2 hrs., respectively.

35 b. The reactor core becomes less stable as fuel burnup progresses and the axial stability index was essentially zero at 12,000 MWD/T.

2. Measurements in the X-Y Plane

Two X-Y xenon oscillation tests were performed at a PWR unit with a core height of 12 feet and 157 fuel assemblies. This unit has the highest power output of any Westinghouse PWR currently in operation (1972). The first test was conducted at a core average burnup of 1540 MWD/MTU and the second at a core average burnup of 12900 MWD/MTU. Both of the X-Y xenon tests show that the core was stable in the X-Y plane at both burnups. The second test shows that the core became more stable as the fuel burnup increased and all Westinghouse PWR's with 121 and 157 assemblies are expected to be stable throughout their burnup cycles.

In each of the two X-Y tests, a perturbation was introduced to the equilibrium power distribution through an impulse motion of one RCC unit located along the diagonal axis. Following the perturbation, the uncontrolled oscillation was monitored using the moveable detector and thermocouple system and the excore power range detectors. The quadrant tilt difference is the quantity that properly represents the diametral oscillation in the X-Y plane of the reactor core in that the differences of the quadrant average powers over two symmetrically opposite quadrants essentially eliminates the contribution to the oscillation from the azimuthal mode. The quadrant tilt difference (QTD) data were fitted in
3 the form of Eq. (2) through a least-square method. A stability index of $-0.076\ hr^{-1}$ with a period of 29.6 hours was obtained from the thermocouple data shown in Figure 4.3.2-42.

It was observed in the second X-Y xenon test that the PWR core with 157 fuel assemblies had become more stable due to an increased fuel depletion and the stability index was not determined.

4.3.2.8.5 Comparison of Calculations with Measurements

The analysis of the axial xenon transient tests was performed in an axial slab geometry using a flux synthesis technique. The direct simulation of the AO data was carried out using the PANDA Code[16]. The analysis of the X-Y xenon transient tests was performed in an X-Y geometry using a modified TURTLE[9] code. Both the PANDA and TURTLE codes solve the two-group time-dependent neutron diffusion equation with time-dependent xenon and iodine concentrations. The fuel temperature and moderator density feed back is limited to a steady-state model. All the X-Y calculations were performed in an average enthalpy plane.

The basic nuclear cross-sections used in this study were generated from a unit cell depletion program which has evolved from the codes LEOPARD[17] and CINDER[18]. The detailed experimental data during the tests including the reactor power level, enthalpy rise and the impulse motion of the control rod assembly, as well as the plant follow burnup data were closely simulated in the study.

The results of the stability calculation for the axial tests are compared with the experimental data in Table 4.3.2-4.The calculations show conservative results for both of the axial tests with a margin of approximately -.01 hr^{-1} in the stability index.

An analytical simulation of the first X-Y xenon oscillation test shows a calculated stability index of -0.081 hr^{-1}, in good agreement with the measured value of -0.076 hr^{-1}. Ad indicated earlier, the second X-Y xenon test showed that the core had become more stable compared to the first test and no evaluation of the stability index was attempted. This increase in the core stability in the X-Y plane due to increased fuel burnup is due mainly to the increased
3 magnitude of the negative moderator temperature coefficient.

Previous studies of the physics of xenon oscillations, including three-dimensional analysis, are reported in the series of topical reports, References 12,13, and 14 A more detailed description of the experimental results and analysis of the axial and X-Y xenon transient tests is presented in References 15 and Section 1 of Reference 19.

4.3.2.8.6 Stability Control and Protection

The excore detector system is utilized to provide indications of xenon-induced spatial oscillations. The readings from the excore detectors are available to the operator and also form part of the protection system.

1. Axial Power Distribution

   For maintenance of proper axial power distributions, the operator is instructed to maintain an axial offset within a prescribed operating band, based on the excore detector readings. Should the axial offset be permitted to move far enough outside this band, the protection limit will be reached and the power will be automatically cut back.

   Full-length control rods are sufficient to shape the axial power
35 distribution and to dampen the axial xenon oscillations effectively.

2. Radial Power Distribution

   The core described herein is calculated to be stable against X-Y xenon
3 induced oscillations at all times in life.

   The X-Y stability of large PWR's will be further verified as part of the startup physics test program at a PWR core with 193 fuel assemblies. The measured X-Y stability of the PWR core with 157 assemblies and the good

4.4 THERMAL AND HYDRAULIC DESIGN

4.4.1 DESIGN BASES

The overall objective of the thermal and hydraulic design of the reactor core is to provide adequate heat transfer which is compatible with the heat generation distribution in the core such that heat removal by the Reactor Coolant System or the Emergency Core Cooling System (when applicable) assures that the following performance and safety criteria requirements are met:

1. Fuel damage* is not expected during normal operation and operational transients (Condition I) or any transient conditions arising from faults of moderate frequency (Condition II). It is not possible, however, to preclude a very small number of rod failures. These will be within the capability of the unit cleanup system and are consistent with the unit design bases.

2. The reactor can be brought to a safe state following a Condition III event with only a small fraction of fuel rods damaged* although sufficient fuel damage might occur to preclude resumption of operation without considerable outage time.

3. The reactor can be brought to a safe state and the core can be kept subcritical with acceptable heat transfer geometry following transients arising from Condition IV events.

In order to satisfy the above criteria the following design bases have been established for the thermal and hydraulic design of the reactor core.

4.4.1.1 Departure from Nucleate Boiling Design Basis

Basis

Departure from nucleate boiling will not occur on at least 95% of the limiting fuel rods during normal operation and operational transients and any transient conditions arising from faults of moderate frequency (Condition I and II events) at a 95% confidence level. Historically this has been conservatively met by limiting the minimum departure from nucleate boiling ratio (DNBR) to 1.30 and for this application a minimum DNBR of 1.30 will continue to be used.

Discussion

By preventing departure from nucleate boiling, adequate heat transfer is assured between the fuel clad and the reactor coolant, thereby preventing clad damage as a result of inadequate cooling. Maximum fuel rod surface temperature is not a design basis as it will be within a few degrees of coolant temperature during operation in the nucleate boiling region. Limits provided by the nuclear control and protection systems are such that this

*Fuel damage as used here is defined as penetration of the fission product barrier (i.e. the fuel rod clad).

design basis will be met for transients associated with Condition II events including overpower transients. There is an additional large DNBR margin at rated power operation and during normal operating transients.

4.4.1.2 Fuel Temperature Design Basis

Basis

During modes of operation associated with Condition I and Condition II events, the maximum fuel temperature shall be less than the melting temperature of UO_2. The UO_2 melting temperature for at least 95% of the peak kW/ft fuel rods will not be exceeded at the 95% confidence level. The melting temperature of UO_2 is taken as 5080°F (Reference 1) unirradiated and reducing 58°F per 10,000
3 MWD/MTU. By precluding UO_2 melting, the fuel geometry is preserved and possible adverse effects of molten UO_2 on the cladding are eliminated. To preclude center melting and as a basis for overpower protection system setpoints, a calculated centerline fuel temperature of 4700°F has been selected as the overpower limit. This provides sufficient margin for uncertainties in the thermal evaluations as described in Subdivision 4.4.2.10.1.

Discussion

Fuel rod thermal evaluations are performed at rated power, maximum overpower and during transients at various burnups. These analyses assure that this design basis as well as the fuel integrity design bases given in Section 4.2 are met. They also provide input for the evaluation of Condition III and IV faults given in Chapter 15.

4.4.1.3 Core Flow Design Basis

Basis

35 A minimum of 92.5% of the thermal flow rate will pass through the fuel rod region of the core and be effective for fuel rod cooling. Coolant flow through the thimble tubes as well as the leakage from the core barrel-baffle region into the core are not considered effective for heat removal.

3

Discussion

Core cooling evaluations are based on the thermal flow rate (minimum flow)
35 entering the reactor vessel. A maximum of 7.5% of this value is allotted as bypass flow. This includes RCC guide thimble cooling flow, head cooling flow, baffle leakage, and leakage to the vessel outlet nozzle.

3 4.4.1.4 Hydrodynamic Stability Design Bases

Basis

Modes of operation associated with Condition I and II events shall not lead to hydrodynamic instability.

Heat Flux Engineering Hot-Channel Factor, F_Q^E

The heat flux engineering hot channel factor is used to evaluate the maximum heat flux. This subfactor is determined by statistically combining the tolerances for the fuel pellet diameter, density, enrichment, eccentricity and the fuel rod diameter, and has a value of 1.03. Measured manufacturing data on recent Westinghouse 17x17 fuel were used to verify that this value was not exceeded for 95% of the limiting fuel rods at a 95% confidence level. Thus, it is expected that a statistical sampling of the fuel assemblies of this plant will yield a value no larger tha[illegible].03.

Enthalpy Rise Engineering Hot-Channel Factor, $F_{\Delta H}^E$

The effect of variations in flow conditions and fabrication tolerances on the hot-channel enthalpy rise is directly considered in the THINC core thermal subchannel analysis (See Subdivision 4.4.3.4.1) under any reactor operating condition. The items considered contributing to the enthalpy rise engineering hot-channel factor are discussed below:

1. Pellet diameter, density and enrichment and fuel rod diameter, pitch and bowing:

   Design values employed in the THINC analysis related to the above fabrication variations are based on applicable limiting tolerances such that these design values are met for 95 percent of the limiting channels at a 95% confidence level. Measured manufacturing data on Westinghouse 17x17 fuel show the tolerances used in this evaluation are conservative. In addition, each fuel assembly is checked to assure the channel spacing design criteria are met. The effect of variations in pellet diameter and enrichment is employed in the THINC analysis as a direct multiplier on the hot channel enthalpy rise while the fuel rod diameter, pitch and bowing variation including in-pile effects is considered in the preparation of the THINC input values such as axial flow area, equivalent hydraulic diameter and lateral crossflow area for the hot channel.

2. Inlet Flow Maldistribution:

   The consideration of inlet flow maldistribution in core thermal performances is discussed in Subdivision 4.4.3.1.2. A design basis of 5% reduction in coolant flow to the hot assembly is used in the THINC-IV analysis.

3. Flow Redistribution

   The flow redistribution accounts for the reduction in flow in the hot channel resulting from the high flow resistance in the channel due to the local or bulk boiling. The effect of the non-uniform power distribution is inherently considered in the THINC analysis for every operating condition which is evaluated.

4. Flow Mixing:

3 The subchannel mixing model incorporated in the THINC Code and used in reactor design is based on experimental data (Reference 51) discussed in Subdivision 4.4.3.4.1. The mixing vanes incorporated in the spacer
35 grid design induce additional flow mixing between the various flow channels in a fuel assembly as well as between adjacent assemblies. This mixing reduces the enthalpy rise in the hot channel resulting from local power peaking or unfavorable mechanical tolerances.

4.4.2.4 Flux Tilt Considerations

Significant quadrant power tilts are not anticipated during normal operation since this phenomenon is caused by some asymmetric perturbation. A dropped or misaligned RCCA could cause changes in hot channel factors; however, these events are analyzed separately in Chapter 15. This discussion will be confined to flux tilts caused by x-y xenon transients, inlet temperature mismatches, enrichment variations within tolerances and so forth.

The design value of the enthalpy rise hot channel factor $F^N_{\Delta H}$, which includes an 8% uncertainty (as discussed in Subdivision 4.3.2.2.7), is assumed to be sufficiently conservative that flux tilts up to and including the alarm point (see Subsection 16.3.12, Technical Specifications) will not result in values of $F^N_{\Delta H}$ greater than that assumed in this submittal. The design value of F_Q does not include a specific allowance for quadrant flux tilts.

3 4.4.2.5 Void Fraction Distribution

The calculated core average and the hot subchannel maximum and average void fractions are presented in Table 4.4.2-3 for operation at full power with design hot channel factors. The void fraction distribution in the core at various radial and axial locations is presented in Reference 52. The void models used in the THINC-IV computer code are described in Subdivision 4.4.2.8.3.

Since void formation due to subcooled boiling is an important promoter of interassembly flow redistribution, a sensitivity study was performed with THINC-IV using the void model referenced above (Reference 52).

The results of this study showed that because of the realistic crossflow model used in THINC-IV, the minimum DNBR in the hot channel is relatively insensitive to variations in this model. The range of variations considered in this sensitivity study covered the maximum uncertainty range of the data used to develop each part of the void fraction correlation.

4.4.2.6 Core Coolant Flow Distribution

Assembly average coolant mass velocity and enthalpy at various radial and axial core locations are given below. Coolant enthalpy rise and flow distributions are shown for the 4 foot elevation (1/3 of core height) in Figure 4.4.2-13, and 8 foot elevation (2/3 of core height) in Figure 4.4.2-14 and at the core exit in Figure 4.4.2-15. These distributions are for the full power conditions as given in Table 4.4.2-1 and for the radial power density distribution shown in Figure 4.3.2-7. The THINC code analysis for this case utilized a uniform core inlet enthalpy and inlet flow distribution.

4.4.2.7 Core Pressure Drops and Hydraulic Loads

4.4.2.7.1 Core Pressure Drops

The analytical model and experimental data used to calculate the pressure drops shown in Table 4.4.2-1 are described in Subdivision 4.4.2.8. The core pressure drop includes the fuel assembly, lower core plate, and upper core plate pressure drops. The full power operation pressure drop values shown in the Table are the unrecoverable pressure drops across the vessel, including the inlet and outlet nozzles, and across the core. These pressure drops are based on the best estimate Flow (most likely value for actual plant operating conditions) as described in Subsection 5.1.1. Subsection 5.1.1 also defines and describes
3 the thermal design flow (minimum flow) which is the basis for reactor core thermal performance and the mechanical design flow (maximum flow) which is used in the mechanical design of the reactor vessel internals and fuel assemblies. Since the best estimate flow is that flow which is most likely to exist in an operating plant, the calculated core pressure drops in Table 4.4.2 1 are based on this best estimate flow rather than the thermal design flow.

Uncertainties associated with the core pressure drop values are discussed in Subdivision 4.4.2.10.2.

The pressure drops quoted in Table 4.4.2-1 are based on seven grids and conservatively estimated grid pressure loss coefficients. Phase 1 of the D-loop tests (Reference 5) resulted in a measured core pressure drop of a magnitude sufficiently lower than the predicted pressure drop that the pressure drops quoted in Table 4.4.2-1 will be conservative even with the addition of an eighth grid. The estimated pressure drop compared to the measured pressure drop in Reference 5 uses the same conservatively estimated grid pressure loss coefficients used for the Table 4.4.2-1 pressure drop calculations. Thus, it was expected that the calculated pressure drop would be conservative (larger) relative to the measured value. The McGuire fuel assembly grids, top nozzle, and bottom nozzle designs are the same as in the prototype assembly tests and the hydraulic resistances measured during the test are therefore directly applicable to the McGuire analysis.

35

4.4.2.7.2 Hydraulic Loads

The fuel assembly hold down springs, Figure 4.2.1-2, are designed to keep the fuel assemblies resting on the lower core plate under transients associated
3 with Condition II and III events. Maximum flow conditions are limiting because hydraulic loads are a maximum. The most adverse flow conditions occur during a LOCA. These conditions are presented in Subsection 15.4.1.

Hydraulic loads at normal operating conditions are calculated based on the mechanical design flow which is described in Section 5.1 and accounting for the minimum core bypass flow based on manufacturing tolerances. Core hydraulic loads at cold unit startup conditions are also based on this flow but are adjusted to account for the coolant density difference. Conservative core hydraulic loads for a pump overspeed transient, which create flow rates 20% greater than the mechanical design flow, are evaluated to be greater than twice the fuel assembly weight.

Core hydraulic loads were measured during the prototype assembly tests described in Section 1.5. Reference 5 contains a detailed discussion of the results.

Lift forces are directly proportional to the pressure drop. The lift force on an eight grid assembly is thus less than 5 percent greater than the lift force on a seven grid assembly for the same flow rate. Reference 5 shows that lift off of an eight grid fuel assembly (5 percent greater than the seven grid assembly shown) is not predicted during a postulated pump overspeed transient even though it is not necessary to preclude lift off.

The hydraulic loads during normal operation can be obtained from Reference 5 by adjusting the loads for the McGuire pressure drop and flow rate. The effect of startup and shutdown transients are shown to be inconsequential in Reference 5.

4.4.2.8 Correlation and Physical Data

4.4.2.8.1 Surface Heat Transfer Coefficients

Forced convection heat transfer coefficients are obtained from the familiar Dittus-Boelter correlation (Reference 53), with the properties evaluated at bulk fluid conditions:

$$\frac{hD_e}{K} = 0.023 \left(\frac{D_e G}{\mu}\right)^{0.8} \left(\frac{C_p \mu}{K}\right)^{0.4} \qquad (4.4\text{-}15)$$

where:

h = heat transfer coefficient, BTU/hr-ft^2 - °F
D_e = equivalent diameter, ft
K = thermal conductivity, BTU/hr-ft - °F
G = mass velocity, lb/hr-ft^2
μ = dynamic viscosity, lb/ft-hr
C_p = heat capacity, BTU/lb - °F

This correlation has been shown to be conservative (Reference 54) for rod bundle geometries with pitch to diameter ratios in the range used by PWRs.

The onset of nucleate boiling occurs when the clad wall temperature reaches the amount of superheat predicted by Thom's (Reference 55) correlation. After this occurrence the outer clad wall temperature is determined by:

$$\Delta T_{sat} = [0.072 \exp(-P/1260)] \quad (q'')^{0.5} \qquad (4.4\text{-}16)$$

where:

ΔT_{sat} = wall superheat, $T_w - T_{sat}$, °F
q'' = wall heat flux, BTU/hr-ft^2
P = pressure, psia
T_w = outer clad wall temperature, °F
T_{sat} = saturation temperature of coolant at P, °F

4.4.2.8.2 Total Core and Vessel Pressure Drop

Unrecoverable pressure losses occur as a result of viscous drag (friction) and/or geometry changes (form) in the fluid flow path. The flow field is assumed to be incompressible, turbulent, single-phase water. These assumptions apply to the core and vessel pressure drop calculations for the purpose of establishing the primary loop flow rate. Two-phase considerations are neglected in the vessel pressure drop evaluation because the core average void is negligible (See Subdivision 4.4.2.5 and Table 4.4.2-3). Two phase flow considerations in the core thermal subchannel analyses are considered and the models are discussed in Subdivision 4.4.3.1.3. Core and vessel pressure losses are calculated by equations of the form:

$$\Delta P_L = \left(K + F \frac{L}{D_e}\right) \frac{\rho V^2}{2 g_c (144)} \qquad (4.4\text{-}17)$$

where:

ΔP_L = unrecoverable pressure drop, lb$_f$/in^2
ρ = fluid density, lbm/ft^3
L = length, ft.
D_e = equivalent diameter, ft
V = fluid velocity, ft/sec

$g_c = 32.174 \dfrac{lb_m\text{-}ft}{lb_f\text{-}sec^2}$

K = form loss coefficient, dimensionless
F = friction loss coefficient, dimensionless

Fluid density is assumed to be constant at the appropriate value for each component in the core and vessel. Because of the complex core and vessel flow geometry, precise analytical values for the form and friction loss coefficients are not available. Therefore, experimental values for these coefficients are obtained from geometrically similar models.

Values are quoted in Table 4.4.2-1 for unrecoverable pressure loss across the reactor vessel, including the inlet and outlet nozzles, and across the core.

The results of full scale tests of core components and fuel assemblies were utilized in developing the core pressure loss characteristic. The pressure drop for the vessel was obtained by combining the core loss with correlation of
3 1/7th scale model hydraulic test data on a number of vessels (References 56 and 57) and form loss relationships (Reference 58). Moody (Reference 59) curves were used to obtain the single phase friction factors.

35 Core pressure drops were confirmed by Phase 1 (Reference 5) and later phases of the D-loop tests. These hydraulic verification tests include hydraulic head losses and effects of velocity changes as well as unrecoverable pressure losses. The effects of velocity changes are small since the static pressure taps are located at elevations of approximately equal flow areas (and therefore approximately equal velocities). When wall static pressure taps are used near ambient fluid conditions, it can be shown analytically that the elevation head losses do not contribute to the measured core pressure drops. Therefore, data from the hydraulic verification tests can be directly applied to confirm the pressure drop values quoted in Table 4.4.2-1 which are based on unrecoverable pressure losses only.

Tests of the primary coolant loop flow rates will be made (See Subdivision 4.4.4.1) prior to initial criticality to verify that the flow rates used in the design, which were determined in part from the pressure losses calculated by the method described here, are conservative.

4.4.2.8.3 Void Fraction Correlation

There are three separate void regions considered in flow boiling in a PWR as illustrated in Figure 4.4.2-16. They are the wall void region (no bubble detachment), the subcooled boiling region (bubble detachment) and the bulk boiling region.

3 In the wall void region, the point where local boiling begins is determined when the clad temperature reaches the amount of superheat predicted by Thom's (Reference 55), correlation (discussed in Subdivision 4.4.2.8.1). The void fraction in this region is calculated using Maurer's (Reference 60) relationship. The bubble detachment point, where the superheated bubbles break away from the wall, is determined by using Griffith's (Reference 61) relationship.

The void fraction in the subcooled boiling region (that is, after the detachment point) is calculated from the Bowring (Reference 62) correlation. This correlation predicts the void fraction from the detachment point to the bulk boiling region.

The void fraction in the bulk boiling region is predicted by using homogeneous flow theory and assuming no slip. The void fraction in this region is therefore a function only of the thermodynamic quality.

4.4.2.9 Thermal Effects of Operational Transients

DNB core safety limits are generated as a function of coolant temperature, pressure, core power and axial power imbalance. Steady-state operation within these safety limits insures that the minimum DNBR is not less than 1.30. Refer to Chapter 15 for a discussion of DNBR limits. These limits provide adequate protection against anticipated operational transients that are slow with respect to fluid transport delays in the primary system. In addition, for fast transients, e.g., uncontrolled rod bank withdrawal at power incident (Subsection 15.2.2) specific protection functions are provided as described in Section 7.2, and the use of these protection functions are described in Chapter 15. The thermal response of the fuel rod is discussed in Subdivision 4.4.3.7.

4.4.2.10 Uncertainties in Estimates

4.4.2.10.1 Uncertainties in Fuel and Clad Temperatures

As discussed in Subdivision 4.4.2.2, the fuel temperature is a function of crud, oxide, clad, gap, and pellet conductances. Uncertainties in the fuel temperature calculation are essentially of two types: fabrication uncertainties such as variations in the pellet and clad dimensions and the pellet density; and model uncertainties such as variations in the pellet conductivity and the gap conductance. These uncertainties have been quantified by comparison of the thermal model to the inpile thermocouple measurements (References 7 through 13), by out-of-pile measurements of the fuel and clad properties (References 16 through 27), and by measurements of the fuel and clad dimensions during fabrication. The resulting uncertainties are then used in all evaluations involving the fuel temperature. The effect of densification on fuel temperature uncertainties is presented in Reference 6.

In addition to the temperature uncertainty described above, the measurement uncertainty in determining the local power, and the effect of density and enrichment variations on the local power are considered in establishing the heat flux hot channel factor. These uncertainties are described in Subdivision 4.3.2.2.1.

Reactor trip setpoints as specified in the Technical Specifications include allowance for instrument and measurement uncertainties such as calorimetric error, instrument drift and channel reproductivity, temperature measurement uncertainties, noise, and heat capacity variations.

Uncertainty in determining the cladding temperature results from uncertainties in the crud and oxide thicknesses. Because of the excellent heat transfer between the surface of the rod and the coolant, the film temperature drop does not appreciably contribute to the uncertainty.

4.4.2.10.2 Uncertainties in Pressure Drops

Core and vessel pressure drops based on the best estimate flow, as described in Section 5.1, are quoted in Table 4.4.2-1. The uncertainties quoted are based on the uncertainties in both the test results and the analytical extension of these values to the reactor application. The magnitude of the uncertainties will be confirmed when the experimental data on the prototype fuel assembly (Section 1.5) is obtained.

A major use of the core and vessel pressure drops is to determine the primary system coolant flow rates as discussed in Section 5.1. In addition, as discussed in Subdivision 4.4.4.1, tests on the primary system prior to initial criticality will be made to verify that a conservative primary system coolant flow rate has been used in the design and analyses of the plant.

4.4.2.10.3 Uncertainties Due to Inlet Flow Maldistribution

The effects of uncertainties in the inlet flow maldistribution criteria used in the core thermal analyses is discussed in Subdivision 4.4.3.1.2.

4.4.2.10.4 Uncertainty in DNB Correlation

The uncertainty in the DNB correlation (Subdivision 4.4.2.3) can be written as a statement on the probability of not being in DNB based on the statistics of the DNB data. This is discussed in Subdivision 4.4.2.3.2.

4.4.2.10.5 Uncertainties in DNBR Calculations

The uncertainties in the DNBR's calculated by THINC analysis (see Subdivision
4.4.3.4.1) due to uncertainties in the nuclear peaking factors are accounted
for by applying conservatively high values of the nuclear peaking factors and
including measurement error allowances. In addition, conservative values for the
engineering hot channel factors are used as discussed in Subdivision 4.4.2.3.4.
3 The results of a sensitivity study (Reference 52) with THINC-IV show that the
minimum DNBR in the hot channel is relatively insensitive to variations in the
core-wide radial power distribution (for the same value of $F^N_{\Delta H}$).

The ability of the THINC-IV computer code to accurately predict flow and enthalpy distributions in rod bundles is discussed in Subdivision 4.4.3.4.1 and in Reference 63. Studies have been performed (Reference 52) to determine the sensitivity of the minimum DNBR in the hot channel to the void fraction correlation (see also Subdivision 4.4.2.8.3); the inlet velocity and exit pressure distributions assumed as boundary conditions for the analysis; and the grid pressure loss coefficients. The results of these studies show that the minimum DNBR in the hot channel is relatively insensitive to variations in these parameters. The range of variations considered in these studies covered the range of possible variations in these parameters.

4.4.2.10.6 Uncertainties in Flow Rates

The uncertainties associated with loop flow rates are discussed in Section 5.1.
For core thermal performance evaluations, a thermal design loop flow is used
which is less than the best estimate loop flow (approximately 4.3%). In addition
35 another 7.5% of the thermal design flow is assumed to be ineffective for core
heat removal capability because it bypasses the core through the various available
vessel flow paths described in Subdivision 4.4.3.1.1.

4.4.2.10.7 Uncertainties in Hydraulic Loads

As discussed in Subdivision 4.4.2.7.2, hydraulic loads on the fuel assembly are
evaluated for a pump overspeed transient which create flow rates 20% greater
3 than the mechanical design flow. The mechanical design flow as stated in
Section 5.1 is greater than the best estimate or most likely flow rate value
for the actual plant operating condition (by approximately 4.5%).

4.4.2.10.8 Uncertainty in Mixing Coefficient

The value of the mixing coefficient, TDC, used in THINC analyses for this
application is 0.038. The mean value of TDC obtained in the "R" grid mixing tests
35 described in Subdivision 4.4.2.3.3 was 0.042 (for 26 inch grid spacing). The

value of 0.038 is one standard deviation below the mean value; and $\sim$90% of the data gives values of TDC greater than 0.038 (Reference 46).

The results of the mixing tests done on 17 x 17 geometry, as discussed in Subdivision 4.4.2.3.3, had a mean value of TDC of 0.059 and standard deviation of σ = 0.007. Hence the current design value of TDC is almost 3 standard deviations below the means for 26 inch grid spacing.

4.4.2.11 Plant Configuration Data

Plant configuration data for the thermal-hydraulic and fluid systems external to the core are provided in the appropriate Chapters 5,6, and 9. Implementation of the Emergency Core Cooling System (ECCS) is discussed in Chapter 15. Some specific areas of interest are the following:

1. Total coolant flow rates for the Reactor Coolant System (RCS) and each loop are provided in Table 5.1-1. Flow rates employed in the evaluation of the core are presented in Section 4.4.

2. Total RCS volume including pressurizer and surge line, RCS liquid volume including pressurizer water at steady state power conditions are given in Table 5.1-1.

3. The flow path length through each volume may be calculated from physical data provided in the above referenced tables.

4. The height of fluid in each component of the RCS may be determined from the physical data presented in Section 5.5. The components of the RCS are water filled during power operation with the pressurizer being approximately 60% water filled.

5. Components of the ECCS are to be located so as to meet the criteria for NPSH described in Section 6.3.

6. Line lengths and sizes for the safety injection system are determined so as to guarantee a total system resistance which will provide, as a minimum, the fluid delivery rates assumed in the safety analyses described in Chapter 15.

7. The minimum flow areas for components of the RCS are presented in Section 5.5, component and Subsystem Design.

8. The steady state pressure drops and temperature distributions through the RCS are presented in Table 5.1-1.

4.4.3 EVALUATION

4.4.3.1 Core Hydraulics

4.4.3.1.1 Flow Paths Considered in Core Pressure Drop and Thermal Design

The following flow paths or core bypass flow are considered:

1. Flow through the spray nozzles into the upper head for head cooling purposes.

2. Flow entering into the RCC guide thimbles to cool the control rods.

3. Leakage flow from the vessel inlet nozzle directly to the vessel outlet nozzle through the gap between the vessel and the barrel.

3
4. Flow entering into the core from the baffle-barrel region through the gaps between the baffle plates.

The above contributions are evaluated to confirm that the design value of core bypass flow is met. The design value of core bypass flow for McGuire is equal
35 to 7.5% of the total vessel flow. Of the total allowance, 5.5% is associated with the internals (Items 1, 3, and 4 above) and 2.0% for the core. Calculations
3 have been performed using drawing tolerances on a worst case basis and accounting for uncertainties in pressure losses. Based on these calculations, the core
35 bypass flow for McGuire is $\leq$ 7.5%. This design bypass value is also used in the evaluation of the core pressure drops quoted in Table 4.4.2-1, and the determination of reactor flow rates in Section 5.1.

Flow model tests results for the flow path through the reactor are discussed in Section 4.4.2.8.2.

4.4.3.1.2 Inlet Flow Distributions

Data has been considered from several 1/7 scale hydraulic reactor model tests (References 56, 57, and 64) in arriving at the core inlet flow maldistribution criteria to be used in the THINC analyses (See Subdivision 4.4.2.3.1). THINC I analyses made using this data have indicated that a conservative design basis is to consider a 5 percent reduction in the flow to the hot assembly. Reference 65. The same design basis of 5% reduction to the hot assembly inlet is used in THINC IV analyses.

The experimental error estimated in the inlet velocity distribution has been considered as outlined in Reference 52 where the sensitivity of changes in
3 inlet velocity distributions to hot channel thermal performance is shown to be small. Studies (Reference 52) made with the improved THINC model (THINC-IV) show that it is adequate to use the 5% reduction in inlet flow to the hot assembly for a loop out of service based on the experimental data in References 56 and 57.

The effect of the total flow rate on the inlet velocity distribution was studied in the experiments of Reference 56. As was expected, on the basis of the theoretical analysis, no significant variation could be found in inlet velocity distribution with reduced flow rate.

4.4.3.1.3 Empirical Friction Factor Correlations

Two empirical friction factor correlations are used in the THINC-IV computer code (described in Subdivision 4.4.3.4.1).

The friction factor in the axial direction, parallel to the fuel rod axis, is evaluated using the Novendstern-Sandberg correlation (Reference 66). This correlation consists of the following:

(Figure 4.4.3-8). The two heated channels were coupled by valves at six axial elevations. Upon detection of flow instability in the hot channel, the coupling valves were opened and both channels became stable, as illustrated in Figure 4.4.3-9. Holding all test parameters constant, the reclosing of these valves alone caused a flow in the hot channel to become unstable again. This behavior was observed on all runs performed and is presented as evidence:

1. That open channels are more stable than closed channels, and
2. Evaluations of the hydrodynamic stability of Westinghouse PWRs with HYDNA are conservative.

4.4.3.6 Temperature Transient Effects Analysis

Waterlogging damage of a fuel rod could occur as a consequence of a power increase on a rod after water has entered the fuel rod through a clad defect. Water entry will continue until the fuel rod internal pressure is equal to the reactor coolant pressure. A subsequent power increase raises the temperature and, hence, could raise the pressure of the water contained within the fuel rod. The increase in hydrostatic pressure within the fuel rod then drives a portion of the water from the fuel rod through the water entry defect. Clad distortion and/or rupture can occur if the fuel rod internal pressure increase is excessive due to insufficient venting of water to the reactor coolant. This occurs when there is both a rapid increase in the temperature of the water within the fuel rod and a small defect. Zircaloy clad fuel rods which have failed due to waterlogging (References 79 and 80) indicate that very rapid power transients are required for fuel failure. Normal operational transients are limited to about 40 cal/gm-min. (peak rod) while the Spert tests (Reference 79) indicate that 120 to 150 cal/gm is required to rupture the clad even with very short transients (5.5 msec.period). Release of the internal fuel rod pressure is expected to have a minimal effect on the Reactor Coolant System (Reference 79) and is not expected to result in failure of additional fuel rods (Reference 80). Ejection of fuel pellet fragments into the coolant stream is not expected (References 79 and 80). A clad breech due to waterlogging is thus expected to be similar to any fuel rod failure mechanism which exposes fuel pellets to the reactor coolant stream. Waterlogging has not been identified as the mechanism for clad distortion or perforation of any Westinghouse Zircaloy-4 clad fuel rods.

An excessively high fuel rod internal gas pressure could cause clad failure. One of the fuel rod design bases (Subdivision 4.2.1.1.1) requires the fuel rod internal gas pressure not to exceed the nominal coolant pressure even at the overpower condition. During operational transients, fuel rod clad rupture due to high internal gas pressure is precluded by meeting the above design basis.

4.4.3.7 Potentially Damaging Temperature Effects During Transients

The fuel rod experiences many operational transients (intentional maneuvers) during its residence in the core. A number of thermal effects must be considered when analyzing the fuel rod performance.

The clad can be in contact with the fuel pellet at some time in the fuel lifetime. Clad-pellet interaction occurs if the fuel pellet temperature is increased after the clad is in contact with the pellet. Clad-pellet interaction is discussed in Subdivision 4.2.1.3.1.

Increasing the fuel temperature results in an increased fuel rod internal pressure. One of the fuel rod design bases requires the fuel rod internal pressures not to exceed the nominal coolant pressure even at the overpower condition (Subdivision 4.2.1.1.1).

The potential effects of operation with waterlogged fuel are discussed in Subdivision 4.4.3.6 which concluded that waterlogging is not a concern during operational transients.

3 Clad flattening, as noted in Subdivision 4.2.1.3.1, has been observed in some operating power reactor. Thermal expansion (axial) of the fuel rod stack against a flattened section of clad could cause failure of the clad. This is no longer a concern because clad flattening is precluded during the fuel residence in the core. (See Subdivision 4.2.1.3.1).

There can be a differential thermal expansion between the fuel rods and the guide thimbles during a transient. Excessive mowing of the fuel rods could occur if the grid assemblies did not allow axial movement of the fuel rods relative to the grids. Thermal expansion of the fuel rods is considered in the grid design so that axial loads imposed on the fuel rods during a thermal transient will not result in excessively bowed fuel rods (See Subdivision 4.2.1.2.2).

4.4.3.8 Energy Release During Fuel Element Burnout

As discussed in Subdivision 4.4.3.3 the core is protected from going through DNB over the full range of possible operating conditions. At full power nominal
35 operation, the minimum DNBR was found to be 1.72. From Figure 4.4.2-8, this means that for these conditions, the probability of a rod going through DNB is less than 0.9% at a 95% confidence level, based on the statistics of the W-3 correlation. In the extremely unlikely event that DNB should occur, the clad temperature will rise due to the steam blanketing at the rod surface and the consequent degradation in heat transfer. During this time there is a potential for chemical reaction between the cladding and the coolant. However, because of the relatively good film boiling heat transfer following DNB, the
3 energy release resulting from this reaction is insignificant compared to the power produced by the fuel.

DNB With Physical Burnout - Westinghouse (Reference 72) has conducted DNB tests in a 25-rod bundle where physical burnout occurred with one rod. After this occurrence, the 25 rod test section was used for several days to obtain more DNB data from the other rods in the bundle. The burnout and deformation of the rod did not affect the performance of neighboring rods in the test section during the burnout or the validity of the subsequent DNB data points as predicted by the W-3 correlation. No occurrences of flow instability or other abnormal operation were observed.

DNB With Return to Nucleate Boiling - Additional DNB tests have been conducted by Westinghouse (Reference 81) in 19 and 21 rod bundles. In these tests, DNB without physical burnout was experienced more than once on single rods in the bundles for short periods of time. Each time, a reduction in power of approximately 10% was sufficient to re-establish nucleate boiling on the surface of the rod. During these and subsequent tests, no adverse effects were observed on this rod or any other rod in the bundle as a consequence of operating in DNB.

4.4.3.9 Energy Release or Rupture of Waterlogged Fuel Elements

A full discussion of waterlogging including energy release is contained in Subdivision 4.4.3.6. It is noted that the resulting energy release is not expected to affect neighboring fuel rods.

4.4.3.10 Fuel Rod Behavior Effects from Coolant Flow Blockage

Coolant flow blockages can occur within the coolant channels of a fuel assembly or external to the reactor core. The effects of fuel assembly blockage within the assembly on fuel rod behavior is more pronounced than external blockages of the same magnitude. In both cases the flow blockages cause local reductions in coolant flow. The amount of local flow reduction, where it occurs in the reactor, and how far along the flow stream the reduction persists are considerations which will influence the fuel rod behavior. The effects of coolant flow blockages in terms of maintaining rated core performance are determined both by analytical and experimental methods. The experimental data are usually used to augment analytical tools such as computer programs similar to the THINC-IV program. Inspection of the DNB correlation (Subdivision 4.4.2.3 and Reference 44) shows that the predicted DNBR is dependent upon the local values of quality and mass velocity.

The THINC-IV code is capable of predicting the effects of local flow blockages on DNBR within the fuel assembly on a subchannel basis, regardless of where the flow blockage occurs. In Reference 63, it is shown that for a fuel assembly similar to the Westinghouse design, THINC-IV accurately predicts the flow distribution within the fuel assembly when the inlet nozzle is completely blocked. Full recovery of the flow was found to occur about 30 inches downstream of the blockage. With the reference reactor operating at the nominal full power conditions specified in Table 4.4.2-1, the effects of an increase in enthalpy and decrease in mass velocity in the lower portion of the fuel assembly would not result in the reactor reaching a minimum DNBR of 1.30.

From a review of the open literature it is concluded that flow blockage in "open lattice cores" similar to the Westinghouse cores cause flow perturbations which are local to the blockage. For instance, A. Oktsubo, et.al (Reference 82), show that the mean bundle velocity is approached asymptotically about 4 inches downstream from a flow blockage in a single flow cell. Similar results were also found for 2 and 3 cells completely blocked. Basmer (Reference 83), et.al., tested an open lattice fuel assembly in which 41% of the subchannels were completely blocked in the center of the test bundle between spacer grids. Their results show the stagnant zone behind the flow blockage essentially disappears after 1.65 L/De or about 5 inches for their test bundle. They also found that leakage flow through the blockage tended to shorten the stagnant zone or in essence the complete recovery length. Thus, local flow blockages

within a fuel assembly have little effect on subchannel enthalpy rise. The
3 reduction in local mass velocity is then the main parameter which affects the DNBR. If the McGuire units were operating at full power and nominal steady state conditions as specified in Table 4.4.2-1, a reduction in local mass
35 veloc.ty greater than 53% would be required to reduce the DNBR from 1.72 to 1.30. The above mass velocity effect on the DNB correlation was based on the assumption of fully developed flow along the full channel length. In reality a local flow blockage is expected to promote turbulence and thus would likely not effect DNBR at all.

Coolant flow blockages induce local crossflows as well as promote turbulence. Fuel rod behavior is changed under the influence of a sufficiently high crossflow component. Fuel rod vibration could occur, caused by this crossflow component, through vortex shedding or turbulent mechanisms. If the crossflow velocity exceeds the limit established for fluid elastic stability, large amplitude whirling results. The limits for a controlled vibration mechanism are established from studies of vortex shedding and turbulent pressure fluctuations. The crossflow velocity required to exceed fluid elastic stability limits is dependent on the axial location of the blockage and the characterization of the crossflow (jet flow or not). These limits are greater than those for vibratory fuel rod wear. Crossflow velocity above the established limits can lead to mechanical wear of the fuel rods at the grid support locations. Fuel rod wear due to flow induced vibration is considered in the fuel rod fretting evaluation (Section 4.2).

4.4.4 TESTING AND VERIFICATION

3 4.4.4.1 Tests Prior to Initial Criticality

A reactor coolant flow test, as noted in Table 14.1.4-1, is performed following fuel loading but prior o initial criticality. Coolant loop pressure drop data is obtained in this test. This data in conjunction with coolant pump performance information allows determination of the coolant flow rates at reactor operating conditions. This test verifies that proper coolant flow rates have been used in the core thermal and hydraulic analysis.

4.4.4.2 Initial Power and Plant Operation

Core power distribution measurements are made at several core power levels (see Subdivision 4.3.2.2.7). These tests are used to insure that conservative peaking factors are used in the core thermal and hydraulic analysis.

Additional demonstration of the overall conservatism of the THINC analysis was obtained by comparing THINC predictions to incore thermocouple measurements. These measurements were performed on the Zion reactor (Reference 84). No further in-pile testing is planned.

4.4.4.3 Component and Fuel Inspections

Inspections performed on the manufactured fuel are delineated in Subdivision 4.2.1.4. Fabrication measurements critical to thermal and hydraulic analysis are obtained to verify that the engineering hot channel factors employed in the design analyses (Subdivision 4.4.2.3.4) are met.

Table 4.1-1 (1 of 3)

Reactor Design Comparison Table

	THERMAL AND HYDRAULIC DESIGN PARAMETERS	MCGUIRE UNITS 1 & 2 17x17 FUEL ASSEMBLY WITH DENSIFICATION EFFECTS	REFERENCE PLANT 17x17 FUEL ASSEMBLY WITH DENSIFICATION EFFECTS
	1. Reactor Core Heat Output, MWt	3411	3411
	2. Reactor Core Heat Output, Btu/hr	$11,641.7 \times 10^6$	$11,641.7 \times 10^6$
	3. Heat Generated in Fuel, %	97.4	97.4
3	4. System Pressure, Nominal, psia	2250	2250
	5. System Pressure, Min. Steady State, psia	2220	2220
	6. Minimum DNBR for Design Transients	>1.30	>1.30
	Coolant Flow		
	7. Total Thermal Flow Rate, lb/hr	144.7×10^6	132.7×10^6
35	8. Effective Flow Rate for Heat Transfer, lb/hr	133.9×10^6	126.7×10^6
3	9. Effective Flow Area for Heat Transfer, ft^2	51.1	51.1
35	10. Average Velocity Along Fuel Rods, ft/sec	16.6	15.7
	11. Average Mass Velocity, lb/hr-ft^2	2.62×10^6	2.48×10^6
3	Coolant Temperature, $^{\circ}F$		
	12. Nominal Inlet	559.1	552.5
35	13. Average Rise in Vessel	58.4	64.2
	14. Average Rise in Core	62.5	66.9
	15. Average in Core	590.4	585.9
	16. Average in Vessel	588.3	584.7
	Heat Transfer		
	17. Active Heat Transfer, Surface Area, ft^2	59,700	59,700
	18. Average Heat Flux, Btu/hr-ft^2	189,800	189,800
	19. Maximum Heat Flux for Normal Operation, Btu/hr-ft^2	440,300[d]	474,500[a]
	20. Average Thermal Output, kw/ft	5.44	5.44
	21. Maximum Thermal Output for Normal Operation, kw/ft	12.5[d]	13.6[a]
	22. Peak Linear Power for Determination of Protection Setpoints, kw/ft	18.0[c]	18.0[c]
	23. Heat Flux Hot Channel Factor, F_Q	2.32[b]	2.50[b]
3	Fuel Central Temperature, $^{\circ}F$		
	24. Peak at 100% Power	3250	3400
	25. Peak at Maximum Thermal Output for Maximum Overpower Trip Point	4150	4100

[a] This limit is associated with the value of $F_Q = 2.50$
[b] Includes the effect of fuel densification
[c] See Subparagraph 4.3.2.2.6
[d] This limit is associated with the value of F_Q -2.32

	CORE MECHANICAL DESIGN PARAMETERS	MCGUIRE UNITS 1 & 2 17x17 FUEL ASSEMBLY WITH DENSIFICATION EFFECTS	REFERENCE PLANT 17x17 FUEL ASSEMBLY WITH DENSIFICATION EFFECTS
	Fuel Assemblies		
	26. Design	RCC Canless	RCC Canless
	27. Number of Fuel Assemblies	193	193
	28. UO_2 Rods per Assembly	264	264
	29. Rod Pitch, in.	0.496	0.496
	30. Overall Dimensions, in.	8.426 x 8.426	8.426 x 8.426
	31. Fuel Weight (as UO_2), pounds	222,739	222,739
	32. Zircaloy Weight, lbs.	50,913	50,913
	33. Number of Grids per Assembly	8-Type R	8-Type R
3	34. Loading Technique	3 region non-uniform	3 region non-uniform
	Fuel Rod		
	35. Number	50,952	50,952
	36. Outside Diameter, in.	0.374	0.374
	37. Diametral Gap, in., Regions 1,2, and 3	0.0065	0.0065
	38. Clad Thickness, in.	0.0225	0.0225
	39. Clad Material	Zircaloy-4	Zircaloy-4
	Fuel Pellets		
	40. Material	UO_2 Sintered	UO_2 Sintered
	41. Density (% of Theoretical)	95	95
	42. Diameter, in., Regions 1,2, and 3	0.3225	0.3225
	43. Length, in.	0.530	0.530
	Rod Cluster Control Assemblies		
35	44. Neutron Absorber	Ag-In-Cd	Ag-In-Cd
3	45. Cladding Material	Type 304 SS-Cold Worked	Type 304 SS-Cold Worked
	46. Clad Thickness, in.	0.0185	0.0185
35	47. Number of Clusters, Full	53	53
	48. Number of Absorber Rods per Cluster	24	24
	Core Structure		
	49. Core Barrel, I.D./O.D., in.	148.0/152.5	148.0/152.5
	50. Thermal Shield, I.D./O.D., in.	Neutron Pad Design	Neutron Pad Design
	NULCEAR DESIGN PARAMETERS		
	Structure Characteristics		
3	51. Core Diameter, in. (Equivalent)	132.7	132.7
	52. Core Average Active Fuel Height, in.	143.7	143.7
	Reflector Thickness and Composition		
	53. Top - Water plus Steel, in.	~10	~10
	54. Bottom - Water plus Steel, in.	~10	~10
	55. Side - Water plus Steel in.	~15	~15
	56. H_2O/U, Cold Molecular Ratio Lattice	3.43	3.43

<u>Fuel Rods</u>	
Number	50,952
Outside Diameter, in.	0.374
Diameter Gap, in.	0.0065
Clad Thickness, in.	0.0225
Clad Material	Zircaloy-4
<u>Fuel Pellets</u>	
Material	UO_2 Sintered
Density (percent of Theoretical)	95
Fuel Enrichments w/o	
Region 1	2.10
Region 2	2.60
3 Region 3	3.10
Diameter, in.	0.3225
Length, in.	0.530
Mass of UO_2 per Foot of Fuel Rod, lb/ft	0.364
<u>Rod Cluster Control Assemblies</u>	
Neutron Absorber	Ag-In-Cd
Composition	80%, 15%, 5%
Diameter, in.	0.341
Density, lbs/in.^{3}	0.367
Cladding Material	Type 304, Cold Worked Stainless Steel
Clad Thickness, in.	0.0185
Number of Clusters	
Full Length	53
35	
Number of Absorber Rods per Cluster	24
3 Full Length Assembly Weight (dry), lb.	157

3

Burnable Poison Rods (First Core)	
35 Number	1518
Material	Borosilicate Glass
Outside Diameter, in.	0.381
Inner Tube, O.D., in.	0.1805
Clad Material	Stainless Steel
Inner Tube Material	Stainless Steel
Boron Loading w/o B_2O_3 in glass rod)	12.5
Weight of Boron - 10 per foot of rod, lb/ft	.000419
Initial Reactivity Worth, %Δρ	7.63 (hot), ~5.5 (cold)
3	
Excess Reactivity	
Maximum Fuel Assembly K∞ (Cold, Clean, Unborated Water)	1.39
Maximum Core Reactivity (Cold, Zero Power, Beginning of Cycle)	1.222

Table 4.3.3-5

Comparison of Measured and Calculated Moderator

Coefficients at HZP, BOL

Unit Type/ Control Bank Configuration	Measured α_{iso}(1) (pcm/°F)	Calculated α_{iso} (pcm/°F)
3-Loop, 157 Assemblies, 12 foot core		
D at 160 steps	-0.50	-0.50
D in, C at 190 steps	-3.01	-2.75
D in, C at 28 steps	-7.67	-7.02
B, C, and D in	-5.16	-4.45
2-Loop, 121 Assemblies, 12 foot core		
D at 180 steps	+0.85	+1.02
D in, C at 180 steps	-2.40	-1.90
C and D in, B at 165 steps	-4.40	-5.58
B, C, and D in A at 174 steps	-8.70	-8.12

(1) Isothermal coefficients, which include the Doppler effect in the fuel.

$$\alpha_{in} = 10^5 \ln \frac{k_2}{k_1} / \Delta T°F$$

Table 4.4.2-1

Reactor Design Comparison Table

	Thermal and Hydraulic Design Parameters	McGuire 1 & 2 17 x 17 With Densification	Reference Plant 17 x 17 With Densification
	Reactor Core Heat Output, MWt	3411	3411
	Reactor Core Heat Output, BTU/hr	$11,641.7 \times 10^6$	$11,641.7 \times 10^6$
3	Heat Generated in Fuel, %	97.4	97.4
	System Pressure, Nominal, psia	2250	2250
	System Pressure, Minimum Steady State, psia	2220	2220
	Minimum DNBR at Nominal Conditions		
35	Typical Flow Channel	2.05	2.04
	Thimble (Cold Wall) Flow Channel	1.72	1.71
	Minimum DNBR for Design Transients	>1.30	>1.30
3	DNB Correlation	"R" (W-3 with modified spacer factor)	"R" (W-3 with modified spacer factor)
	Coolant Flow		
	Total Thermal Flow Rate, lb/hr	144.7×10^6	132.7×10^6
35	Effective Flow Rate for Heat Transfer, lb/hr	133.9×10^6	126.7×10^6
3	Effective Flow Area for Heat Transfer, Ft^2	51.1	51.1
35	Average Velocity Along Fuel Rods, ft/sec	16.6	15.7
	Average Mass Velocity, lb/hr-ft^2	2.62×10^6	2.48×10^6
3	Coolant Temperature		
	Nominal Inlet, °F	559.1	552.5
	Average Rise in Vessel, °F	58.4	64.2
35	Average Rise in Core, °F	62.5	66.9
	Average in Core, °F	590.4	585.9
	Average in Vessel, °F	588.3	584.7

Table 4.4.2-1

Reactor Design Comparison Table

Thermal and Hydraulic Design Parameters	McGuire 1 & 2 17 x 17 With Densification	Reference Plant 17 x 17 With Densification
Heat Transfer		
Active Heat Transfer, Surface Area, Ft^2	59,700	59,700
Average Heat Flux, $BTU/hr\text{-}ft^2$	183,800	189,800
Maximum Heat Flux, for normal operation $BTU/hr\text{-}ft^2$	440,300[d]	474,500[a]
Average Thermal Output, kW/ft	5.44	5.44
Maximum Thermal Output, for normal operation kW/ft	12.6[d]	13.6[a]
Peak Linear Power for Determination of protection setpoints, kW/ft	18.0[c]	18.0[c]
Fuel Central Temperature		
Peak at 100% Power, ^{o}F	3250	3400
Peak at Thermal Output Maximum for Maximum Overpower Trip Point, ^{o}F	4150	4150
Pressure Drop[b]		
35 Across Core, psi	27.6 $\pm$ 5.6	25.0 $\pm$ 5.0
Across Vessel, including nozzle, psia	47.9 $\pm$ 7.2	42.6 $\pm$ 6.4

[a] This limit is associated with the value of $F_Q = 2.50$

[b] Based on best estimate reactor flow as discussed in Section 5.1

[c] See Subdivision 4.3.2.2.6

[d] This limit is associated with the value of $F_Q = 2.32$

Table 4.4.2-2

Thermal-Hydraulic Design Parameters For

One Of Four Coolant Loops Out Of Service

	Without Loop Stop Valves
Total Core Heat Output, MWt	2389
Total Core Heat Output, 10^6 BTU/hr	8154
Heat Generated in Fuel, %	97.4
Nominal System Pressure, psia	2250
Coolant Flow	
Effective Thermal Flow Rate for Heat Transfer, 10^6 lbs/hr	96.4
Effective Flow Area for Heat Transfer, ft^2	51.1
Average Velocity Along Fuel Rods, ft/sec	11.8
Average Mass Velocity, 10^6 lb/hr-ft^2	1.89
Coolant Temperature, ^oF	
Design Nominal Inlet	551.7
Average Rise in Core	62.3
Average in Core	584.4
Heat Transfer	
Active Heat Transfer Surface Area, ft^2	59,700
Average Heat Flux, BTU/hr-ft^2	132,900
Minimum DNB Ratio at Nominal Conditions	>1.74
Minimum DNB Ratio for Design and Anticipated Transients	$\geq$1.30

(Figure 4.2.3-4 deleted)

(Figure 4.2.3-12 deleted)

	H	G	F	E	D	C	B	A
8	1.097							
9	1.029	1.120						
10	1.154	1.074	1.186					
11	1.167	1.210	1.163	1.187				
12	1.225	1.171	1.190	1.065	1.241			
13	1.127	1.163	1.094	1.088	.915	.969		
14	1.026	1.026	.991	.976	.823	.463		
15	.715	.779	.662	.560				

CALCULATED $F^{N}_{\Delta H} = 1.34$

KEY:
VALUE REPRESENTS ASSEMBLY RELATIVE POWER

NORMALIZED POWER DENSITY DISTRIBUTION NEAR BEGINNING OF LIFE, UNRODDED CORE, HOT FULL POWER, EQUILIBRIUM XENON

McGUIRE NUCLEAR STATION

Figure 4.3.2-7
Revision 3
New Figure

	H	G	F	E	D	C	B	A
8	0.70							
9	0.94	1.09						
10	1.19	1.11	1.23					
11	1.25	1.28	1.20	1.14				
12	1.33	1.25	1.22	0.95	0.75			
13	1.24	1.26	1.13	1.02	0.75	0.81		
14	1.13	1.12	1.05	0.97	0.76	0.41		
15	0.80	0.86	0.71	0.58				

CALCULATED $F^N_{\Delta H} = 1.44$

KEY:
VALUE REPRESENTS
ASSEMBLY
RELATIVE POWER

NORMALIZED POWER DENSITY
DISTRIBUTION NEAR BEGINNING
OF LIFE, GROUP D 28% INSERTED,
HOT FULL POWER, EQUILIBRIUM XENON

McGUIRE NUCLEAR STATION
Figure 4.3.2-8
Revision 35

(Figure 4.3.2-9 deleted)

	H	G	F	E	D	C	B	A
8	1.154							
9	1.189	1.152						
10	1.153	1.189	1.153					
11	1.212	1.152	1.208	1.143				
12	1.137	1.191	1.128	1.149	1.203			
13	1.143	1.087	1.123	1.048	.993	.945		
14	.972	1.039	.940	.962	.797	.483		
15	.707	.739	.657	.542				

CALCULATED $F^N_{\Delta H} = 1.30$

KEY:
VALUE REPRESENTS ASSEMBLY
RELATIVE POWER

NORMALIZED POWER DENSITY
DISTRIBUTION NEAR MIDDLE OF
LIFE, UNRODDED CORE, HOT FULL
POWER, EQUILIBRIUM XENON

McGUIRE NUCLEAR STATION

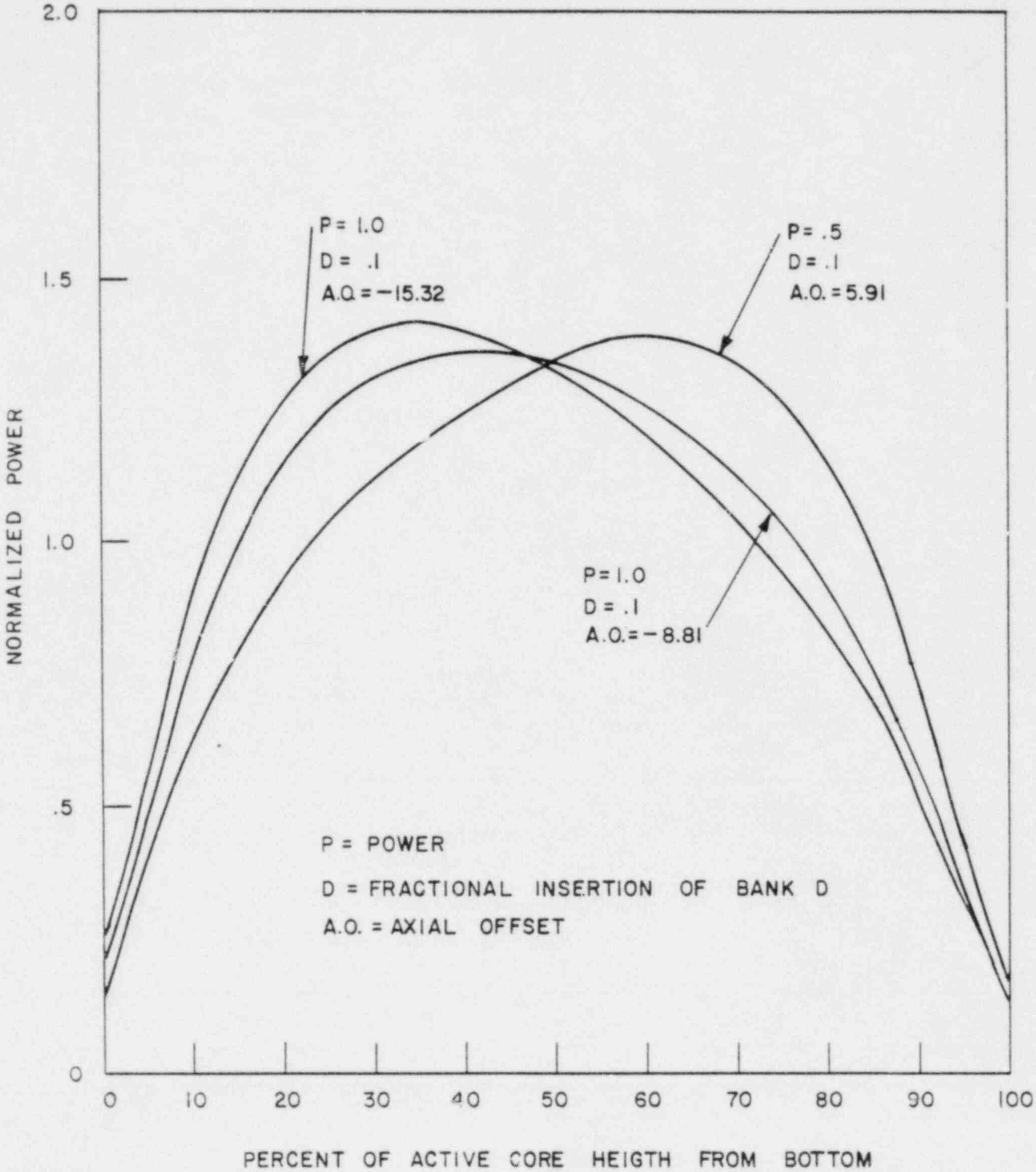
Figure 4.3.2-10
Revision 3
New Figure

0.98																
0.98	0.99															
0.98	1.00	1.03														
0.99	1.02	1.07	X													
1.00	1.03	1.09	1.11	1.09												
1.01	1.06	X	1.11	1.11	X											
1.00	1.04	1.08	1.07	1.08	1.10	1.08										
1.00	1.04	1.08	1.07	1.08	1.10	1.08	1.08									
1.01	1.06	X	1.09	1.10	X	1.10	1.10	X								
1.00	1.04	1.08	1.07	1.08	1.10	1.08	1.08	1.10	1.08							
1.00	1.04	1.08	1.07	1.08	1.10	1.08	1.08	1.10	1.08	1.08						
1.01	1.06	X	1.11	1.11	X	1.10	1.10	X	1.10	1.10	X					
1.00	1.03	1.09	1.11	1.09	1.11	1.08	1.08	1.10	1.08	1.08	1.11	1.09				
0.99	1.02	1.07	X	1.11	1.11	1.07	1.07	1.09	1.07	1.07	1.11	1.11	X			
0.98	1.00	1.03	1.07	1.09	X	1.08	1.08	X	1.08	1.08	X	1.09	1.07	1.03		
0.98	0.99	1.00	1.02	1.03	1.06	1.04	1.04	1.06	1.04	1.04	1.06	1.03	1.02	1.00	0.98	
0.98	0.98	0.98	0.99	1.00	1.01	1.00	1.00	1.01	1.00	1.00	1.01	1.00	0.99	0.98	0.98	0.98

RODWISE POWER DISTRIBUTION
IN A TYPICAL ASSEMBLY (G-9)
NEAR END OF LIFE, HOT FULL
POWER, EQUILIBRIUM XENON,
UNRODDED CORE

McGUIRE NUCLEAR STATION

Figure 4.3.2-13
Revision 3
New Figure

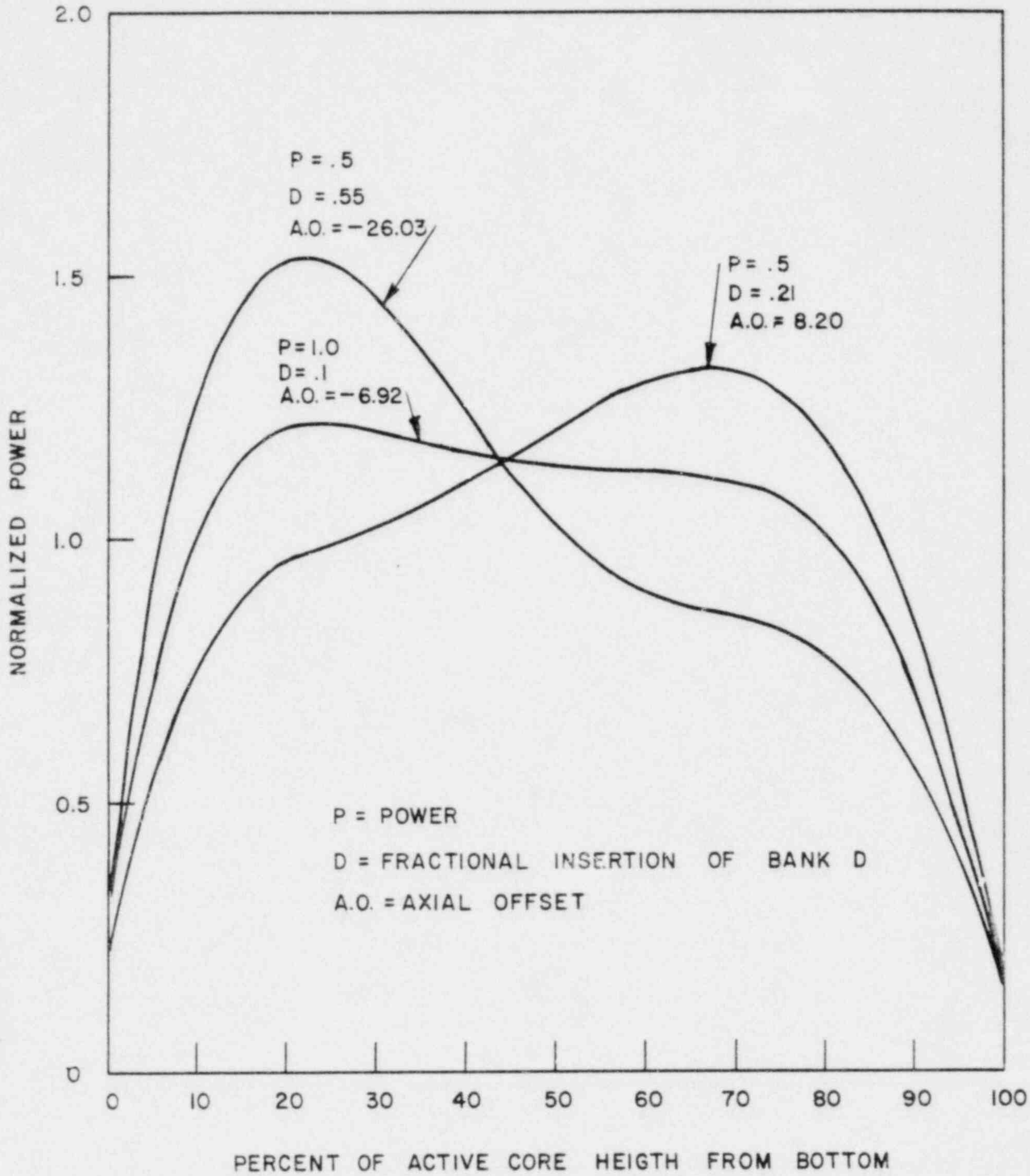


TYPICAL AXIAL POWER SHAPES
OCCURRING AT BEGINNING OF LIFE

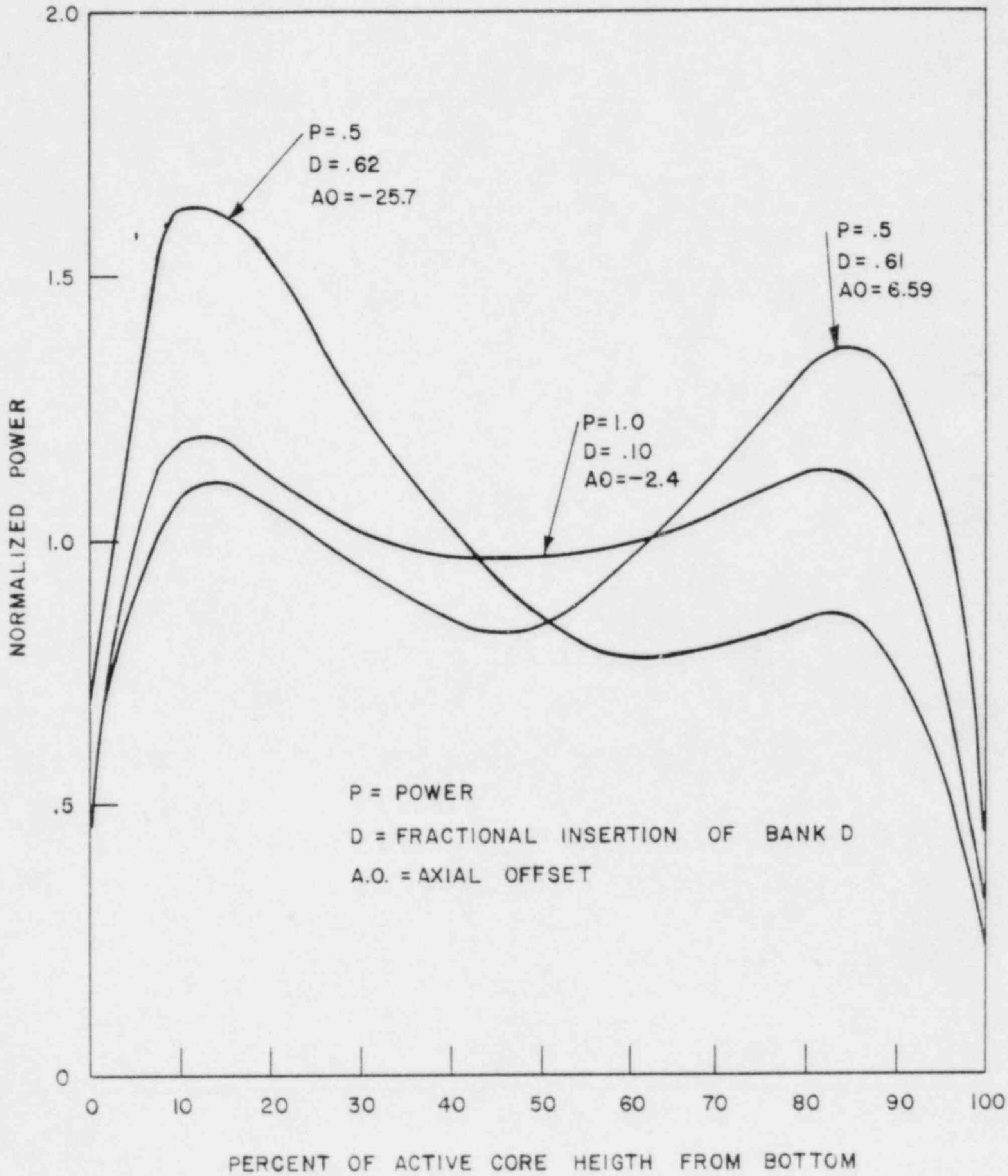
McGUIRE NUCLEAR STATION

Figure 4.3.2-14

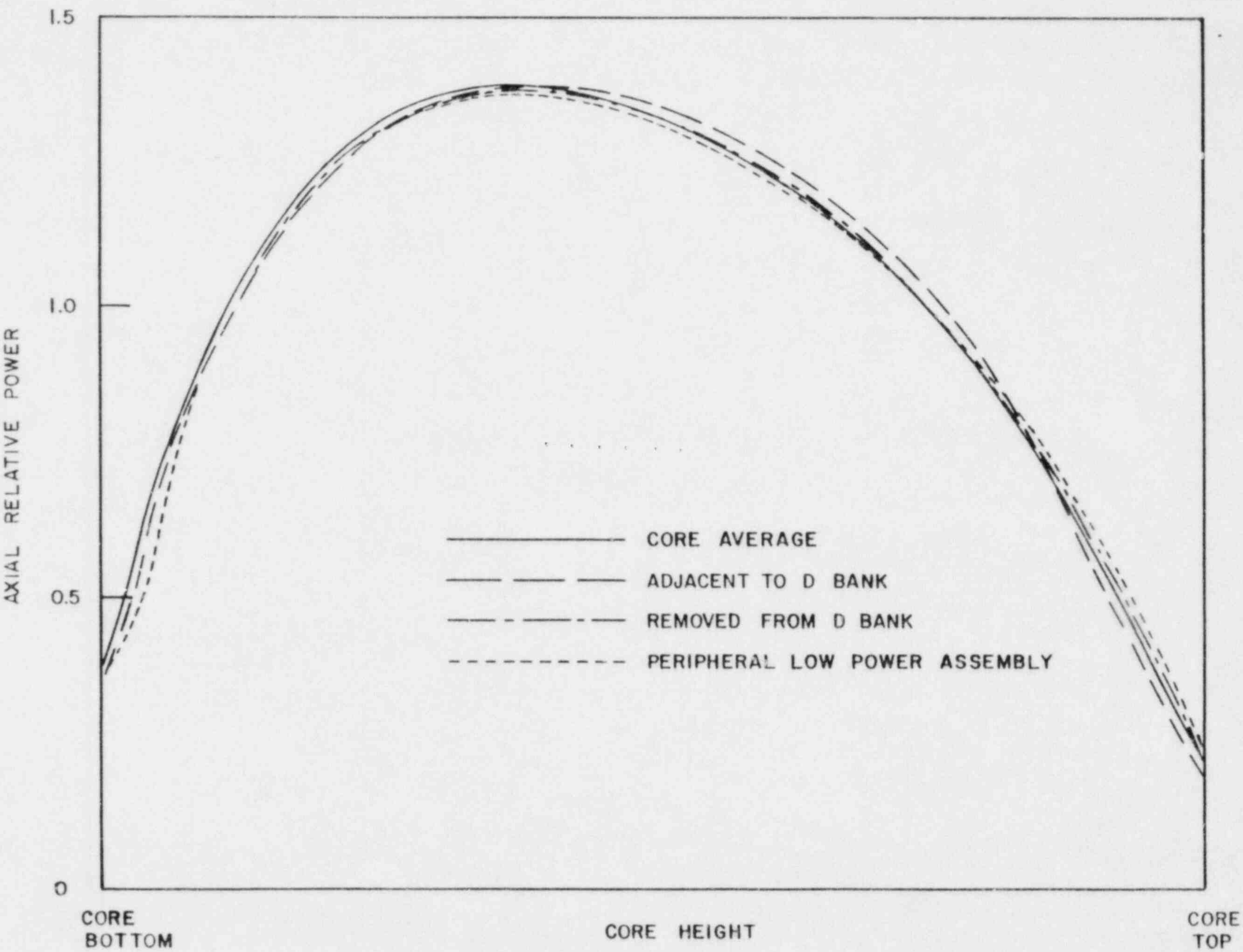
Revision 35



TYPICAL AXIAL POWER SHAPES
OCCURRING AT MIDDLE OF LIFE
McGUIRE NUCLEAR STATION
Figure 4.3.2-15
Revision 35

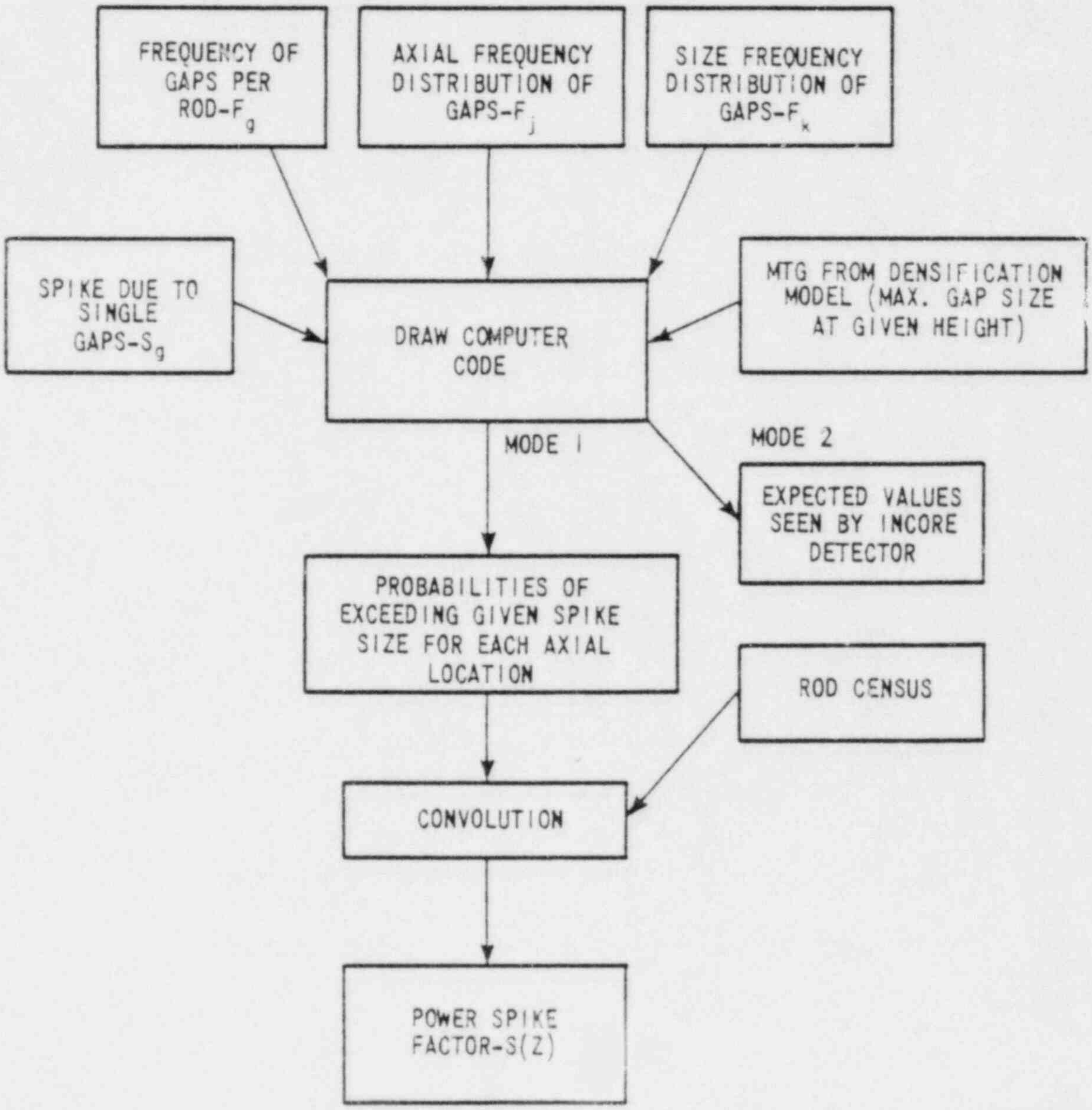


TYPICAL AXIAL POWER SHAPES
OCCURRING AT END OF LIFE
McGUIRE NUCLEAR STATION
Figure 4.3.2-16
Revision 35



COMPARISON OF ASSEMBLY AXIAL POWER DISTRIBUTION WITH CORE AVERAGE AXIAL DISTRIBUTION, BANK D SLIGHTLY INSERTED

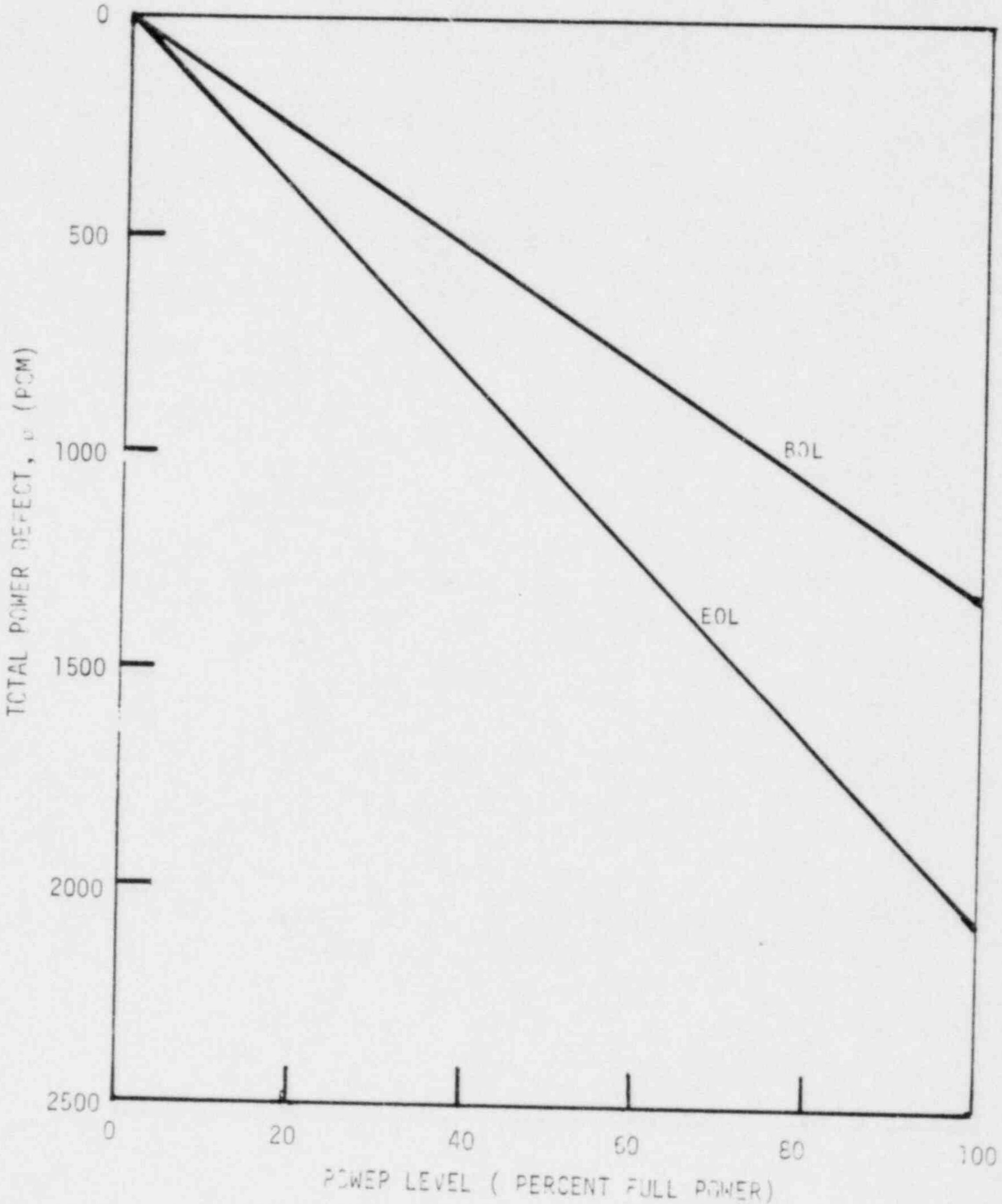
McGUIRE NUCLEAR STATION
Figure 4.3.2-17
Revision 35



FLOW CHART FOR DETERMINING
SPIKE MODEL

McGUIRE NUCLEAR STATION

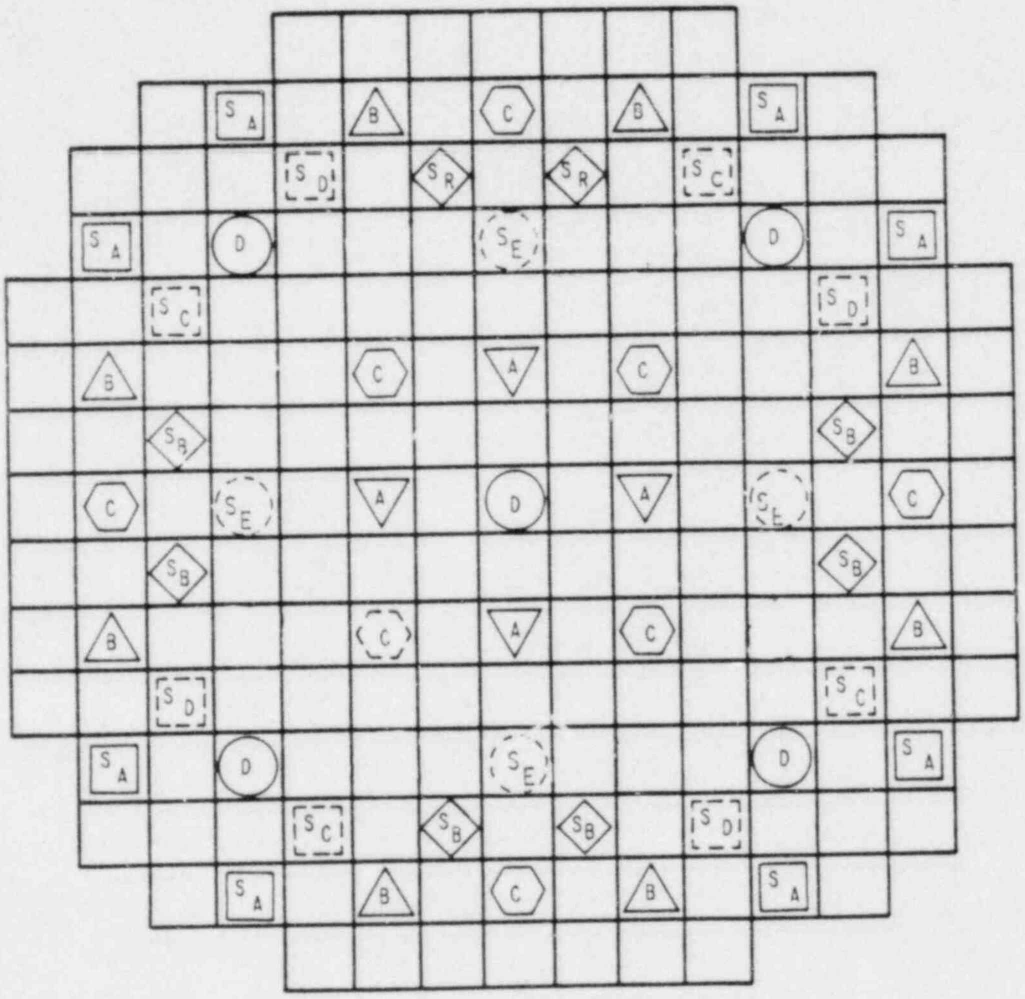
Figure 4.3.2-18
Revision 3
New Figure



TOTAL POWER DEFECT - BEGINNING OF LIFE, END OF LIFE, CYCLE 1

McGUIRE NUCLEAR STATION

Figure 4.3.2-36
Revision 3
New Figure

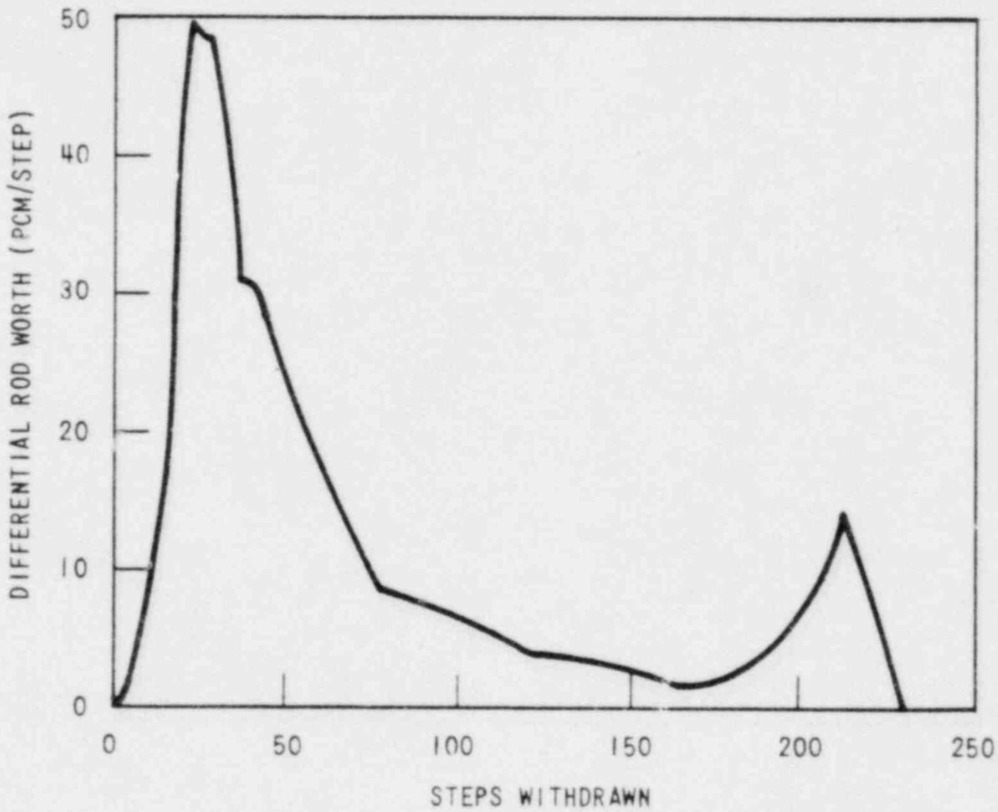


FUNCTION		NUMBER OF ROD CLUSTERS
SHUTDOWN BANK	S_A	8
SHUTDOWN BANK	S_B	8
SHUTDOWN BANK	S_C & S_D	4 ; 4
CONTROL BANK	A	4
CONTROL BANK	B	8
CONTROL BANK	C	8
CONTROL BANK	D	9
SHUTDOWN BANK	S_E	4

ROD CLUSTER CONTROL
ASSEMBLY PATTERN

McGUIRE NUCLEAR STATION

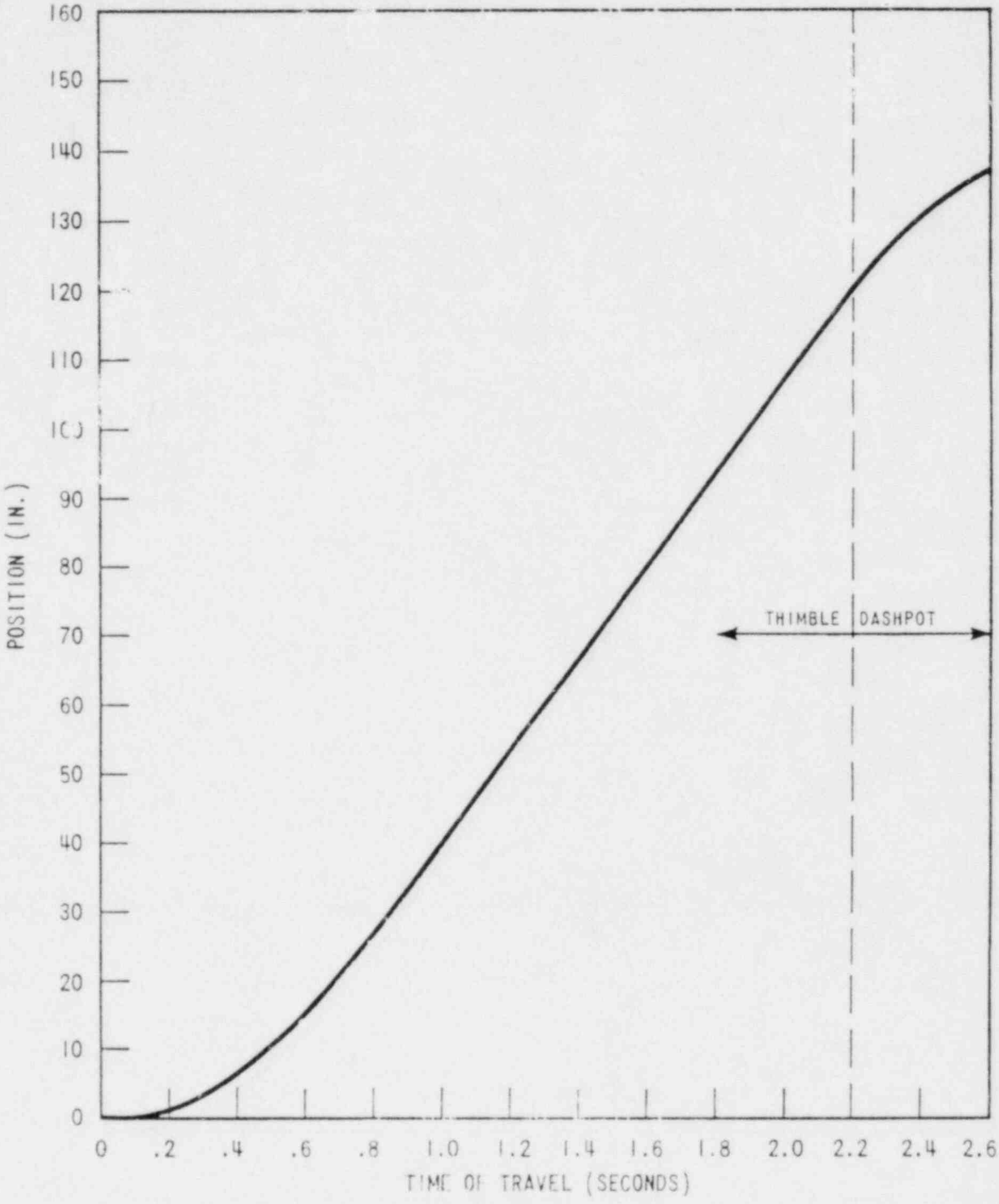
Figure 4.3.2-37
Revision 35



ACCIDENTAL SIMULTANEOUS
WITHDRAWAL OF TWO CONTROL
BANKS EOL, HZP BANKS B AND
D MOVING IN THE SAME PLANE

McGUIRE NUCLEAR STATION

Figure 4.3.2-38
Revision 35



DESIGN - TRIP CURVE

McGUIRE NUCLEAR STATION

Figure 4.3.2-39
Revision 3
New Figure

b. Pressurizer Relief Tank Level

The pressurizer relief tank level transmitter supplies a signal for an indicator and high and low level alarms.

The Reactor Coolant System design and operating pressure together with the safety, power relief and pressurizer spray valve setpoints, and the protection system setpoint pressures are listed in Table 5.2.2-2.

The design pressure allows for operating transient pressure changes. The selected design margin considers core thermal lag, coolant transport times and pressure drops, instrumentation and control accuracy and response characteristics, and system relief valve characteristics.

Process control instrumentation for the Residual Heat Removal System is provided for the following purposes:

1. Furnish input signals for monitoring and/or alarming purposes for:

    a. Temperature indications
    b. Pressure indications
    c. Flow indications

2. Furnish input signals for control purposes of such processes as follows:

    a. Control valve in the residual heat removal pump bypass line so that it opens at flows below a preset limit and closes at flows above a preset limit.

    b. Residual heat removal inlet valves control circuitry. See Section 7.6 for the description of the interlocks and requirements for automatic closure.

    c. Control valve in the residual heat removal heat exchanger bypass line to control temperature of reactor coolant returning to reactor coolant loops during cooldown.

    d. Residual heat removal pump circuitry for starting residual heat removal pumps on "S" signal.

Table 5.1-1
System Design and Operating Parameters

	Unit Design Life, years	40
	Nominal Operating Pressure, psig	2235
26	Total System Volume, including pressurizer and surge line, ft^3	12,040
26	System Liquid Volume, including pressurizer water level (60% full) at maximum guaranteed power, ft^3	11,298
	NSSS Power, BTU/hr	$11,687 \times 10^6$

System Thermal and Hydraulic Data Temperatures
(Based on Thermal Design Flow)

35	Thermal Design Flow, GPM/Loop	97,500
	Total Reactor Coolant Flow, lb/hr	144.7×10^6
	Reactor Vessel Inlet Temperature, °F	559.1
	Reactor Vessel Outlet Temperature, °F	617.5
	Steam Generator Outlet Temperature, °F	558.8
	Steam Pressure at Full Power, psia	1000.0
	Steam Generator Steam Temperature, °F	544.6
	Steam Flow at Full Power, lb/hr (total)	15.14×10^6
	Feedwater Inlet Temperature, °F	440.0
	Pressurizer Spray Rate, max., gpm	900
	Pressurizer Heat Capacity, kw	1800
	Pressurizer Relief Tank Volume, ft^3	1800

Flows and Pressure Drops
(Based on Best Estimate Flow)

35	Best Estimate Flow, GPM/Loop	101,700
	Pump Head at B.E. Flow, ft	276
	Reactor Vessel ΔP, psi	47.9
	Steam Generator ΔP, psi	36.7
	Piping ΔP, psi	8.0

The long term containment transient calculation was reevaluated using the LOTIC-I computer code with initial containment temperatures of 60°F and reduced power. This calculation was performed using the same input and assumptions as those detailed in Subdivision 6.2.1.3.4 except postfroth core power was 5% of ESDR power, and the initial temperature in the upper, lower and dead end compartment was 60°F. It should be noted that this calculation has the following conservatisms:

1) The blowdown, reflood, and froth mass and energy release were done at 100% of ESDR power.

2) All structural heat sinks were at temperatures consistent with the current analyses.

For this case the ice bed melted out at approximately 67500 seconds. The peak pressure following ice bed meltant was less than 4.0 psig. Since design pressure is 15.0 psig, this transient results in insignificant consequences with respect to demonstration of containment structural integrity.

The insulation cavity is filled with a low density, closed cell, foam concrete. The nominal density of the foam concrete is 35 lbs/ft^3; the compressive strength is 110 psi. The thermal conductivity per inch thickness is nominally 1.0 $Btu/hr\text{-}F\text{-}ft^2$. The insulation cavity for the foam concrete is sealed by a vapor barrier to provide additional assurance that the insulation section resists infusion of water vapor thus retains a high thermal resistance. The top surface of the foam concrete is covered with a course of grouting which provides seating surface for the floor plate and cooling coil assemblies.

Floor Drain

Special consideration has been given in the design to prevent freezing of the floor drains and to minimize check valve leakage.

The floor drains employ a low thermal conductivity (transite) section of pipe 12 inches in diameter, inserted vertically below the wear slab
2 to minimize heat gain to the ice bed. The horizontal run is a 12 inch diameter steel pipe embedded in the subfloor which is at a relatively warm temperature. The drain check valve is a 12 inch diameter horizontal valve fabricated from 304 or 316 SS welded per AWS D1.1-1972. The valve is designed to remain closed against the cold air head in the ice condenser to minimize any heat inleakage and air outleakage during normal operation. The valve is designed to tolerate a 15 psi back pressure when closed. The check valve is in a warm environment and no freezing will occur. To
35 minimize air currents induced by the temperature difference between the ice bed and the floor drain piping, the floor drain openings are covered with water soluble paper held in place with water soluble tape.

6.2.2.1.3 Design Evaluation

Wear Slab

The wear slab, during normal operating conditions, is subject only to its dead weight consisting of concrete, steel reinforcing, steel plates and piping. Six inches of 100 percent density ice is assumed to be uniformly distributed over the entire floor. The dead weight amounts to 11,200 lbs per bay, the equivalent of 0.56 psi. The live load for maintenance purposes is assumed to be 250 lbs/ft^2. The vertical seismic input is 0.35 g for 1/2 SSE and 0.55 g for SSE. The dead load plus seismic loads are insignificant because the highest load on the floor is contributed by blowdown pressure during design accident conditions. The blowdown pressure is 9 psi, and added to this value, for design purposes, is a 40 percent design margin, and a dynamic load factor of 1.53. This results in a minimum value for design of 19.28 psi.

The most severe loading condition is the combination of the dead load, the SSE seismic acceleration of 0.55 g, the 19.28 psi pressure load and 8.1 psi locally near the deflectors due to flow inpulse loadings. The wear slab is designed to accommodate the heatup and cooldown cycles and 1/2 SSE without overstressing the concrete and coolant piping.

Floor Cooling System

The embedded piping for floor cooling is 1/2 inch schedule 80 pipe. The maximum coolant pressure in the pipe is approximately 100 psi. ANSI B31.5-68 data shows that the pipe can tolerate internal pressures of 4812 psi.

In addition, the piping is tested to 200 psi. The pipe is sized to allow for at least 38 mils of corrosion. Nevertheless, the glycol coolant contains corrosion inhibitors, and as a result pipe corrosion is negligible. The 1/4 inch floor plate is integrated with the concrete through 1/2 inch diameter anchors welded to the plate on 12 inch centers. These anchors prevent thermal loads from concentrating in the piping.

Insulation Section

The insulation section supports wear slab loads. For a conservative analysis the wear slab dead weight + seismic + DBA loads were assumed to be transferred to the foam concrete section. The compressive strength of the foam concrete is sufficient to accept these floor loads.

Floor Drain

Drains are provided at the bottom of the ice condenser compartment to allow the melt/condensate water to flow out of the compartment during a loss-of-coolant accident. These drains are provided with check valves that are designed to seal the ice condenser during normal plant operation to prevent loss of cold air from the ice condenser. These check valves remain closed against the cold air head (1 psf) of the ice condenser and open before the water head reaches a value of 18 inches of water.

For a small pipe break, the water inventory in the ice condenser is produced in proportion to the energy added from the accident. The water collecting on the floor of the condenser compartment then flows out through the drains. For intermediate and large pipe breaks the ice condenser doors are open and water drains through both the doors and the drains.

For a large pipe break, a short time of the order of seconds is required for the water to fall from the ice condenser to the floor of the compartment. Results of fullscale section tests performed at Waltz Mill show that, for the design blowdown accident, a major fraction of the water drained from the ice condenser, and no increase in Containment pressure was indicated even for the severe case with no drains.

A number of tests were performed with the reference flow proportional-type door installed at the inlet to the ice condenser and a representative hinged door installed at the top of the condenser. Tests were conducted with and without the reference water drain area, equivalent to 15 ft^2 for the plant, at the bottom of the condenser compartment.

d. Does withstand accident transient and environmental conditions.

e. Is missile protected.

Missile protection for isolation valves is the same as that provided for contain-
1 ment penetrations. Subdivision 3.9.2.8 describes penetration design criteria.
Penetrations and their isolation valves are located in areas which receive design considerations with respect to missiles. In addition to design criteria and physical considerations with respect to missiles, valve specifications speak directly to the requirement of post-accident and environmental operation which must be substantiated by the valve supplier by actual test results.

Main steam line capability to withstand the dynamic forces of inadvertant isolation valve closure is described in Subdivision 3.9.1.1.

Isolation valves and valve operators are selected with the same consideration for safety that accompanies the selection of all ANS Safety Class 2 equipment.

6.2.4.3 Safety Evaluation

Based on the criteria set forth in the design basis, leakage through all piping penetrations is minimized during accident conditions. By using double automatic barriers, isolation is assured upon actuation of the engineered safety systems or upon a high radiation signal. All postulated conditions of adverse environment do not alter the proper operation of these valves. Selection of valves of the proper quality plus periodic testing of the valve operability and leak tightness assures a reliable isolation system under all conceivable conditions.

All manually operated isolation valves have permanent tags attached stating that the valve is to remain closed during unit operation. The operating proce-
9 dure controlling unit startup includes requirements to verify that these valves are closed prior to unit operation.

No system is provided to continuously monitor the leak-tightness of the Containment. However, the fact that redundant isolation is provided in all cases for valves required to operate following an incident, the administrative procedures concerning locked closed valves, and the testing procedures outlined in Techni-
9 cal Specifications provide confidence that the Containment Isolation System will perform its intended function.

6.2.4.4 Tests and Inspections

Periodic testing of valves designed to effect containment isolation during accident conditions is performed to determine their leak-tightness. Test connections, test vents, drain connections and manual isolation valves are provided to permit pressurization from the containment side of the valves
9 with the following exceptions:

a. Any globe or relief valve may be tested with pressure under the seat.

b. Butterfly valves are tested in the direction identified in Table 6.2.4-2.
14 The butterfly valves to be tested in the reverse direction have a seat construction designed for sealing against pressure on either side as

shown in Figure 6.2.4-2. Those butterfly valves are leak tested in both
14 directions by the manufacturer to demonstrate equivalent leakage from
either direction. Final certification provided by the manufacturer
35 indicated that the value seat leakage tests were performed with acceptable
results when tested from either direction. Leak rate tests in the reverse
direction are considered acceptable by the ASME Code, Section XI, IWV-3420,
since the test pressure is less than 15 psi.

14 c. Double disk gate valves may be tested by pressurizing between the seats.

35 d. Diaphragm valves may be tested by pressurizing from either direction. Equivalent leakage rates were verified by testing (See Table 6.2.4-4).

Table 6.2.4-2 lists those valves subject to Type C leakrate testing per
9 Appendix J to 10CFR50. Also listed is the direction of test pressurization
relative to post-accident containment pressure; however, this may be modified, consistent with the above exceptions.

Table 6.2.4-3 lists the penetrations not subject to Type C leakrate testing. These penetrations are designed either as an open flow path through the con-
20 tainment vessel during post accident conditions or do not communicate with
the containment and are therefore not required for leaktightness. Also, with the exception of the four main feedwater penetrations, these penetrations all connect to closed, seismic systems outside containment.

In addition to those listed in Table 6.2.4-3, the following penetrations are not subject to Type C leakrate testing:

a. Main steam penetrations - main steam line isolation is provided for the purpose of preventing reverse steam flow following a steam line rupture. Containment atmosphere leak protection is provided by the seismic design of the steam generators and main steam piping within the containment.

b. Fuel transfer tube-testing of this penetration is not required when the spent fuel pool is flooded since the pool water supplies a back pressure Q 042.60
9 greater than maximum containment pressure.

c. Containment pressure sensing lines - these penetrations must remain open at all times to provide monitoring of containment pressure.

d. Residual heat removal normal letdown line - this penetration connects directly to an active ECCS subsystem.

e. Process instrument impulse line penetrations.

Containment isolation valves located in lines where service can be interrupted are exercised as described in subdivision 7.3.2.2. Containment isolation valves which cannot be exercised during normal operation are tested during shutdown periods. These valves are specified in the Technical Specifications.

6.2.4.5 Instrumentation Application

Refer to Section 7.3 for a discussion of instrumentation employed for the actuation of the Containment Isolation System.

2

Containment atmosphere is circulated by natural convection through a recombiner where hydrogen is removed by heating to a temperature sufficient to cause recombination with the Containment oxygen.

The recombiner consists of a thermally insulated vertical metal duct with electric resistance metal sheathed heaters provided to heat a continuous flow of Containment air (containing a low concentration of hydrogen), up to a temperature which is sufficient to cause a reaction between hydrogen and oxygen. The recombiner is provided with an outer enclosure to keep out water coming from the Containment Spray System. The recombiner consists of an inlet preheater section, a heater-recombination section and a mixing chamber.

The unit is manufactured primarily of 300 series stainless steel or other corrosion resistant, high temperature material for major structural components, except for the base which is carbon steel. The electric hydrogen recombiner uses conventional type electric resistance heaters sheathed with Incoloy-800 which is an excellent corrosion resistant material for this service. These heaters are designed to operate with sheath temperatures equal to those used in certain commercial heaters; however, these recombiner heaters are designed to operate at significantly lower power densities than is commercial practice.

Air is drawn into the recombiner by natural convection and passes first through the preheater section. This section consists of a shroud placed around the central heater section to take advantage of heat conduction through the walls to preheat the incoming air. This accomplishes the dual functions of reducing heat losses from the recombiner and of preheating the air.

The warmed air passes through an orifice plate and then enters the electric heater section where it is heated to approximately 1150-1400 F causing recombination to occur. Tests have verified that the recombination is not a catalytic surface effect associated with the heaters but occurs due to the increased temperature of the process gases. Since the phenomenon is not a catalytic effect, poisoning of the unit as by fission products will not occur. The heater section consists of five assemblies of electric heaters stacked vertically. Each assembly contains individual heating elements. Operation of the unit is virtually unaffected in the event of a few individual heating elements failing to function properly.

Table 6.2.5-1 gives the recombiner design parameters.

The power panel for the recombiner is located outside the Reactor Building in an area accessible after a Loss of Coolant Accident. The panel contains an isolation transformer plus a controller to regulate power into the recombiner. This equipment is not exposed to the Post-Loss of Coolant Accident environment.
The control panel for the power source will be located in the Electrical Pene-
31 tration Room. For equipment test and periodic checkout, a thermocouple readout
instrument is also provided on the control panel for monitoring temperatures
in the recombiner. To control the recombination process, the correct power
input which will bring the recombiner above the threshold temperature for
35 recombination will be set on the controller. Setting of the controller is
accomplished at the local control panel and power input monitored by a
31 wattmeter. This predetermined power setting will cover variations in Con-
tainment pressure and hydrogen concentration in the Post-Loss of Coolant
Accident environment.

15 | Table 6.2.4-1 (Sheet 1 of 11)

Containment Piping Penetrations & Isolation Valve Data

Item Number	Penetration	(2) System	Pen. No.	Thru(3) Line Leak Class	(5) Pen Class	Line Size	Valve Location	(4) Valve Type	Valve Operator	(1) Actuation Signal	Valve Position Normal	Valve Position Fail	Valve Position Incident	FSAR Figure No.
1	Pressurizer Relief Tank Makeup	NC	M216	2	B1	3"	Inside	Check	---	---	---	---	---	5.1-2
							Outside	Gate	Motor	T	Open	As Is	Closed	5.1-2
2	Nitrogen to Pressurizer Relief Tank	NC	M212	2	A1	1"	Inside	Globe	Motor	T	Open	As Is	Closed	5.1-2
							Outside	Globe	Motor	T	Open	As Is	Closed	5.1-2
3	Containment Spray, Residual Heat Removal, Safety Injection and CVCS Relief to Pressurizer Relief Tank	NC	M274	1A	B3	4"	Inside	Check	---	---	---	---	---	5.1-2
		NS				1½"	Outside	Relief	---	---	---	---	---	6.5.2-1
		NS				1½"	Outside	Relief	---	---	---	---	---	6.5.2-1
		ND				1½"	Outside	Relief	---	---	---	---	---	5.5.7-1
		ND				1½"	Outside	Relief	---	---	---	---	---	5.5.7-1
		ND				1"	Outside	Relief	---	---	---	---	---	5.5.7-1
		NI				1"	Outside	Relief	---	---	---	---	---	6.3.2-3
		NI				1"	Outside	Relief	---	---	---	---	---	6.3.2-3
		NI				1"	Outside	Relief	---	---	---	---	---	6.3.2-3
		NI				1"	Outside	Relief	---	---	---	---	---	6.3.2-3
		NV				1½"	Outside	Relief	---	---	---	---	---	9.3.4-3
4	Reactor Coolant Pump Seal Water Supply	NV	M339	1A	D6	2"	Inside	Check	---	---	---	---	---	9.3.4-1
5	Reactor Coolant Pump Seal Water Supply	NV	M343	1A	D6	2"	Inside	Check	---	---	---	---	---	9.3.4-1
6	Reactor Coolant Pump Seal Water Supply	NV	M344	1A	D6	2"	Inside	Check	---	---	---	---	---	9.3.4-1
7	Reactor Coolant Pump Seal Water Supply	NV	M350	1A	D6	2"	Inside	Check	---	---	---	---	---	9.3.4-1
8	Reactor Coolant Pump Seal Water Return	NV	M256	1A	A7	4"	Inside	Gate	Motor	T	Open	As Is	Closed	9.3.4-1
						3/4"	Inside	Check	---	---	---	---	---	9.3.4-1
						4"	Outside	Gate	Motor	T	Open	As Is	Closed	9.3.4-1
9	Charging Line	NV	M329	1A	B1	3"	Inside	Check	---	---	---	---	---	9.3.4-1
						3"	Outside	Gate	Motor	S	Open	As Is	Closed	9.3.4-3
10	Letdown Line	NV	M347	1A	A6	2"	Inside	Globe	Diaphragm	T	Open	Closed	Closed	9.3.4-1
						2"	Inside	Globe	Diaphragm	T	Closed	Closed	Closed	9.3.4-1
						2"	Inside	Globe	Diaphragm	T	Closed	Closed	Closed	9.3.4-1
						2"	Inside	Relief	---	---	---	---	---	9.3.4-1
						3"	Outside	Globe	Motor	T	Open	As Is	Closed	9.3.4-1

Table 6.2.4-1 (Sheet 2 of 11)

Containment Piping Penetrations & Isolation Valve Data

Item Number	Penetration	(2) System	Pen. No.	Thru Line Leak Class (3)	(5) Pen. Class	Line Size	Valve Location	(4) Valve Type	Valve Operator	(1) Actuation Signal	Valve Position Normal	Valve Position Fail	Valve Position Incident	FSAR Figure No.
11	Reactor Makeup Water Tank to NV System	NB	M259	2	B1	1"	Inside	Check	---	---	---	---	---	9.3.6-3
						1"	Outside	Globe	Motor	T	Closed	As Is	Closed	9.3.6-3
12	Ice Condenser Ice Blowing Air In	NF	M394	2	C3	5"	Inside	Weld Cap	---	---	---	---	---	---
						5"	Outside	Weld Cap	---	---	---	---	---	---
13	Ice Condenser Ice Blowing Air Out	NF	M395	2	C3	6"	Inside	Weld Cap	---	---	---	---	---	---
						6"	Outside	Weld Cap	---	---	---	---	---	---
14	Ice Condenser Glycol In	NF	M373	2	B1	4"	Inside	Check	---	---	---	---	---	---
						4"	Outside	Diaphragm	Diaphragm	T	Open	As Is	Closed	---
15	Ice Condenser Glycol Out	NF	M372	2	A7	4"	Inside	Gate	Motor	T	Open	As Is	Closed	---
						4"	Outside	Gate	Diaphragm	T	Open	As Is	Closed	---
16	Steam Generator Blowdown	BB	M30[illegible]	1B	A1	2"	Inside	Globe	Motor	T	Open	As Is	Closed	10.4.8-1
						2"	Outside	Globe	Motor	T	Open	As Is	Closed	10.4.8-1
17	Steam Generator Blowdown	BB	M301	1B	A1	2"	Inside	Globe	Motor	T	Open	As Is	Closed	10.4.8-1
						2"	Outside	Globe	Motor	T	Open	As Is	Closed	10.4.8-1
18	Steam Generator Blowdown	BB	M304	1B	A1	2"	Inside	Globe	Motor	T	Open	As Is	Closed	10.4.8-1
						2"	Outside	Globe	Motor	T	Open	As Is	Closed	10.4.8-1
19	Steam Generator Blowdown	BB	M303	1B	A1	2"	Inside	Globe	Motor	T	Open	As Is	Closed	10.4.8-1
						2"	Outside	Globe	Motor	T	Open	As Is	Closed	10.4.8-1
20	RHR Out From Loops	ND	M314	1A	D5	14"	Inside	Gate	Motor	Remote Manual	Closed	As Is	Open/Closed	5.5.7-1
						4"	Inside	Relief	---	---	---	---	---	5.5.7-1
21	Boron Injection	NI	M351	1A	B4	3"	Inside	Check	---	---	---	---	---	6.3.2-1
						3/4"	Inside	Globe	Diaphragm	Manual	Closed	Closed	---	6.3.2-1
						4"	Outside	Gate	Motor	S	Closed	As Is	Open	6.3.2-1
						4"	Outside	Gate	Motor	S	Closed	As Is	Open	6.3.2-1
						1"	Outside	Globe	Manual	Locked Closed	Closed	As Is	---	6.3.2-1
22	Nitrogen to Accumulators	NI	M330	2	B1	1"	Inside	Check	---	---	---	---	---	6.3.2-2
						1"	Outside	Globe	Motor	T	Closed	As Is	Closed	6.3.2-2

Table 6.2.4-1 (Sheet 3 of 1[illegible])

Containment Piping Penetrations & Isolation Valve Data

Item Number	Penetration	(2) System	Pen. No.	Thru Line Leak Class (3)	(5) Pen. Class	Line Size	Valve Location	(4) Valve Type	Valve Operator	(1) Actuation Signal	Valve Position Normal	Valve Position Fail	Valve Position Incident	FSAR Figure No.
23	Safety Injection Test Line	NI	M321	2	A1	3/4"	Inside	Globe	Motor	T	Closed	As Is	Closed	6.3.2-2
						3/4"	Outside	Globe	Motor	T	Closed	As Is	Closed	6.3.2-2
						3/4"	Outside	Globe	Motor	T	Closed	As Is	Closed	6.3.2-3
24	Safety Injection Pump To Hot Leg	NI	M316	1A	B5	1"	Inside	Check	---	---	---	---	---	6.3.2-3
						2"	Inside	Check	---	---	---	---	---	6.3.2-3
						2"	Inside	Check	---	---	---	---	---	6.3.2-3
						4"	Outside	Gate	Motor	Remote Manual	Closed	As Is	Open/Closed	6.3.2-3
						3/4"	Inside	Globe	Motor	T	Closed	As Is	Closed	6.3.2-3
25	Safety Injection Pump To Hot Leg	NI	M319	1A	B5	2"	Inside	Check	---	---	---	---	---	6.3.2-3
						2"	Inside	Check	---	---	---	---	---	6.3.2-3
						4"	Outside	Gate	Motor	Remote Manual	Closed	As Is	Open/Closed	6.3.2-3
						3/4"	Inside	Globe	Diaphragm	Manual	Closed	Closed	---	6.3.2-3
26	Safety Injection Pump To Cold Leg	NI	M352	1A	B5	2"	Inside	Check	---	---	---	---	---	6.3.2-3
						2"	Inside	Check	---	---	---	---	---	6.3.2-3
						2"	Inside	Check	---	---	---	---	---	6.3.2-3
						2"	Inside	Check	---	---	---	---	---	6.3.2-3
						4"	Outside	Gate	Motor	Remote Manual	Open	As Is	Open	6.3.2-3
						1"	Inside	Globe	Diaphragm	Manual	Closed	Closed	---	6.3.2-3
27	RHR Pump To Cold Leg	ND	M336	1A	C5	6"	Inside	Check	---	---	---	---	---	6.3.2-3
						6"	Inside	Check	---	---	---	---	---	6.3.2-3
						1"	Inside	Globe	Diaphragm	Manual	Closed	Closed	---	6.3.2-3
						8"	Outside	Gate	Motor	Remote Manual	Open	As Is	Open	6.3.2-3
28	RHR Pump To Cold Leg	ND	M306	1A	B5	6"	Inside	Check	---	---	---	---	---	6.3.2-3
						6"	Inside	Check	---	---	---	---	---	6.3.2-3
						1"	Inside	Globe	Diaphragm	Manual	Closed	Closed	---	6.3.2-3
						8"	Outside	Gate	Motor	Remote Manual	Open	As Is	Open	6.3.2-3
29	RHR Pump To Hot Leg	ND	M277	1A	C5	8"	Inside	Check	---	---	---	---	---	6.3.2-3
						8"	Inside	Check	---	---	---	---	---	6.3.2-3
						12"	Outside	Gate	Motor	Remote/Manual	Closed	As Is	Open/Closed	6.3.2-3
						1"	Inside	Globe	Diaphragm	Manual	Closed	As Is	---	6.3.2-3
30	RHR Out From Sump	NI	M278	1A	C1	18"	Outside	Gate	Motor	Remote Manual	Closed	As Is	Open/Closed	6.3.2-3
31	RHR Out From Sump	NI	M302	1A	C1	18"	Outside	Gate	Motor	Remote Manual	Closed	As Is	Open/Closed	6.3.2-3
32	Upper Head Injection	NI	M334	1A	B1	12"	Inside	Check	---	---	---	---	---	6.3.2-4
						12"	Outside	Gate	Piston	Remote Manual	Open	As Is	Open/Closed	6.3.2-4
33	Upper Head Injection	NI	M349	1A	B1	12"	Inside	Check	---	---	---	---	---	6.3.2-4
						12"	Outside	Gate	Piston	Remote Manual	Open	As Is	Open/Closed	6.3.2-4
						*2"	Outside	Globe	Motor	T	Closed	As Is	Closed	6.3.2-4
						*3/4"	Outside	Globe	Motor	T	Closed	As Is	Closed	6.3.2-4

*These isolation valves are common to penetrations M334 and M349

15 | Table 6.2.4-1 (Sheet 4 of 1[illegible])

Containment Piping Penetrations & Isolation Valve Data

	Item Number	Penetration	(2) System	Pen. No.	Thru(3) Line Leak Class	(5) Pen. Class	Line Size	Valve Location	(4) Valve Type	Valve Operator	(1) Actuation Signal	Valve Position Normal	Valve Position Fail	Valve Position Incident	FSAR Figure No.
	34	Upper Head Injection Test Line	NI	M348	2	A6	2" 2" 2" 2"	Inside Inside Outside Inside	Globe Globe Globe Relief	Motor Motor Motor Spring	T T T ---	Closed Closed Closed Closed	As Is As Is As Is Auto	Closed Closed Closed Auto	6.3.2-4 6.3.2-4 6.3.2-4 6.3.2-4
	35	Containment Spray In	NS	M362	1A	B1	8" 8"	Inside Outside	Check Gate	--- Motor	--- P	--- Closed	--- As Is	--- Open	6.5.2-1 6.5.2-1
	36	Containment Spray In	NS	M370	1A	B1	8" 8"	Inside Outside	Check Gate	--- Motor	--- P	--- Closed	--- As Is	--- Open	6.5.2-1 6.5.2-1
8	37	Containment Spray In	NS	M380	1A	B1	8" 8"	Inside Outside	Check Gate	--- Motor	--- P	--- Closed	--- As Is	--- Open	6.5.2-1 6.5.2-1
	38	Containment Spray In	NS	M387	1A	B1	8" 8"	Inside Outside	Check Gate	--- Motor	--- P	--- Closed	--- As Is	--- Open	6.5.2-1 6.5.2-1
	39	RHR To Containment Spray	NS	M369	1A	B1	8" 8"	Inside Outside	Check Gate	--- Motor	--- Remote Manual	--- Closed	--- As Is	--- Open/Closed	6.5.2-1 6.5.2-1
	40	RHR To Containment Spray	NS	M381	1A	B1	8" 8"	Inside Outside	Check Gate	--- Motor	--- Remote Manual	--- Closed	--- As Is	--- Open/Closed	6.5.2-1 6.5.2-1
5	41	Containment Floor Sump Incore Inst. Sump Discharge	WL	M374	2	A6	4" 4" 3/4"	Inside Outside Inside	Diaphragm Diaphragm Relief	Motor Motor ---	T T ---	Open Open Closed	As Is As Is Auto	Closed Closed Auto	11.2.2-1 11.2.2-1 11.2.2-1
	42	Reactor Coolant Drain Tank Gas Space to Waste Gas System	WL	M360	2	A1	3/4" 3/4"	Inside Outside	Diaphragm Diaphragm	Motor Motor	T T	Open Open	As Is As Is	Closed Closed	11.2.2-2 11.2.2-2
20	43	Reactor Coolant Drain Tank Heat Exchanger Discharge	WL	M375	2	A7	3" 1/2" 3"	Inside Inside Outside	Diaphragm Check Diaphragm	Motor --- Motor	T --- T	Open --- Open	As Is --- As Is	Closed --- Closed	11.2.2-2 11.2.2-2 11.2.2-2
	44	Equipment Decontamination	WE	M356	2	A4	1½" 1½"	Inside Outside	Globe Globe	Manual Manual	Locked Closed Locked Closed	Closed Closed	As Is As Is	Closed Closed	11.2.2-9 11.2.2-9
1	45	Pressurizer Sample	NM	M235	2	A6	3/4" 3/4" 3/4" 3/4"	Inside Inside Inside Outside	Globe Globe Relief Globe	Motor Motor Spring Motor	T T -- T	Closed Closed Closed Closed	As Is As Is Auto As Is	Closed Closed Auto Closed	9.3.2-1 9.3.2-1 9.3.2-1 9.3.2-1

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Table 6.2.4-1 (Sheet 5 of 1[illegible])

Containment Piping Penetrations & Isolation Valve Data

Item Number	Penetration	(2) System	Pen. No.	Thru(3) Line Leak Class	(5) Pen. Class	Line Size	Valve Location	(4) Valve Type	Valve Operator	(1) Actuation Signal	Valve Position Normal	Valve Position Fail	Valve Position Incident	FSAR Figure No.
46	Reactor Coolant Hot Leg Sample	NM	M309	2	A6	3/4"	Inside	Globe	Motor	T	Closed	As is	Closed	9.3.2-1
						3/4"	Inside	Globe	Motor	T	Closed	As is	Closed	9.3.2-1
						3/4"	Inside	Relief	Spring	---	Closed	Auto	Auto	9.3.2-1
						3/4"	Outside	Globe	Motor	T	Closed	As is	Closed	9.3.2-1
47	Safety Injection Sample	NM	M280	1B	A6	3/4"	Inside	Globe	Motor	T	Closed	As is	Closed	9.3.2-1
						3/4"	Inside	Globe	Motor	T	Closed	As is	Closed	9.3.2-1
						3/4"	Inside	Globe	Motor	T	Closed	As is	Closed	9.3.2-1
						3/4"	Inside	Globe	Motor	T	Closed	As is	Closed	9.3.2-1
						3/4"	Inside	Relief	Spring	---	Closed	Auto	Auto	9.3.2-1
						3/4"	Outside	Globe	Motor	T	Closed	As is	Closed	9.3.2-1
48	Steam Generator Blowdown Sample	NM	M335	1B	A6	3/4"	Inside	Globe	Motor	T	Closed	As is	Closed	9.3.2-3
						3/4"	Inside	Globe	Motor	T	Closed	As is	Closed	9.3.2-3
						3/4"	Inside	Relief	Spring	---	Closed	Auto	Auto	9.3.2-3
						3/4"	Outside	Globe	Motor	T	Closed	As is	Closed	9.3.2-3
49	Steam Generator Blowdown Sample	NM	M338	1B	A6	3/4"	Inside	Globe	Motor	T	Closed	As is	Closed	9.3.2-3
						3/4"	Inside	Globe	Motor	T	Closed	As is	Closed	9.3.2-3
						3/4"	Inside	Relief	Spring	---	Closed	Auto	Auto	9.3.2-3
						3/4"	Outside	Globe	Motor	T	Closed	As is	Closed	9.3.2-3
50	Steam Generator Blowdown Sample	NM	M340	1B	A6	3/4"	Inside	Globe	Motor	T	Closed	As is	Closed	9.3.2-3
						3/4"	Inside	Globe	Motor	T	Closed	As is	Closed	9.3.2-3
						3/4"	Inside	Relief	Spring	---	Closed	Auto	Auto	9.3.2-3
						3/4"	Outside	Globe	Motor	T	Closed	As is	Closed	9.3.2-3
51	Steam Generator Blowdown Sample	NM	M341	1B	A6	3/4"	Inside	Globe	Motor	T	Closed	As is	Closed	9.3.2-3
						3/4"	Inside	Globe	Motor	T	Closed	As is	Closed	9.3.2-3
						3/4"	Inside	Relief	Spring	---	Closed	Auto	Auto	9.3.2-3
						3/4"	Outside	Globe	Motor	T	Closed	As is	Closed	9.3.2-3
52	Component Cooling To Reactor Coolant Drain Tank HX	KC	M376	2	B1	3"	Inside	Check	---	---	---	---	---	[illegible]
						3"	Outside	Diaphragm	Motor	T	Open	As is	Closed	[illegible]
53	Component Cooling From Reactor Coolant Drain Tank HX	KC	M355	2	A7	3"	Inside	Diaphragm	Motor	T	Open	As is	Closed	[illegible]
						1"	Inside	Check	---	---	---	---	---	[illegible]
						3"	Outside	Diaphragm	Motor	T	Open	As is	Closed	[illegible]
54	Component Cooling To Reactor Vessel Support Coolers & RCP Coolers	KC	M327	2	B1	8"	Inside	Check	---	---	---	---	---	[illegible]-3
						8"	Outside	Diaphragm	Motor	P	Open	As is	Closed	[illegible]

Table 6.2.4-1 (Sheet 6 of 11)

Containment Piping Penetrations & Isolation Valve Data

	Item Number	Penetration	(2) System	Pen. No.	Thru(3) Line Leak Class	(5) Pen. Class	Line Size	Valve Location	(4) Valve Type	Valve Operator	(1) Actuation Signal	Valve Position			FSAR Figure No.
												Normal	Fail	Incident	
35	55	Component Cooling From Reactor Vessel Support Coolers & RCP Coolers	KC	M320	2	A7	8"	Inside	Diaphragm	Motor	P	Open	As Is	Closed	9.2.4-3
8							1"	Inside	Check	---	---	---	---	---	9.2.4-3
35							8"	Outside	Diaphragm	Motor	P	Open	As Is	Closed	9.2.4-3
	56	Component Cooling To Excess Letdown HX	KC	M218	1B	D2	3"	Outside	Gate	Motor	T	Closed	As Is	Closed	9.2.4-3
30															
	57	Component Cooling From Excess Letdown HX	KC	M217	1B	D2	3"	Outside	Gate	Motor	T	Closed	As Is	Closed	9.2.4-3
5	58	Component Cooling To Component Cooling Drain Tank	KC	M322	2	A7	2"	Inside	Globe	Motor	T	Closed	As Is	Closed	9.2.4-4
							3/4"	Inside	Check	---	---	----	---	---	9.2.4-4
							2"	Outside	Globe	Motor	T	Closed	As Is	Closed	9.2.4-4
	59	Nuclear Service Water To Reactor Coolant Pump	RN	M307	2	A7	6"	Inside	Diaphragm	Motor	P	Open	As Is	Closed	9.2.2-4
							6"	Outside	Diaphragm	Motor	P	Open	As Is	Closed	9.2.2-4
35															
	60	Nuclear Service Water From Reactor Coolant Pump	RN	M315	2	A7	6"	Inside	Diaphragm	Motor	P	Open	As Is	Closed	9.2.2-4
							6"	Outside	Diaphragm	Motor	P	Open	As Is	Closed	9.2.2-4
	61	Incore Instrumentation Room Purge In	VP	M213	2	A5	12"	Inside	Butterfly	Diaphragm	T	Closed	Closed	Closed	9.4.5-1
								Outside	Butterfly	Diaphragm	T	Closed	Closed	Closed	9.4.5-1
5															
	62	Incore Instrumentation Room Purge Out	VP	M138	2	A5	24"	Inside	Butterfly	Diaphragm	T	Closed	Closed	Closed	9.4.5-1
								Outside	Butterfly	Diaphragm	T	Closed	Closed	Closed	9.4.5-1
	63	Upper Compartment Purge Inlet	VP	M367	2	A5	24"	Inside	Butterfly	Diaphragm	T	Closed	Closed	Closed	9.4.5-1
								Outside	Butterfly	Diaphragm	T	Closed	Closed	Closed	9.4.5-1
	64	Upper Compartment Purge Inlet	VP	M454	2	A5	24"	Inside	Butterfly	Diaphragm	T	Closed	Closed	Closed	9.4.5-1
20								Outside	Butterfly	Diaphragm	T	Closed	Closed	Closed	9.4.5-1
	65	Lower Compartment Purge Inlet	VP	M357	2	A5	24"	Inside	Butterfly	Diaphragm	T	Closed	Closed	Closed	9.4.5-1
								Outside	Butterfly	Diaphragm	T	Locked Closed	Closed	Closed	9.4.5-1
	66	Lower Compartment Purge Inlet	VP	M456	2	A5	24"	Inside	Butterfly	Diaphragm	T	Closed	Closed	Closed	9.4.5-1
20								Outside	Butterfly	Diaphragm	T	Locked Closed	Closed	Closed	9.4.5-1
	67	Containment Purge Exhaust	VP	M368	2	A5	24"	Inside	Butterfly	Diaphragm	T	Closed	Closed	Closed	9.4.5-1
								Outside	Butterfly	Diaphragm	T	Closed	Closed	Closed	9.4.5-1

Table [illegible] 1 (Sheet 7 of 11)

Containment Piping Penetrations & Isolation Valve Data

	Item Number	Penetration	(2) System	Pen No.	Thru (3) Line Leak Class	(5) Pen. Class	Line Size	Valve Location	(4) Valve Type	Valve Operator	(1) Actuation Signal	Valve Position Normal	Valve Position Fail	Valve Position Incident	FSAR Figure No.	
	68	Containment Purge Exhaust	VP	M455	2	A5	24"	Inside	Butterfly	Diaphragm	T	Closed	Closed	Closed	9.4.5-1	
20							24"	Outside	Butterfly	Diaphragm	T	Closed	Closed	Closed	9.4.5-1	Q 6.17
	69	Containment Purge Exhaust	VP	M119	2	A5	24"	Inside	Butterfly	Diaphragm	T	Closed	Closed	Closed	9.4.5-1	
							24"	Outside	Butterfly	Diaphragm	T	Closed	Closed	Closed	9.4.5-1	
	70	Feedwater	CF	M440	1B	D1	18"	Outside	Gate	Piston	S	Open	As Is	Closed	10.4.7-3	
							1"	Outside	Globe	Manual	---	Closed	As Is	Closed	10.4.7-3	
							2"	Outside	Globe	Motor	S	Closed	As Is	Closed		
	71	Feedwater	CF	M308	1B	D1	18"	Outside	Gate	Pitson	S	Open	As Is	Closed	10.4.7-3	
							1"	Outside	Globe	Manual	---	Closed	As Is	Closed	10.4.7-3	
20							2"	Outside	Globe	Motor	S	Closed	As Is	Closed		
	72	Feedwater	CF	M262	1B	D1	18"	Outside	Gate	Piston	S	Open	As Is	Closed	10.4.7-3	
							1"	Outside	Globe	Manual	---	Closed	As Is	Closed	10.4.7-3	
							2"	Outside	Globe	Motor	S	Closed	As Is	Closed		
	73	Feedwater	CF	M153	1B	D1	18"	Outside	Gate	Piston	S	Open	As Is	Closed	10.4.7-3	
							1"	Outside	Globe	Manual	---	Closed	As Is	Closed	10.4.7-3	
							2"	Outside	Globe	Motor	S	Closed	As Is	Closed		
	74	Main Steam	SM	M441	1B	D3	34"	Outside	Gate	Piston	P	Open	Closed	Closed	10.3.2-1	
							6"	Outside	Safety	Spring	---	Closed	Auto	Auto	10.3.2-1	
8							6"	Outside	Safety	Spring	---	Closed	Auto	Auto	10.3.2-1	
							6"	Outside	Safety	Spring	---	Closed	Auto	Auto	10.3.2-1	
							6"	Outside	Safety	Spring	---	Closed	Auto	Auto	10.3.2-1	
							6"	Outside	Safety	Spring	---	Closed	Auto	Auto	10.3.2-1	
							6"	Outside	Gate	Diaphragm	---	Closed	Closed	Closed	10.3.2-1	
20							3"	Outside	Gate	Diaphragm	P	Closed	Closed	Closed	10.3.2-1	
							2"	Outside	Globe	Diaphragm		Open	Open	Open	10.3.2-1	
	75	Main Steam	SM	M393	1B	D3	34"	Outside	Gate	Piston	P	Open	Closed	Closed	10.3.7-1	
							6"	Outside	Safety	Spring	---	Closed	Auto	Auto	10.3.2-1	
							6"	Outside	Safety	Spring	---	Closed	Auto	Auto	10.3.2-1	
							6"	Outside	Safety	Spring	---	Closed	Auto	Auto	10.3.2-1	
							6"	Outside	Safety	Spring	---	Closed	Auto	Auto	10.3.7-1	
							6"	Outside	Safety	Spring	---	Closed	Auto	Auto	10.3.2-1	
							6"	Outside	Gate	Diaphragm	---	Closed	Closed	Closed	10.3.2-1	
							3"	Outside	Gate	Diaphragm	P	Closed	Closed	Closed	10.3.2-1	
[illegible]							2"	Outside	Globe	Diaphragm		Open	Open	Open	10.3.2-1	
	76	Main Steam	S[illegible]	[illegible]	1B	D3	3[illegible]	Outside	Gate	Piston	P	Open	Closed	Closed	[illegible]	
							6"	Outside	Safety	Spring	---	Closed	Auto	Auto	[illegible]	
							6"	Outside	Safety	Spring	---	Closed	Auto	Auto	[illegible]	
							6"	Outside	Safety	Spring	---	Closed	Auto	Auto	10.3.2-1	
							6"	Outside	Safety	Spring	---	Closed	Auto	Auto	10.3.7-1	
							[illegible]	Outside	Safety	Spring	---	Closed	Auto	Auto	10.3.2-1	
							6"	Outside	Gate	Diaphragm	---	Closed	Closed	Closed	10.3.[illegible]-1	
							3"	Outside	Gate	Diaphragm	P	Closed	Closed	Closed	10.3.2-1	
20							2"	Outside	Globe	Diaphragm		Open	Open	Open	10.3.2-1	

Table 6.2.4-1 (Sheet 8 of 11)

Containment Piping Penetrations & Isolation Valve Data

Item Number	Penetration	(2) System	Pen. No.	Thru[3] Line Leak Class	(5) Pen. Class	Line Size	Valve Location	(4) Valve Type	Valve Operator	(1) Actuation Signal	Valve Position Normal	Valve Position Fail	Valve Position Incident	FSAR Figure No.
77	Main Steam	SM	M154	1B	D3	34"	Outside	Gate	Piston	P	Open	Closed	Closed	10.3.2-1
						6"	Outside	Safety	Spring	---	Closed	Auto	Auto	10.3.2-1
						6"	Outside	Safety	Spring	---	Closed	Auto	Auto	10.3.2-1
						6"	Outside	Safety	Spring	---	Closed	Auto	Auto	10.3.2-1
						6"	Outside	Safety	Spring	---	Closed	Auto	Auto	10.3.2-1
						6"	Outside	Safety	Spring	---	Closed	Auto	Auto	10.3.2-1
						6"	Outside	Gate	Diaphragm	---	Closed	Closed	Closed	10.3.2-1
						3"	Outside	Gate	Diaphragm	P	Closed	Closed	Closed	10.3.2-1
						2"	Outside	Globe	Diaphragm		Open	Open	Open	10.3.2-1
78	Demineralized Water	YM	M337	2	B1	2"	Inside	Check	---	---	---	---	---	9.2.6-8
						2"	Outside	Globe	Motor	T	Open	As Is	Closed	9.2.6-8
79	Containment Ventilation Cooling Water In	RV	M240	2	A7	12"	Inside	Butterfly	Motor	P	Open	As Is	Closed	10.4.5-3
						12"	Outside	Butterfly	Motor	F	Open	As Is	Closed	10.4.5-3
						3/4"	Inside	Check	---	---	---	---	---	10.4.5-3
80	Containment Ventilation Cooling Water In	RV	M385	2	A7	6"	Inside	Diaphragm	Motor	P	Open	As Is	Closed	10.4.5-3
						6"	Outside	Diaphragm	Motor	P	Open	As Is	Closed	10.4.5-3
81	Containment Ventilation Cooling Water Out	RV	M390	2	A7	6"	Inside	Diaphragm	Motor	P	Open	As Is	Closed	10.4.5-3
						6"	Outside	Diaphragm	Motor	P	Open	As Is	Closed	10.4.5-3
82	Containment Ventilation Cooling Water Out	RV	M279	2	A7	12"	Inside	Butterfly	Motor	P	Open	As Is	Closed	10.4.5-3
						12"	Outside	Butterfly	Motor	P	Open	As Is	Closed	10.4.5-3
						3/4"	Inside	Check	---	---	---	---		
83	Instrument Air	VI	M220	2	B1	2"	Inside	Check	---	---	---	---	---	9.3.1-2
						2"	Outside	Globe	Motor	T	Open	As Is	Closed	9.3.1-2
84	Station Air	VS	M219	2	B1	2"	Inside	Check	---	---	---	---	---	9.3.1-3
						2"	Outside	Globe	Motor	T	Open	As Is	Closed	9.3.1-3
85	Breathing Air	VB	M215	2	B1	2"	Inside	Check	---	---	---	---	---	
						2"	Outside	Gate	Motor	T	Open	As Is	Closed	
86	Containment Pressure Sensor		M118	1F	D4									
87			M247	1F	D4									
88			M402	1F	D4									
89			M313	1F	D4									
90	Containment Sample Out	VX	M378	2	A2	1"	Inside	Diaphragm	Diaphragm	T	Closed	Closed	Closed	6.6.2-1
						1"	Inside	Diaphragm	Diaphragm	T	Closed	Closed	Closed	6.6.2-1
						1"	Outside	Diaphragm	Manual	Locked Closed	Closed	Closed	Closed	6.6.2-1
91	Containment Sample In	VX	M325	2	B1	1"	Inside	Check	---	---	---	---	---	6.6.2-1
						1"	Outside	Diaphragm	Manual	Locked Closed	Closed	As Is	Closed	6.6.2-1

Table 6.2.4-1 (Sheet 9 of 11)

Containment Piping Penetrations & Isolation Valve Data

	Item Number	Penetration	(2) System	Pen. No.	Line Leak Class	Thru(3) (5) Pen. Class	Line Size	Valve Location	(4) Valve Type	Valve Operator	(1) Actuation Signal	Valve Position Normal	Valve Position Fail	Valve Position Incident	FSAR Figure No.
	92	Auxiliary Feedwater	CA	M156	1B	D1	4"	Outside	Gate	Motor	---	Open	As Is	Open	
					1B	D1	4"	Outside	Gate	Motor	---	Open	As Is	Open	
					1B	D1	6"	Outside	Gate	Motor	S	Closed	As Is	Closed	
	93	Auxiliary Feedwater	CA	M286	1B	D1	4"	Outside	Gate	Motor	---	Open	As Is	Open	
					1B	D1	4"	Outside	Gate	Motor	---	Open	As Is	Open	
					1B	D1	6"	Outside	Gate	Motor	S	Closed	As Is	Closed	
20	94	Auxiliary Feedwater	CA	M310	1B	D1	4"	Outside	Gate	Motor	---	Open	As Is	Open	
					1B	D1	4"	Outside	Gate	Motor	---	Open	As Is	Open	
					1B	D1	6"	Outside	Gate	Motor	S	Closed	As Is	Closed	
	95	Auxiliary Feedwater	CA	M465	1B	D1	4"	Outside	Gate	Motor	---	Open	As Is	Open	
					1B	D1	4"	Outside	Gate	Motor	---	Open	As Is	Open	
					1B	D1	6"	Outside	Gate	Motor	S	Closed	As Is	Closed	
	96	Refueling Cavity To RW Pump	FW	M358	2	A4	4"	Inside	Diaphragm	Manual	Locked Closed	Closed	As Is	Closed	9.2.5-1
							4"	Outside	Diaphragm	Manual	Locked	Closed	As Is	Closed	9.2.5-1
							3/4"	Inside	Check	---	---	---	---	---	9.2.5-1
35	97	Refueling Cavity From RW Tank	FW	M377	2	B2	6"	Inside	Check	---	---	---	---	---	9.2.5-1
							6"	Outside	Gate	Manual	Locked	Closed	---	---	9.2.5-1
35	98	Hydrogen Purge In	VE	M331	2	B1	4"	Inside	Check	---	---	---	---	---	6.2.3-1
							4"	Outside	Diaphragm	Motor	T	Closed	As Is	Closed/Open	6.2.3-1
30	99	Hydrogen Purge Out	VE	M346	1F	A5	4"	Inside	Gate	Motor	T	Closed	As Is	Closed/Open	6.2.3-1
							4"	Outside	Gate	Motor	T	Closed	As Is	Closed/Open	6.2.3-1
	100	Fuel Transfer Canal	KF		1E	C2	24"	Inside	Blind Flange	---	Bolted	Closed	As Is	Closed	
								Outside	Gate	Manual	Locked Closed	Closed	As Is	Closed	
35	101	RCP Motor Drain Tank Pump to Waste Oil Storage	NC	M326	2	A4	2"	Inside	Diaphragm	Manual	Locked Closed	Closed	As Is	Closed	---
							2"	Outside	Diaphragm	Manual	Locked Closed	Closed	As Is	Closed	---
20	102	Containment Ventilation Units Cond. Drains to Cond. Drain Tank	WL	M221	2	A7	6"	Inside	Butterfly	Motor	P	Open	As Is	Closed	11.2.2-6
							6"	Outside	Butterfly	Motor	P	Open	As Is	Closed	11.2.2-6
30							1"	Inside	Check	---	---	---	---	---	
	103	Instrument Air	VI	M359	2	B1	2"	Inside	Check	---	---	---	---	---	9.3.1-2
20							2"	Outside	Globe	Motor	T	Open	As Is	Closed	9.3.1-2
	104	Instrument Air	VI	M386	2	B1	2"	Inside	Check	---	---	---	---	---	9.3.1-1
							2"	Outside	Globe	Motor	T	Open	As Is	Closed	9.3.1-1
	105	Instrument Air	VI	M317	2	B1	2"	Inside	Check	---	---	---	---	---	9.3.1-1
							2"	Outside	Globe	Motor	T	Open	As Is	Closed	9.3.1-1

Table 6.2.4-1 (10 of 11)

Containment Piping Penetrations & Isolation Valve Data

Item Number	Penetration	(2) System	Pen. No.	Thru[3] Line Leak Class	(5) Pen. Class	Line Size	(4) Valve Location	Valve Type	(1) Valve Operator	Actuation Signal	Valve Position Normal	Valve Position Fail	Valve Position Incident	FSAR Figure No
106	Containment Air Release	VQ	M243	2	A1	6"	Inside	Diaphragm	Piston	T	Closed	Closed	Closed	9.5.12-1
							Outside	Diaphragm	Piston	T	Closed	Closed	Closed	9.5.12-1
107	Containment Air Addition	VQ	M384	2	B1	6"	Inside	Diaphragm	Piston	T	Closed	Closed	Closed	9.5.12-1
							Outside	Diaphragm	Piston	T	Closed	Closed	Closed	9.5.12-1
108	Reactor Coolant Pump Motor Oil Supply	NC	M361	2	A5	2"	Inside	Globe	Motor	T	Closed	As Is	Closed	
							Outside	Globe	Motor	T	Closed	As Is	Closed	
							Inside	Check	---	--	---	---	---	
109	Fire Protection Header	RF	M353	2	B2	4"	Inside	Check	---	--	---	---	---	9.5.1-2
						6"	Outside	Diaphragm	Manual	--	Locked Closed	Closed	Closed	9.5.1-2
110	Auxiliary Spray to Pressurizer	NV	M228	1A	B2	3"	Inside	Check	---	--	---	---	---	9.3.4-1
						2"	Outside	Globe	Diaphragm	--	Closed	Closed	Closed	9.3.4-1

Table 6.2.4-2

Containment Isolation Valves Subject
to Type C Leakrate Tests

	Penetration	FSAR Fig. No.	Valve Number	Valve Type	Line Size	Actuation Signal	Direction of Test Press. Relative to Cont. Press.
20	6 M216	5.1-2	NC57	Check	3"		Normal
		5.1-2	NC56B	Diaphragm	3"	T	Normal
	M212	5.1-2	NC54A	Globe	1"	T	Normal
		5.1-2	NC53B	Globe	1"	T	Normal
35	M326	---	NC141	Diaphragm	2"	L.C.	Reverse
		---	NC142	Diaphragm	2"	L.C.	Normal
	M330	6.3.2-2	NI48	Check	1"		Normal
		6.3.2-2	NI47A	Globe	1"	T	Normal
20	M321	6.3.2-2	NI55A	Globe	3/4"	T	Reverse
		6.3.2-2	NI96B	Globe	3/4"	T	Normal
		6.3.2-3	NI120B	Globe	3/4"	T	Normal
35			NI436	Check	1"		Normal
	M259	9.3.6-3	NB260B	Globe	1"	T	Normal
		9.3.6-3	NB262	Check	1"		Normal
20	M373	---	NF228A	Diaphragm	4"	T	Normal
		---	NF229	Check	4"		Normal
	M372	---	NF234A	Diaphragm	4"	T	Normal
35		---	NF233B	Diaphragm	4"	T	Reverse
	M348	6.3.2-4	NI266A	Globe	2"	T	Reverse
		6.3.2-4	NI267A	Globe	2"	T	Reverse
		6.3.2-4	NI264B	Globe	2"	T	Normal
20		6.3.2-4	NI336	Relief	2"		Reverse
	M374	11.2.2-1	WL65B	Diaphragm	4"	T	Normal
		11.2.2-1	WL64A	Diaphragm	4"	T	Reverse
35 20		11.2.2-1	WL264	Relief	3/4"		Reverse
35	M360	11.2.2-2	WL39A	Diaphragm	3/4"	T	Reverse
20		11.2.2-2	WL41B	Diaphragm	3/4"	T	Normal
35	M375	11.2.2-2	WL2A	Diaphragm	3"	T	Reverse
20		11.2.2-2	WL1B	Diaphragm	3"	T	Normal
		11.2.2-2	WL24	Check	1/2"		Normal

Containment Isolation Valves Subject
to Type C Leakrate Tests

	Penetration	FSAR Fig. No.	Valve Number	Valve Type	Line Size	Actuation Signal	Direction of Test Press. Relative to Cont. Press.
	M356	11.2.2-9	WE13	Globe	1 1/2"	L.C.	Normal
		11.2.2-9	WE23	Globe	1 1/2"	L.C.	Reverse
	M235	9.3.2-1	NM3A	Globe	3/4"	T	Normal
		9.3.2-1	NM6A	Globe	3/4"	T	Normal
		9.3.2-1	NM67	Relief	3/4"		Reverse
		9.3.2-1	NM7B	Globe	3/4"	T	Normal
	M309	9.3.2-1	NM26B	Globe	3/4"	T	Normal
		9.3.2-1	NM25A	Globe	3/4"	T	Normal
		9.3.2-1	NM22A	Globe	3/4"	T	Normal
20		9.3.2-1	NM68	Relief	3/4"		Reverse
	M280	9.3.2-1	NM72B	Globe	3/4"	T	Normal
		9.3.2-1	NM75B	Globe	3/4"	T	Normal
		9.3.2-1	NM78B	Globe	3/4"	T	Normal
		9.3.2-1	NM81B	Globe	3/4"	T	Normal
		9.3.2-1	NM69	Relief	3/4"		Reverse
		9.3.2-1	NM82A	Globe	3/4"	T	Normal
	M322	9.2.4-4	KC47	Check	3/4"		Normal
		9 2.4-4	KC429B	Globe	2"	T	Normal
		9.2.4-4	KC430A	Globe	2"	T	Normal
	M307	9.2.2-4	RN252B	Diaphragm	6"	P	Normal
		9.2.2-4	RN253A	Diaphragm	6"	P	Reverse
35							
	M315	9.2.2-4	RN277B	Diaphragm	6"	P	Normal
		9.2.2-4	RN276A	Diaphragm	6"	P	Reverse
	M213	9.4.5-1	VP17A	Butterfly	12"	T	Reverse
		9.4.5-1	VP18B	Butterfly	12"	T	Normal
	M138	9.4.5-1	VP19A	Butterfly	24"	T	Reverse
		9.4.5-1	VP20B	Butterfly	24"	T	Normal
20	M367	9.4.5-1	VP1B	Butterfly	24"	T	Normal
		9.4.5-1	VP2A	Butterfly	24"	T	Reverse
	M454	9.4.5-1	VP3B	Butterfly	24"	T	Normal
		9.4.5-1	VP4A	Butterfly	24"	T	Reverse
	M357	9.4.5-1	VP6B	Butterfly	24"	T	Normal
		9.4.5-1	VP7A	Butterfly	24"	T	Reverse

Table 6.2.4-2

Containment Isolation Valves Subject
to Type C Leakrate Tests

	Penetration	FSAR Fig. No.	Valve Number	Valve Type	Line Size	Actuation Signal	Direction of Test Press. Relative to Cont. Press.
	M456	9.4.5-1	VP8B	Butterfly	24"	T	Normal
		9.4.5-1	VP9A	Butterfly	24"	T	Reverse
	M368	9.4.5-1	VP10A	Butterfly	24"	T	Reverse
		9.4.5-1	VP11B	Butterfly	24"	T	Normal
20	M455	9.4.5-1	VP12A	Butterfly	24"	T	Reverse
		9.4.5-1	VP13B	Butterfly	24"	T	Normal
	M119	9.4.5.1	VP15A	Butterfly	24"	T	Reverse
		9.4.5.1	VP16B	Butterfly	24"	T	Normal
	M337	9.2.6-8	YM116	Check	2"		Normal
		9.2.6-8	YM115B	Globe	2"	T	Normal
35	M240	10.4.5-3	RV32A	Butterfly	12"	P	Normal
		10.4.5-3	RV33B	Butterfly	12"	P	Normal
20		10.4.5-3	RV130	Check	3/4"		Normal
	M390	10.4.5-3	RV101A	Diaphragm	6"	P	Reverse
		10.4.5-3	RV102B	Diaphragm	6"	P	Normal
	M385	10.4.5-3	RV79A	Diaphragm	6"	P	Normal
35		10.4.5-3	RV80B	Diaphragm	6"	P	Reverse
	M279	10.4.5-3	RV76A	Butterfly	12"	P	Normal
		10.4.5-3	RV77B	Butterfly	12"	P	Normal
		10.4.5-3	RV126	Check	3/4"		
20	M215		VB49B	Gate	2"	T	Normal
			VB50	Check	2"		Normal
35	M219	9.3.1-3	VS12B	Globe	2"	T	Normal
20		9.3.1-3	VS13	Check	2"	T	Normal
	M378	6.6.2-1	VX34	Diaphragm	1"	L.C.	Normal
35		6.6.2-1	VX33B	Diaphragm	1"	T	Reverse
		6.6.2-1	VX31A	Diaphragm	1"	T	Reverse
20	M325	6.6.2-1	VX30	Check	1"	L.C.	Normal
35		6.6.2-1	VX40	Diaphragm	1"		Reverse
20	M358	9.2.5-2	FW11	Diaphragm	4"	L.C.	Normal
		9.2.5-2	FW13	Diaphragm	4"	L.C.	Normal
35		9.2.5-2	FW67	Check	3/4"	--	Normal

Containment Isolation Valves Subject
to Type C Leakrate Tests

	Penetration	FSAR Fig. No.	Valve Number	Valve Type	Line Size	Actuation Signal	Direction of Test Press. Relative to Cont. Press.
	M377	9.2.5-1	FW4	Gate	6"	L.C.	Normal
		9.2.5-1	FW5	Check	6"		Normal
	M221	11.2.2-6	WL321A	Butterfly	6"	P	Normal
		11.2.2-6	WL322B	Butterfly	6"	P	Normal
		11.2.2-6	WL385	Check	½"		Normal
	M359	9.3.1-1	V1161	Check	2"		Normal
20		9.3.1-1	V1160B	Globe	2"	T	Normal
	M386	9.3.1-1	V1149	Check	2"		Normal
		9.3.1-1	V1148B	Globe	2"	T	Normal
	M317	9.3.1-1	V1124	Check	2"		Normal
		9.3.1-1	V1150B	Globe	2"	T	Normal
	M220	9.3.1-2	V1129B	Globe	2"	T	Normal
		9.3.1-2	V1140	Check	2"	T	Normal
35	M331	6.2.3-1	VE10A	Diaphragm	4"		Normal
20			VE11	Check	4"		Normal
	M243	9.5.12-1	VQ1A	Diaphragm	6"	T	Normal
35		9.5.12-1	VQ2B	Diaphragm	6"	T	Normal
30	M384	9.5.12-1	VA6A	Gate	6"	T	Normal
		9.5.12-1	VQ5B	Gate	6"	T	Normal
	M361	---	NC195	Globe	2"	T	Normal
20		---	NC196	Globe	2"	T	Normal
35			NC259	Check	3/4"		Normal
	M353	9.5.1-2	RF823	Check	4"		Normal
		9.5.1-2	RF821	Globe	6"	L.C.	Normal
30							
	M376	9.2.4-3	KC322	Check	3"		Normal
		9.2.4-3	KC320A	Diaphragm	3"	T	Normal
35	M355	9.2.4-3	KC333A	Diaphragm	3"	T	Normal
		9.2.4-3	KC332B	Diaphragm	3"	T	Normal
30		9.2.4-3	KC280	Check	1"		Normal
35							
	M327	9.2.4-3	KC338B	Diaphragm	8"	T	Normal
30		9.2.4-3	KC340	Check	8"		Normal
	M320	9.2.4-3	KC425A	Diaphragm	8"	T	Normal
35		9.2.4-3	KC424B	Diaphragm	8"	T	Normal
		9.2.4-3	KC279	Check	1"		Normal

Table 6.2.4-3

Containment Penetrations Not Subject
To Type C Leakrate Tests

	Penetrations	FSAR Fig. No.	Line Size	Function
20				
	M340	9.3.2-3	3/4"	Steam Generator Sample
	M335	9.3.2-3	3/4"	Steam Generator Sample
	M341	9.3.2-3	3/4"	Steam Generator Sample
30				
	M218	9.2.4-3	3"	Component Cooling
20	M217	9.2.4-3	3"	Component Cooling

Table 6.2.4-4
Bi-directional Leakage Test Results for Typical Diaphragm Value

TEST DESCRIPTION

A 2" diaphram valve was flanged on both sides. Test connections were provided on both flanges. A leak was imposed by inserting a 26 gauge wire under the valve seat.

Twelve (12) test runs with different leak magnitudes (by changing the torque on the valve seat) were performed at 14.2 PSIG.

RUN	A LEFT DIRECTION LEAK RATE (SCCM)	B RIGHT DIRECTION LEAK RATE (SCCM)	C RIGHT DIRECTION LEAK RATE REPEATED (SCCM)	D % LEAK RATE $D=\frac{B-A}{C}\times 100$ (%)	E REPEATABILITY $E=\frac{C-B}{C}\times 100$ (%)
1	146	146		0.	
2	160	160		0.	
3	178	185		3.780	
4	195	197		1.011	
5	213	224		4.910	
6	239	241		0.829	
7	256	256		0.	
8	283	278	281	1.766	1.060
9	307	300	303	2.280	0.990
10	332	324	326	2.409	0.613
11	357	348	350	2.521	0.571
12	382	371	375	2.880	1.066

7.2 REACTOR PROTECTION SYSTEM

7.2.1 DESCRIPTION

7.2.1.1 System Description

The Reactor Protection System automatically keeps the reactor operating within a safe region by shutting down the reactor whenever the limits of the region are approached. The safe operating region is defined by several considerations such as mechanical/hydraulic limitations on equipment, and heat transfer phenomena. Therefore, the Reactor Protection System keeps surveillance on process variables which are directly related to equipment mechanical limitations, such as pressure, pressurizer water level (to prevent water discharge through safety valves, and uncovering heaters) and also on variables which directly affect the heat transfer capability of the reactor (e.g. flow, reactor coolant temperatures). Still other parameters utilized in the Reactor Protection System are calculated from various process variables. In any event, whenever a direct process or calculated variable exceeds a setpoint the reactor is shut down in order to protect against either gross damage to fuel cladding or loss of system integrity which could lead to release of radioactive fission products into the Containment.

The following subsystems make up the Reactor Protection System:

35 a. Process Instrumentation (Reference 1)

b. Nuclear Instrumentation System (Reference 2)

c. Solid State Logic Protection System (Reference 3)

d. Reactor Trip Switchgear (Reference 3)

e. Manual Actuation Circuit

The Reactor Protection System consists of sensors which, when connected with analog circuitry consisting of two to four redundant channels, monitor various unit parameters and digital circuitry, consisting of two redundant logic trains which receive inputs from the analog protection channels to complete the logic necessary to automatically open the reactor trip breakers.

Each of the two logic trains, A and B, is capable of opening a separate and independent reactor trip breaker, RTA and RTB, respectively. The two trip breakers in series connect three phase ac power from the rod drive motor generator sets to the rod drive power cabinets, as shown on Figure 7.2.1-1, Sheet 2. During unit power operation, a dc under-voltage coil on each reactor trip breaker holds a trip plunger out against its spring, allowing the power to be available at the rod control power supply cabinets. For reactor trip, a loss of dc voltage to the undervoltage coil releases the trip plunger and trips open the breaker. When either of the trip breakers opens, power is interrupted to the rod drive power supply, and the control rods fall, by gravity, into the core. The rods cannot be withdrawn until the trip breakers are manually reset. The trip breakers cannot be reset until the abnormal condition which initiated the trip is corrected. Bypass breakers BYA and BYB are provided to permit testing of the trip breakers, as discussed in 7.2.2.2.3.

the pressurizer pressure measurements (compensated for rate of change) fall below preset limits. The trip logic is automatically enabled above P-7. This signal is compensated to account for the fact that the measurement is in the pressurizer rather than in the core proper. This trip is blocked below P-7 because at low power levels the trip is not required. The trip logic and interlocks are given in Table 7.2.1-1.

The trip logic is shown on Figure 7.2.1-1, Sheet 6. A detailed functional description of the process equipment associated with the function is contained in Reference 1.

2. Pressurizer high pressure trip

The purpose of this trip is to protect the Reactor Coolant System against system overpressure.

The same sensors and transmitters that are used for the pressurizer low pressure trip are also used for the high pressure trip except that separate bistables are used for high pressure trip. These bistables trip when uncompensated pressurizer pressure signals exceed preset limits on coincidence as listed in Table 7.2.1-1. There are no interlocks or permissives associated with this trip function. This trip protects against overstressing the reactor coolant pressure boundary.

The logic for this trip is shown on Figure 7.2.1-1, Sheet 6. The detailed functional description of the process equipment associated with this trip is provided in Reference 1.

3. Pressurizer high water level trip

The purpose of this trip is to prevent water relief through the pressurizer safety valves and therefore provide for equipment protection. This trip is automatically blocked below P-7 to permit startup. The coincidence logic and interlocks of pressurizer high water level signals are given in Table 7.2.1-1.

The trip logic for this function is shown on Figure 7.2.1-1, Sheet 6. A detailed description of the process equipment associated with this function is contained in Reference 1.

d. Reactor Coolant System Low Flow Trips

These trips protect the core from DNB in the event of a loss of coolant flow. The means of sensing the loss of coolant flow are as follows:

1. Low reactor coolant flow

The parameter sensed is reactor coolant flow. Four elbow taps in each coolant loop are used as a flow sensing device that indicates the status of reactor coolant flow. The basic function of this device is to provide reduction in flow information. An output signal from two out of the three bistables in a loop indicates a low flow in that loop.

Above P-7 two-out-of-four loop low flow trips the reactor; above P-8 low flow in any one loop causes a reactor trip.

The coincidence logic and interlocks are given in Table 7.2.1-1.

The detailed functional description of the process equipment associated with the trip function is contained in Reference 1.

2. Reactor coolant pump bus undervoltage trip

5 This trip is required in order to protect against low flow which can result from loss of voltage to the reactor coolant pumps (e.g., from plant blackout).

35 There is one undervoltage sensing monitor connected to the motor side of each reactor coolant pump breaker. (These reactor coolant pump breakers are located in the Category 1 Auxiliary Building.) These adjustable monitors provide an output signal when the voltage goes below approximately 60-80 percent of normal operating voltage. Signals from monitors connected to any two of the pumps (time delayed up to approximately 0.20 seconds to prevent spurious trips cause by short
5 term voltage perturbations) trip the reactor if the power level is above P-7. The coincidence logic and interlocks are given in Table 7.2.1-1.

3. Reactor coolant pump bus underfrequency trip

This trip is required for the protection of the reactor from low flow resulting from bus underfrequency (e.g., major power grid frequency disturbance). This trip trips the reactor for an underfrequency condition. The setpoint of the underfrequency monitors is adjustable between 54 and
5 59 Hz.

One underfrequency sensing monitor is connected to each reactor coolant pump bus. (The reactor coolant pump bus is located in the Category 1
24 Auxiliary Building.) Signals from monitors connected to any two of the
5 buses (time delayed up to spproximately 0.2 seconds to prevent spurious trips caused by short term frequency perturbations) will cause a direct
35 trip of the reactor if the power level is above P-7. An underfrequency condition will trip the reactor coolant pump breakers at any power level.

Figure 7.2.1-1, Sheet 5, shows the logic for the Reactor Coolant System low flow trips.

e. Steam Generator Trip

The specific trip function generated is as follows:

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1. Low-low steam generator water level trip

This trip protects the reactor from loss of heat sink in the event
35 of a sustained steam/feedwater flow mismatch. This trip is actuated on two-out-of-three low-low water level signals occurring in any steam generator.

The logic is shown on Figure 7.2.1-1, Sheet 7. A detailed functional description of the process equipment associated with this trip is provided in Reference 1.

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f. Safety Injection Signal Actuation Trip

A reactor trip occurs when the Safety Injection System is actuated. The means of actuating the Safety Injection System are described in Section 7.3. This trip protects the core against a loss of reactor coolant or steam.

Figure 7.2.1-1, Sheet 8, shows the logic for this trip. A detailed functional description of the process equipment associated with this trip function is provided in Reference 1.

g. Manual Trip

The manual trip consists of two switches with two outputs on each
35 switch. One output is used to actuate the train A trip breaker; the other output actuates the train B trip breaker. Operating either manual trip switch removes the voltage from the undervoltage trip coil, energizes the shunt trip coil, and trips the reactor.

There are no interlocks which can block this trip. Figure 7.2.1-1, Sheet 3, shows the manual trip logic.

7.2.1.1.3 Reactor Protection System Interlocks

a. Power Escalation Permissives

The overpower protection provided by the out-of-core nuclear instrumentation consists of three discrete, but overlapping, ranges. Continuation of startup operation or power increase requires a permissive signal from the higher range instrumentation channels before the lower range level trips can be manually blocked by the operator.

A one-of-two intermediate-range permissive signal (P-6) is required prior to source range level trip blocking. A source-range manual block is provided for each logic train and the blocks must be in effect on both trains in order to proceed in the intermediate range. Source range level trips are automatically reactivated and high voltage restored when both intermediate range channels are below the permissive (P-6) level. There is a manual reset switch for administratively reactivating the source range level trip and detector high voltage when between the permissive P-6 and P-10 level, if required. Source range level trip block and high voltage cutoff are always maintained when power is above the permissive P-10 level in order to prevent detector damage.

The intermediate-range level trip and power-range (low setpoint) trip can only be blocked after satisfactory operation and permissive information are obtained from two-of-four power-range channels. Individual blocking switches are provided so that the low setpoint power range trip and intermediate-range trip can be independently blocked. These trips are automatically reactivated when any three of the four power range channels are below the permissive (P-10) level, thus ensuring automatic activation to more restrictive trip protection.

The development of permissives P-6 and P-10 is shown on Figure 7.2.1-1, Sheet 4. All of the permissives are digital; they are derived from analog signals in the nuclear power range and intermediate-range channels.

See Table 7.2.1-2 for the list of protection system interlocks.

b. Blocks of Reactor Trips at Low Power

Interlock P-7 blocks a reactor trip at low power (below approximately 10 percent of full power) from low reactor coolant flow, reactor coolant pump
5 under voltage, reactor coolant pump underfrequency, pressurizer low pressure,
25 or, pressurizer high water level. See Figure 7.2.1-1,
35 Sheets 5, 6 and 16, for permissive applications. The low power signal is derived from three-out-of-four power range neutron flux signals below the setpoint in coincidence with two-out-of-two turbine impulse chamber pressure signals below the setpoint (low unit load).

The P-8 interlock blocks a low reactor coolant flow reactor trip when the unit is below approximately 47 percent of full power. The block action (absence of the P-8 interlock signal) occurs when three-out-of-four neutron flux power range signals are below the setpoint. Thus, below the P-8 setpoint, the reactor is allowed to operate with one inactive loop and trip does not occur until two loops are indicating low flow.

See Figure 7.2.1-1, Sheet 4, for derivation of P-8, and Sheet 5 for applicable logic. See Table 7.2.1-2 for the list of protection system blocks.

7.2.1.1.4 Coolant Temperature Sensor Arrangement

The hot and cold leg resistance temperature detectors are inserted into reactor coolant bypass loops. A bypass loop from upstream of the steam generator to downstream of the steam generator is used for the hot leg resistance temperature detectors and a bypass loop from downstream of the reactor coolant pump to upstream of the pump is used for the cold leg resistance temperature detectors. The complete bypass loop is inside the Containment. The resistance temperature detectors are located in manifolds and are directly inserted into the reactor coolant bypass loop flow without thermowells. Thermowells are not used in order to keep the detector thermal lag small. The bypass arrangement permits replacement of defective temperature elements while the unit is at hot shutdown without draining or depressurizing the reactor coolant loops.

Three sampling probes are installed in a cross sectional plane of each hot leg at approximately 120 degree intervals. Each of the sampling probes, which extends several inches into the hot leg coolant stream, contains five inlet orifices distributed along its length. In this way a total of fifteen locations in the hot leg stream are sampled providing a representative coolant temperature measurement. The two inch diameter pipe leading to the resistance temperature manifold provides mixing of the samples to provide average hot leg temperature measurement.

Care is taken to distribute the flow evenly among the five orifices of each probe by effectively restricting the flow through the orifices. This is done by designing a smaller overall orifice flow area than that of the common flow channel within the probe. This arrangement is also applied to the flow transition from the three probe flow channels to the pipe leading to the temperature element manifold. The total flow area of these channels is, therefore, designed to be less than that of the two inch pipe connecting the probes to the manifold.

The cold leg reactor coolant flow is well mixed by the reactor coolant pump thereby eliminating any cold leg temperature spatial dependence. Therefore, the cold leg sample is taken directly from a two inch pipe tap off the cold leg downstream of the pump.

7.2.1.1.5 Analog System

The process analog system is described in Reference 1.

7.2.1.1.6 Solid State Logic Protection System

The solid state logic protection system takes digital inputs (voltage/no voltage) from the process and nuclear instrument channels corresponding to conditions (normal/abnormal) of unit parameters. The system combines these signals in the required logic combination and generates a trip signal (no voltage) to the undervoltage coils of the reactor trip circuit breakers when the necessary combination of signals occur. The system also provides annunciator, status light and computer input signals which indicate the condition of bistable input signals, partial trip and full trip functions and the status of the various blocking, permissive and actuation functions. In addition, the system includes means for semi-automatic testing of the logic circuits. A detailed description of this system is given in Reference 3.

7.2.1.1.7 Isolation Amplifiers

In certain applications, it is considered advantageous to employ control signals derived from individual protection channels through isolation amplifiers contained in the protection channel, in accordance with IEEE-279.

In all of these cases, analog signals derived from protection channels for non-protective functions are obtained through isolation amplifiers located in the analog protection racks. By definition, non-protective functions include those signals used for control, remote process indication, and computer monitoring.

Isolation amplifier qualification type tests are described in References 4 and 5.

7.2.1.1.8 Energy Supply and Environmental Variations

The energy supply for the Reactor Protection System, including the voltage and frequency variations, is described in Section 7.6 and Chapter 8. The environmental variations, throughout which the system performs, is given in Section 3.11 and Chapter 8.

7.2.1.1.9 Setpoints

The setpoints that require trip action, when reached, are given in the Technical Specifications.

7.2.1.1.10 Seismic Design

The seismic design considerations for the Reactor Protection System are given in Section 3.10. This design meets the requirements of Criterion 2 of the 1971 GDC.

7.2.1.2 <u>Design Bases Information</u>

The information given below presents the design bases information requested by Section 3 of IEEE 279, 1971 Reference 8. Functional logic diagrams are presented in Figure 7.2.1-1.

7.2.1.2.1 Unit Conditions

The following are the generating station conditions requiring reactor trip.

a. DNBR approaching 1.30.

b. Power density (kilowatts per foot) approaching rated value for Condition II faults (See Chapter 4 for fuel design limits).

c. Reactor Coolant System overpressure creating stresses approaching the limits specified in Chapter 5.

7.2.1.2.2 Unit Variables

The following are the variables required to be monitored in order to provide reactor trips (See Table 7.2.1-1).

a. Neutron flux

b. Reactor coolant temperature

c. Reactor Coolant System pressure (pressurizer pressure)

d. Pressurizer water level

e. Reactor coolant flow

25 f. Reactor coolant pump operational status (bus voltage and frequency)

g. Steam generator water level

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7.2.1.2.3 Spatially Dependent Variables

The following variable is spatially dependent: Reactor coolant temperature - see Section 7.3.1.2 for a discussion of this variable spatial dependence.

7.2.1.2.4 Limits, Margins and Levels

The parameter values that require reactor trip are given in Technical Specifications, and in Chapter 15, Accident Analysis. The Accident Analysis proves that the setpoints used in the Technical Specifications are conservative.

The setpoints for the various functions in the Reactor Protection System have been analytically determined such that the operational limits so prescribed prevent fuel rod clad damage and loss of integrity of the Reactor Coolant System as a result of any Condition II incident (anticipated malfunction). As such, the Reactor Protection System limits the following parameters to:

a. Minimum DNBR = 1.3

b. Maximum System Pressure = 2750 psia

c. Fuel rod maximum linear power = 21.1 KW/ft.

The accident analyses described in Section 15.2 demonstrate that the functional requirements as specified for the Reactor Protection System are adequate to meet the above considerations, even assuming, for conservatism, adverse combinations of instrument errors. A discussion of the safety limits associated with the reactor core and Reactor Coolant System, plus the limiting safety system setpoints, are presented in the Technical Specifications.

7.2.1.2.5 Abnormal Events

The malfunctions, accidents or other unusual events which could physically damage Reactor Protection System components or could cause environmental changes are as follows:

a. Earthquakes (refer to Chapter 3 and Chapter 2).

b. Fire (refer to Section 9.5).

c. Explosion (Hydrogen Buildup inside Containment). (Refer to Section 6.2).

d. Missiles (refer to Section 3.5).

e. Flood (refer to Chapter 2 and 3).

f. Wind and Tornadoes (refer to Section 3.3).

7.2.1.2.6 Minimum Performance Requirements

The performance requirements are as follows:

a. System response times:

The time delays are defined as the time required for the reactor trip (i.e., the time the rods are free and begin to fall) to be initiated following a step change in the variable being monitored from 5 percent below to 5 percent above the trip setpoint. During preliminary startup tests, it is demonstrated that actual time delays of installed equipment are equal to or less than the values assumed in the accident analyses.

1. Typical maximum allowable time delays in generating the reactor trip signal:

		Time (sec.)
a)	Power range nuclear power (High and low setpoint)	0.5
b)	Neutron flux rates (positive and negative)	0.5
c)	Overtemperature ΔT (maximum) (including transport time of 2 sec.)	6.0
d)	Overpower ΔT (maximum) (including transport time of 2 sec.)	6.0
e)	Pressurizer pressure (low and high)	1.0
f)	Pressurizer high water level	2.0
g)	Low reactor coolant flow	1.0
h)	Reactor coolant pump bus underfrequency	0.6
5 i)	Reactor coolant pump bus under voltage	1.5
35 j)	Low-low steam generator water level	2.0

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b. Reactor trip accuracies are given in Table 7.2.1-3.

7.2.1.3 Final System Drawings

Functional block diagrams, electrical elementaries and other drawings required to assure electrical separation and perform a safety review are provided in the McGuire Electrical Schematics.

7.2.2 ANALYSES

7.2.2.1 Failure Mode and Effects Analyses

A failure mode and effects analyses of the Reactor Protection System has been performed. Results of this study and a fault tree analysis are presented in Reference 6.

7.2.2.2 Evaluation of Design Limits

While most setpoints used in the Reactor Protection System are fixed, there are variable setpoints, most notably the overtemperature ΔT and overpower ΔT setpoints. All setpoints in the Reactor Protection System have been selected on the basis of engineering design and safety studies. The capability of the Reactor Protection System to prevent loss of integrity of the fuel cladding and/or Reactor Coolant System pressure boundary during Condition II and III transients is demonstrated in the Accident Analysis, Chapter 15. These safety analyses are carried out using those setpoints determined from results of the engineering design studies. Setpoint limits are presented in the Technical Specifications. A discussion of the intent for each of the various reactor trips and the accident analysis (where appropriate) which utilizes this trip is presented below. It should be noted that the selected trip setpoints all provide for margin before protective action is actually required to allow for uncertainties and instrument errors. The design meets the requirements of Criteria 10 and 20 of the 1971 GDC.

7.2.2.2.1 Trip Setpoint Discussion

It has been pointed out previously that below a DNB ratio of 1.3 there may be a potential for local fuel cladding failure. The DNB ratio existing at any point in the core for the core design has been determined as a function of the core inlet temperature, power output, operating pressure and flow. Consequently, core safety limits in terms of a DNBR equal to 1.30 for the hot channel have been developed as a function of core ΔT, T_{avg} and pressure for a specified flow as illustrated by the solid lines in Figure 7.2.1-3. Also shown as solid lines in Figure 7.2.1-3 are the loci of conditions equivalent to 118 percent reactor power as a function of ΔT and T_{avg} representing the overpower (KW/ft) limit on the fuel. The dashed lines indicate the maximum permissible set point (ΔT) as a function of T_{avg} and pressure for the overtemperature and overpower reactor trip. Actual setpoint constants in the equation representing the dashed lines are as given in the Technical Specification. These values are conservative to allow for instrument errors. The design meets the requirements of Criteria 10, 15, 20 and 29 of the 1971 GDC.

DNBR is not a directly measurable quantity; however, the process variables that determine DNBR are sensed and evaluated. Small isolated changes in various process variables may not individually result in violation of a core safety limit. The design concept of the Reactor Protection System

takes cognizance of this situation by providing reactor trips associated with individual process variables in addition to the overpower/overtemperature safety limit trips. Process variable trips prevent reactor operation whenever a change in the monitored value is such that a core or system safety limit is in danger of being exceeded should operation continue. Basically, the high pressure, low pressure and overpower/overtemperature ΔT trips provide sufficient protection for slow transients as opposed to
35 such trips as reactor coolant pump undervoltage, high flux, or rate, which trip the reactor for faster changes in flow or flux, respectively, that would result in fuel damage before actuation of the slower responding ΔT trips could be effected.

Therefore, the Reactor Protection System is designed to provide protection for fuel cladding and Reactor Coolant System pressure boundary integrity where: (1) a rapid change in a single variable or factor could potentially result in exceeding a core or a system safety limit, and (2) a slow change in one or more variables will have an integrated effect which potentially could cause safety limits to be exceeded. Overall, the Reactor Protection System offers diverse and comprehensive protection against fuel cladding failure and/or loss of Reactor Coolant System integrity for Condition II and III accidents. This is demonstrated by Table 7.2.1-4 which lists the various trips of the Reactor Protection System, the corresponding Technical Specification on safety limits and safety system settings and the appropriate accident discussed in the Accident Analyses in which the trip could be utilized.

It should be noted that the Reactor Protection System automatically provides core protection during non-standard operating configuration, i.e. operation with a loop out of service. Although operating with a loop out of service over an extended time is considered to be an unlikely event, no protection system setpoints need to be reset. This is because the nominal value of the power (P-8) interlock setpoint restricts the power levels such that DNB ratios less than 1.30 are not realized during Condition II transients occurring during this mode of operation. This restricted power level is considerably below the boundary of permissible values as defined by the core safety limits for operation with a loop out of service. By first resetting the coefficient setpoints in the overtemperature ΔT function to more restrictive values as listed in Technical Specifications, the P-8 setpoint can then be increased to the maximum value consistent with maintaining DNBR above 1.30 for Condition II transients in the one loop shutdown mode. Thus the P-8 interlock acts essentially as a high nuclear power reactor trip when operating with one loop not in service. The resetting of the ΔT overtemperature trip and P-8 is carried out under prescribed administrative procedures and only under the direction of authorized supervision.

Manual adjustment of the Overtemperature ΔT reactor trip setpoint to a more restrictive setting when operating with a loop out of service is not essential for unit safety. If the more restrictive setting is not made,
5 reactor trip occurs when the power exceeds the P-8 setpoint. Thus the Q223.B
use of administrative procedures permits higher power when operating with a loop out of service, but implementation of the procedures is not required to assure protection action.

The Reactor Protection System design was evaluated in detail with respect to common mode failure and is presented in Reference 6 and 7. The design meets the requirements of Criterion 21 of the 1971 GDC.

Preoperational testing is performed on Reactor Protection System components and systems to determine equipment readiness for startup. This testing serves as a further evaluation of the system design.

Analyses of the results of Condition I, II, III and IV events, including considerations of instrumentation installed to mitigate their consequences are presented in Chapter 15. The instrumentation installed to mitigate the consequences of load rejection and turbine trip is given in Section 7.4.

7.2.2.2.2 Reactor Coolant Flow Measurement

The elbow taps used on each loop in the primary coolant system are instrument devices that indicate the status of the reactor coolant flow. The basic function of this device is to provide information as to whether or not a reduction in flow has occurred. The correlation between flow and elbow tap signal is given by the following equation:

$$\frac{\Delta P}{\Delta P_o} = \left(\frac{w}{w_o}\right)^2$$

where ΔP_o is the pressure differential at the reference flow w_o, and ΔP is the pressure differential at the corresponding flow, w. The full flow reference point is established by extrapolating along the correlation curve. The expected absolute accuracy of the channel is within ± 10 percent of full flow and field results have shown the repeatability of the trip point to be within ± 1 percent.

7.2.2.2.3 Evaluation of Compliance to Applicable Codes and Standards

The Reactor Protection System meets the criteria of the AEC General Design Criteria as indicated. The Reactor Protection System meets the criteria of IEEE-Standard 279, Reference 8, as indicated below.

a. Single Failure Criterion

The protection system is designed to provide two, three, or four instrumentation channels for each protective function and two logic train circuits. These redundant channels and trains are electrically isolated and physically separated. Thus, any single failure within a channel or train does not prevent protective action at the system level when required. Loss of input power to a channel or logic train results in a signal calling for a trip. This design meets the requirements of Criterion 23 of the 1971 GDC.

To prevent the occurrence of common mode failures, such additional measures as functional diversity, physical separation, and testing as well as administrative control during design, production, installation and operation are employed, as discussed in Reference 6. The design meets the requirements of Criteria 21 and 22 of the 1971 GDC.

b. Quality of Components and Modules

For a discussion of the quality of the components and modules used in the Reactor Protection System, refer to Chapter 17. The quality assurance applied conforms to Criterion 1 of the 1971 GDC.

c. Equipment Qualification

For a discussion of the type tests made to verify the performance requirements, refer to Section 3.11. The test results demonstrate that the design meets the requirements of Criterion 4 of the 1971 GDC.

d. Independence

Channel independence is maintained throughout the system, extending from the sensor through to the devices actuating the protective function. Physical separation is used to achieve separation of redundant transmitters. Separation of wiring is achieved by using separate wireways, cable trays, conduit runs and Containment penetrations for each redundant channel. Redundant analog equipment is separated by locating modules in separate protection cabinets. Each redundant channel is energized from a separate ac power feed. This design meets the requirements of Criterion 21 of the 1971 GDC.

Independence of the logic trains is discussed in Reference 3. Two reactor trip breakers, actuated by separate logic matrices, interrupt power to the control rod drive mechanisms. The breaker main contacts are connected in series with the power supply so that opening either breaker interrupts power to all full length control rod drive mechanisms, permitting the rods to free fall into the core. See Figure 7.2.1-4.

The design philosophy is to make maximum use of a wide variety of measurements. The protection system continuously monitors numerous diverse system variables. The extent of this diversity has been evaluated for a wide variety of postulated accidents and is discussed in Reference 7. Generally, two or more diverse protection functions would terminate an accident before intolerable consequences could occur. The design meets the requirements of Criterion 22 of the 1971 GDC.

e. Control and Protection System Interaction

The protection system is designed to be independent of the control system. In certain applications the control signals and other non-protective functions are derived from individual protective channels through isolation amplifiers. The isolation amplifiers are classified as part of the protection system and are located in the analog protective racks. Non-protective functions include those signals used for control, remote process indication, and computer monitoring. The isolation amplifiers are designed such that a short circuit, open circuit, or the application of 118VAC or 140VDC on the isolated portion of the circuit (i.e., the non-protective side of the circuit) will not effect the input (protective) side of the circuit. The signals obrained through the isolation amplifiers are never returned to the protective racks. This design meets the requirements of Criterion 24 of the GDC and Section 4.7 of IEEE 279, 1971, Reference 8.

A detailed discussion of the design and testing of the isolation amplifiers is given in Reference 4 and 5. These reports include the results of applying various malfunction conditions on the output portion of the isolation amplifiers. The results show that no significant disturbance to the isolation amplifier input signal occurred.

The redundant, isolated control signal cables leaving the protection racks come into close proximity at locations such as the control board. It could be postulated that electrical faults, or interference, at these locations might be propagated into all redundant racks and degrade protection circuits because of the close proximity of protection and control wiring within each rack.

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Westinghouse test programs have demonstrated that Class 1E protection systems Nuclear Instrumentation System (NIS), Solid State Protection System (SSPS) and 7300 Process Control System (7300 PCS) are not degraded by non-Class 1E circuits, sharing the same enclosure, which could be postulated to carry electrical faults or interference into the enclosures.

Tests conducted on the as-built designs of the NIS and SSPS were reported and accepted by the NRC in support of the Diablo Canyon application (Docket No's 50-275 and 50-323). Westinghouse considers these programs as applicable to all plants, including McGuire. Westinghouse tests on the 7300 PCS were covered in a report entitled "7300 Series Process Control System Noise Tests" subsequently reissued as WCAP-8892 (Reference 10). In a letter dated April 20, 1977, R. Tedesco to C. Eicheldinger, the NRC accepted the report in which the applicability of the McGuire plant is established.

f. Capability for Testing

The Reactor Protection System is capable of being tested during power operation. Where only parts of the system are tested at any one time, the testing sequence provides the necessary overlap between the parts to assure complete system operation. The testing capabilities are in conformance with Regulatory Guide 1.22 as discussed in Subdivision 7.1.2.1.

The protection system is designed to permit periodic testing of the analog channel portion of the Reactor Protection System during reactor power operation without initiating a protective action unless a trip condition actually exists. This is because of the coincidence logic required for reactor trip. Note, however, that the source and intermediate range high neutron flux trips must be bypassed during testing.

The following types of sensors provide input to the protection sets of Process Control System and Solid State Protection System for Engineered Safety Features Actuation and Reactor Protection System:

1) Differential Pressure Transmitters (Level)
2) Differential Pressure Transmitters (Flow)
3) Pressure Transmitters
4) Frequency Transmitters
5) Voltage Transmitters
6) Circuit Breaker Auxiliary Contacts
7) Pressure Switches
8) Valve Limit Switches
9) Resistance Temperature Detectors

The response time of differential pressure transmitters (level), differential pressure transmitters (flow), pressure transmitters, frequency transmitters, and voltage transmitters are tested by one of the following methods: Q 223.35

1) Test sensor in place by perturbing the process being monitored using existing equipment used for normal plant operation

2) Test sensor in place by perturbing the process input using additional equipment provided for response time testing.

3) Remove the sensor from service and bench test the device.

Circuit breaker auxiliary contacts, pressure switches, and valve limit switches are tested for operation only. Since these are bistable devices, no significant change in response time is anticipated when compared to the overall response time of the system.

No significant deterioration in response time of resistance temperature detector elements is anticipated. For this reason, the response time of RTD elements are not tested. Duke Power is following the EPRI study concerning sensor response time testing and will consider implementation of any recommendations resulting from the study.

The operability of the process sensors is ascertained by comparison with redundant channels monitoring the same process variables or those with a fixed known relationship to the parameter being checked. The incontainment process sensors can be calibrated during unit shutdown if required.

Analog channel testing is performed at the analog instrumentation rack set by individually introducing dummy input signals into the instrumentation.

channels and observing the tripping of the appropriate output bistables. Process analog output to the logic circuitry is interrupted during individual channel test by a test switch which, when thrown, de-energizes the associated logic input and inserts a proving lamp in the bistable output. Interruption of the bistable output to the logic circuitry for any cause (test, maintenance purposes, or removed from service) causes that portion of the logic to be actuated (partial trip) accompanied by a partial trip alarm and channel status light actuation in the Control Room. Each channel contains those switches, test points, etc. necessary to test the channel. See Reference 1 for additional information.

The power range channels of the Nuclear Instrumentation System are tested by superimposing a test signal on the actual detector signal being received by the channel at the time of testing. The positive and negative flux rate trips are tested by applying an appropriate step change to the input of each power range channel. This is accomplished in the test mode by alternating between 2 test signals where one test signal is adjusted to be the required percent of full power above the second test signal. Turning the Test Signal Selector from the higher valve signal to the lower valve signal introduces the negative step change. Turning the selector
5 back to the higher valve introduces the positive step change. Bistable Q
action is verified by control board annunciator and trip status lights. 223.9
The test signals are introduced at a point equivalent to the detector signal inputs on each power range drawer assembly.

Figure 7.2.1-1 (Sheet 3) of the FSAR shows the output of the "rate trip." The power range channel showing test injection points is shown in Figure 2-3 of WCAP-8255, Reference 2. A description of the test circuit operation for each channel is also included on Page 3-13 in Section 3.6 of WCAP-8255, Reference 2. The output of the bistable is not placed in a tripped condition prior to testing. Also, since the power range channel logic is two-out-of-four, bypass of this reactor trip function is not required.

To test a power range channel, a TEST OPERATE switch is provided to require deliberate operator action. The operation of the test switch initiates the CHANNEL TEST annunciator in the Control Room. Bistable operation is tested by increasing the test signal level to bistable trip setpoint and verifying bistable relay operation by control board annunciator and trip status lights.

It should be noted that a valid trip signal causes the channel under test to trip at a lower actual reactor power level. A reactor trip occurs when a second bistable trips. No provision is made in the channel test circuit for reducing the channel signal level below that signal being received from the Nuclear Instrumentation System detector.

A Nuclear Instrumentation System channel which can cause a reactor trip through one of two protection logic (source or intermediate range) is provided with a bypass function which prevents the initiation of a reactor trip from that particular channel during the short period that it is undergoing test. These bypasses are annunciated in the Control Room.

For a detailed description of the Nuclear Instrumentation System see Reference 2.

The design of the Reactor Coolant Pump Monitor Panel provides a high degree
of flexibility for testing of the trip logic circuits. Each of the under-
voltage and underfrequency trip signals is generated by an individual two-
out-of-four logic system. Use of the two-out-of-four SSPS logic permits
calibration and/or testing of one channel at a time during reactor operation
5 without jeopardizing overall system performance. Key lock test switches are Q223.4
provided to break the potential inputs to the voltage sensing circuits to
functionally test each channel. The underfrequency channels can be tested
in the same manner. The RCP undervoltage and underfrequency monitors comply
with the requirements of IEEE 279-1971.

The reactor logic trains of the Reactor Protection System are designed to be capable of complete testing at power. Annunciation is provided in the Control Room to indicate when a train is in test (train output bypassed) and when a reactor trip breaker is bypassed. Details of the logic system testing are given in Reference 3.

The reactor coolant pump breakers cannot be tripped at full power without causing a unit upset by loss of power to a coolant pump.

Manual trip cannot be tested at power without causing a reactor trip since
35 operation of either manual trip switch actuates both trains. Note,
however, that manual trip could also be initiated from outside the Control
Room by manually tripping one of the reactor trip breakers.

The reactor trip function that is derived from the automatic safety injection signal is tested at power as follows:

a. The analog signals, from which the automatic safety injection signal is
derived, are tested at power in the same manner as the other analog
signals and as described in Subdivision 7.2.2.2.3. The processing of
these signals in the Solid State Protection System (SSPS) wherein their
8 channel orientation converts to a logic train orientation is tested at Q223.27
power by the built-in semi-automatic test provisions of the SSPS as
described in Reference 3. The reactor trip breakers are tested at power
as discussed in Subdivision 7.2.2.2.3. The testing of reactor trip from
safety injection during refueling refers only to the manual safety
injection actuation function.

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hot leg and cold leg bypass loop resistance temperature detectors as a function of unit power is also checked during unit startup tests. The absolute value of ΔT versus unit power is not important, per se, as far as reactor protection is concerned. Reactor Protection System setpoints are based upon percentages of the indicated ΔT at nominal full power rather than on absolute values of ΔT. This is done to account for loop differences which are inherent. Therefore, the percent ΔT scheme is relative, not absolute, and provides better protective action. For this reason, the linearity of the ΔT signals as a function of power is of importance rather than the absolute values of the ΔT. As part of the unit startup tests, the bypass loop resistance temperature detector signals are compared with the core exit thermocouple signals.

Reactor control is based upon signals derived from protection system channels after isolation by isolation amplifiers such that no feedback effect can perturb the protection channels.

Since control is based on the temperature of the loop with the highest average temperature, control rod movement is based upon the most conservative temperature measurement with respect to DNB margin. A spurious low average temperature measurement from any loop temperature control channel does not effect control action. A spurious high average temperature measurement causes rod insertion (safe direction).

Individual low flow alarms with individual status lights for each reactor coolant loop bypass flow are provided on the main control board. The alarm and status lights provide the operator with immediate indication of a low flow condition in the bypass loops associated with any reactor coolant loop.

Local indicators are provided to monitor total flow through the resistance temperature detector bypass manifolds for each loop. The indicators are located inside Containment but are accessible during power operations.

Flow is locally monitored:

a. Prior to restoring temperature channels to normal service following reopening of bypass loop stop valves whenever a bypass loop has been out of service.

b. Following any bypass loop low flow alarm (see above).

In addition, channel deviation signals in the control system give an alarm if any temperature channel deviates significantly from the auctioneered (highest) value. Automatic rod withdrawal blocks and turbine runback (power demand reduction) also occur if any two of the four overtemperature or overpower ΔT channels indicate an adverse condition.

7.2.2.3.3 Pressurizer Pressure

The pressurizer pressure protection channel signals are used for high and low pressure protection and as inputs to the overtemperature ΔT trip protection function. Isolated output signals from these channels are used for pressure control. These are used to control pressurizer spray and heaters and power operated relief valves. Pressurizer pressure is sensed by fast response pressure transmitters.

A spurious high pressure signal from one channel causes decreasing pressure by actuation of either spray or relief valves. Additional redundancy is provided in the low pressurizer pressure reactor trip logic and in the logic for safety injection to ensure low pressure protection.

In the unlikely event that the Control Room must be evacuated, the Reactor
Coolant System pressure can be maintained by use of the auxiliary spray supply Q212.40
9 valves and the pressurizer heaters. Controls for these valves and heaters
are located on the auxiliary shutdown panel.

The pressurizer heaters are incapable of overpressurizing the Reactor Coolant System. Overpressure protection is based upon the maximum positive surge of the reactor coolant produced as a result of turbine trip under full load, assuming the core continues to produce full power with normal feed flow maintained. The self-actuated safety valves are sized on the basis of steam flow from the pressurizer to accommodate this surge at a setpoint of 2500 psia and an accumulation of 3 percent. Note that no credit is taken for the relief capability provided by the power operated relief valves, or the steam dump system.

In addition, operation of any one of the power operated relief valves maintains pressure below the high pressure trippoint for most transients. The rate of pressure rise achievable with heaters is slow, and ample time and pressure alarms are available to alert the operator of the need for appropriate action.

Redundancy is not compromised by having a shared tap since the logic for this trip is two out of four. If the shared tap is plugged, the affected channels remain static. If the impulse line bursts, the indicated pressure drops to zero. In either case the fault is easily detectable, and the protective function remains operable.

7.2.2.3.4 Pressurizer Water Level

Three pressurizer water level channels are used for high pressurizer water level reactor trip. Isolated signals from these channels are used for pressurizer water level control. A failure in the level control system could fill or empty the pressurizer at a slow rate (on the order of half an hour or more), which allows ample time for corrective action by the operator.

The high water level trip setpoint provides sufficient margin such that the undesirable condition of discharging liquid coolant through the safety valves is avoided. Even at full power conditions, which produces the worst thermal expansion rates, a failure of the water level control does not lead to any liquid pressurizer pressure reactor trip actuating at a pressure sufficiently below the safety valve setpoint, or to the high pressurizer water level reactor trip.

For control failures which tend to empty the pressurizer, two out of four logic for safety injection action on low pressure ensures that the protection system can withstand an independent failure in another channel. In addition, ample time and alarms exist to alert the operator of the need for appropriate action.

7.2.2.3.5 Steam Generator Water Level

35 The basic function of the reactor protection circuits associated with low-low
steam generator water level is to preserve the steam generator heat sink for

removal of long-term residual heat. Should a complete loss of feedwater occur, the reactor would be tripped on low-low steam generator water level. In addition, redundant auxiliary feedwater pumps are provided to supply feedwater in order to maintain residual heat removal after trip. This reactor trip acts before the steam generators are dry. This reduces the required capacity, increases the time interval before auxiliary feedwater pumps are required, and minimized the thermal transient on the Reactor Coolant System and steam generators. Therefore, a low-low steam generator water level reactor trip circuit is provided for each steam generator to ensure that sufficient initial thermal capacity is available in the steam generator at the start of the transient. Two-out-of-four low-low steam generator water level trip logic ensures a reactor trip if needed even with an independent failure in another channel used for control and when degraded by an additional second postulated random failure.

A spurious low signal from the feedwater flow channel being used for control would cause an increase in feedwater flow. The mismatch between steam flow and feedwater flow produced by the spurious signal would actuate alarms to alert the operator of the situation in time for manual correction. If the condition continues, a two-out-of-three high-high steam generator water level signal in any loop, independent of the indicated feedwater flow, will cause feedwater isolation and trip the turbine. The high-high steam generator water level trip is an equipment protective trip preventing excessive moisture carryover which could damage the turbine blading.

In addition, the three element feedwater controller incorporates reset action on the level error signal, such that with expected controller settings a rapid increase or decrease in the flow signal would cause only a small change in level before the controller would compensate for the level error. A slow change in the feedwater signal would have no effect at all. A spurious low or high steam flow signal would have the same effect as high or low feedwater signal, discussed above.

A spurious high steam generator water level signal from the protection channel used for control will tend to close the feedwater valve. A spurious low steam generator water level signal will tend to open the feedwater valve. Before a reactor trip would occur, two -out-of-four channels in a loop would have to indicate a low-low water level. Any slow drift in the water level signal will permit the operator to respond to the level alarms and take corrective action.

Automatic protection is provided in case the spurious high level reduces feedwater flow sufficiently to cause low-low level in the steam generator. Automatic protection is also provided in case the spurious low level signal increases feedwater flow sufficiently to cause high level in the steam generator. A turbine trip and feedwater isolation would occur on two-out-of-three high-high steam generator water level in any loop.

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7.2.2.4 Additional Postulated Accidents

Loss of station instrument air or loss of component cooling water is discussed in Section 7.3.2. Load rejection and turbine trip are discussed in further detail in Section 7.7.

The control interlocks, called rod stops, that are provided to prevent abnormal power conditions which could result from excessive control rod withdrawal are discussed in Subdivision 7.7.1.4.1 and listed on Table 7.7.1-1. Excessively high power operation (which is prevented by blocking of automatic rod withdrawal), if allowed to continue, might lead to a safety limit (as given in Technical Specifications) being reached. Before such a limit is reached, protection is available from the Reactor Protection System. At the power levels of the rod block setpoints, safety limits have not been reached, and therefore these rod withdrawal stops do not come under the scope of safety related systems, and are considered as control systems.

7.2.3 TESTS AND INSPECTIONS

The Reactor Protection System meets the testing requirements of IEEE 338, 1971, Reference 9, as discussed in Subdivision 7.1.2.1. The testability of the system is discussed in Subdivision 7.2.2.2.3. The initial test intervals are specified in Chapter 16. Written test procedures and documentation, conforming to the requirements of Reference 9, will be available for audit by responsible personnel. Periodic testing complies with Regulatory Guide 1.22 as discussed in 7.1.2.1 and 7.2.2.2.3.

7.2.3.1 In-Service Tests and Inspections

Periodic surveillance of the Reactor Protection System is performed to ensure proper protective action. This surveillance consists of checks, calibrations, and channel functional testing which are summarized as follows:

a. Checks

A check consists of a qualitative determination of acceptability by observation of channel behavior during operation. It includes comparison of the channel with other independent channels measuring the same variable. Failures such as blown instrument fuses, defective indicators, or faulted amplifiers which result in "upscale" or "downscale" indication can be easily recognized by simple observation of the functioning of the instrument or system. Furthermore, in many cases such failures are revealed by alarm or annunciator action, and a check supplements this type of surveillance.

b. Calibration

A channel calibration consists of adjustment of channel output such that it responds, within acceptable range and accuracy, to known values of the parameter which the channel measures. Calibration encompasses the entire channel including alarm and/or trip, and includes the channel functional test discussed below. Thus, the calibration ensures the acquisition and presentation of accurate information.

c. Channel functional test

A channel functional test consists of injecting a simulated signal into the signal conditioning portion of the channel to verify its operability, including alarm and/or trip initiating action.

The minimum frequency for checks, calibration and testing are defined in Technical Specifications. Based on experience with both conventional and nuclear systems, when the unit is in operation the minimum checking frequencies set forth therein are considered adequate.

7.2.3.2 Periodic Testing of the Nuclear Instrumentation System

The following periodic tests of the Nuclear Instrumentation System are performed:

a. Testing at unit shutdown

   1. Source range testing

   2. Intermediate range testing

   3. Power range testing

b. Testing between P-6 and P-10 permissive power levels

   1. Source range testing

   2. Intermediate range testing

   3. Power range testing

c. Testing above P-10 permissive power level

   1. Source range testing

   2. Intermediate range testing

   3. Power range testing

Any deviations noted during the performance of these tests are investigated and corrected in accordance with the established calibration and trouble shooting procedures provided in the technical manual for the Nuclear Instrumentation System. Control and protection trip settings are indicated in the technical manual under Precautions, Limitations and Setpoints.

7.2.3.3 Periodic Testing of the Process Analog Channels of the Protection Circuits

The following periodic tests of the analog channels of the protection circuits are performed:

a. T_{avg} and ΔT protection channels

b. Pressurizer pressure protection channels

c. Pressurizer water level protection channels

d. Steam flow protection channels

e. Steam generator water level protection channels

f. Reactor coolant low flow protection channels

g. Impulse chamber pressure channels

h. Containment pressure

i. Steam pressure protection channels

The following conditions are required for these tests:

a. These tests may be performed at any unit power from cold shutdown to full power.

b. Before starting any of these tests with the unit at power, all redundant reactor trip channels associated with the function to be tested must be in the normal (untripped) mode in order to avoid spurious trips.

c. Setpoints are referenced in Precautions, Limitations and Setpoints portion of the manual.

d. Reference is made to Supplier(s) Systems Manual for systems description, static and dynamic testing. (To be supplied with the equipment).

e. Reference is made to Westinghouse's or Supplier's Block Diagrams which identify the functions provided in the instrument channels.

7.2.4 REFERENCES

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5 Corporation, WCAP 8255, January, 1974.

3. Katz, D. N., Solid State Logic Protection System Description, Westinghouse Electric Corporation, WCAP 7488-L, March, 1971 (Westinghouse NES Proprietary); and WCAP 7672, May, 1971 (Non-proprietary).

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7. Burnett, T. W. T., Reactor Protection System Diversity in Westinghouse Pressurized Water Reactors, Westinghouse Electric Corporation, WCAP-7306, April, 1969.

8. The Institute of Electrical and Electronic Engineers, Inc., IEEE Standard: Criteria for Protection Systems for Nuclear Power Generating Stations, IEEE Std. 279-1971.

9. The Institute of Electrical and Electronic Engineers, Inc., IEEE Trial Use Criteria for the Periodic Testing of Nuclear Power Generating Station Protection Systems, IEEE Std. 338-1971.

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35 Noise Tests, "WCAP 8892A June 1977 (Non Proprietary)

7.3 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM

In addition to the requirements for a reactor trip for anticipated abnormal transients, the facility is provided with adequate instrumentation and controls to sense accident situations and initiate the operation of necessary Engineered Safety Features. The occurrence of a limiting fault, such as a loss of coolant accident or a steam break, requires a reactor trip plus actuation of one or more of the Engineered Safety Features in order to prevent or mitigate damage to the core and Reactor Coolant System components, and insure Containment integrity.

In order to accomplish these design objectives the Engineered Safety Features system has proper and timely initiating signals which are supplied by the sensors, transmitters and logic components making up the various instrumentation channels of the Engineered Safety Features Actuation System.

7.3.1 DESCRIPTION

The Engineered Safety Features Actuation System senses selected unit parameters, determines whether or not predetermined safety limits are being exceeded and, if they are, combines the signals into logic matrices sensitive to combinations indicative of primary or secondary system boundary ruptures (Class III or IV faults). Once the required logic combination is completed, the system sends actuation signals to those engineered safety features components whose aggregate function best serves the requirements of the accident. The Engineered Safety Features Actuation System meets the requirements of Criteria 13 and 20 of the 1971 GDC.

7.3.1.1 <u>System Description</u>

The Engineered Safety Features Actuation System is a functionally defined system described in this section. The equipment which provides the actuation functions identified in Subdivision 7.3.1.1.1 is listed below and discussed in this section and the referenced WCAPs.

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1. Process Instrumentation
   (Reference 1)

2. Solid State Logic Protection System
   (Reference 3)

3. Engineered Safety Features Test Cabinet
   (Reference 4)

4. Manual Actuation Circuits

The Engineered Safety Features Actuation System consists of two discrete portions of circuitry: 1) An analog portion consisting of three to four redundant channels per parameter or variable to monitor various unit parameters such as the Reactor Coolant System and steam system pressures, temperatures and flows and Containment pressures; and 2) a digital portion consisting of two redundant logic trains which receive inputs from the analog protection channels and perform the needed logic to actuate the engineered safety features. Each digital train is capable of actuating the engineered safety features equipment required. The intent is that any single failure within the Engineered Safety Features Actuation System does not prevent system action when required.

tolerate a single failure without the loss of its core protective functions. This failure is limited to an active failure during the short term (injection) phase following a LOCA, or to an active or passive failure during the long-term (recirculation) phase. An active failure being defined as the failure of a powered component to act to accomplish its design functions. A passive failure is a structural failure.

The Safety Injection System is automatically actuated by a Safety Injection Signal as a consequence of one of the following events:

A. Low steam line pressure.

B. Low pressurizer pressure.

35 C. High Containment pressure.

D. Manual Actuation.

The SIS is electrically interlocked to start the injection mode on a Safety Injection Signal as follows:

A. The centrifugal charging pumps start. Simultaneously, pump suction valves from the RWST (1NV221A and 1NV222B), BIT suction and discharge Q223.3 valves (1N14A, 1N15B, 1N19A, 1N110B) open to provide a clear flow path from RWST to RCS. Simultaneously, normal charging path valves 1NV141A, 1NV142B, and 1NV244A, 1NV245B close as do the valves on the mini-flow line 1NV150B, 1NV151A. Valves 1N141A and 1N123A and 1N124B are closed to isolate the boron injection recirculation loop.

B. The SIP's and RHRP's start.

C. The normally open accumulator isolation valves (1N154A, 1N145B, 1N176A, and 1N188B) open if any have been closed.

The injection mode continues until the low level is reached in the refueling water storage tank (RWST). The water level in the RWST is measured by three separate channels of instrumentation each with read-outs on the main control board. Two-out-of-three logic from all three channels provides an alarm when the low level is reached in the RWST. At this point it is necessary to switch from the injection phase to the cold leg recirculation phase of operation. Additional two-out-of-three logic provides an alarm when the low-low level is
13 reached in the RWST indicating that only 10,000 gallons of water is available for removal from the tank. Switchover from the injection to the recirculation mode is accomplished manually by the operator. The switchover sequence (outlined in Table 6.3.2-3) is followed regardless of which power supply is available (offsite or emergency onsite). Controls for the Safety Injection System are grouped together on the main control board. Component position indication lights are also provided to verify that the function of a given switch has been completed.

In addition to the manual switchover, an automatic switchover is provided to
backup the operator. When the automatic switchover level is reached in the
RWST, two-out-of-three logic from all three channels automatically opens the
Containment sump valves (1NI184B and 1NI185A). The Containment sump valves Q223.3
13 are interlocked with the RWST isolation valves to the RHR pumps (1ND4B and
1ND19A) such that these isolation valves will close when the Containment sump
valves leave their seats. This automatic switchover provides an uninterrupted
flow of water to the RHR pumps.

There are four accumulator tanks in SIS. Each tank contains dilute boric
acid with a pressurized non-reactive cover gas (nitrogen) over it. Contents
of the tanks are used to flood the core following Reactor Coolant System
depressurization as a result of a LOCA. Water from only three tanks need to
5 be injected in order to partially cover the core. During normal plant operation Q212.9
each accumulator tank is isolated from the Reactor Coolant System by two check
valves in series. There is one motor operated valve in each accumulator tank
discharge line which may be used to isolate the accumulator from the rest of
the system. Each of these valves is interlocked to open completely within ten
seconds after either (a) the primary coolant system pressure exceeds a pre-
8 selected value (refer to the Technical Specifications) or (b) a safety Q223.39
injection signal has been initiated. Both signals are provided to the valves.
Diverse position indication is provided in the Control Room by means of two
diverse sensors mounted on each valve. An audible alarm is actuated by a
sensor on the valve when the valve is not in the fully open position. The
position and audible alarm are independent of the motor control center power.
Due to this diverse interlock and indication scheme, no local controls have
been provided in the control scheme to close these isolation valves from
outside the Control Room. However, means are available at the motor control Q212.9
center to close these valves under extremely critical circumstances.

7.4.1.6.1.2 Analysis

IEEE 279-1971 "Criteria for Protection Systems for Nuclear Power Generating
Stations", establishes minimum requirements for the reactor protective and
engineered safety features instrumentation and control systems. Conformance
with the applicable portions of IEEE 279, Section 4 is discussed in the Q223.3
following sections.

7.4.1.6.1.2.1 General Functional Requirements

5

The instrumentation and controls provided for the Safety Injection System
enable the operator to evaluate the system performance and detect malfunctions.
The Safety Injection System instrumentation essential to the system safety
function is designed for operation under the environmental conditions
specified in Section 3.11.

7.4.1.6.1.2.2 Single Failure Criterion

The instrumentation and controls required for Safety Injection System are
designed such that no single failure can prevent proper action at the system
level. Single failures considered include electrical faults (e.g., open,
shorted or grounded circuits) and physical events (e.g., fires, missiles)
resulting in mechanical damage. Compliance with single failure criterion is
accomplished by providing separation of the redundant elements electrically
and physically to achieve the required independence. Electrical separation is
assured through the provision of independent power supplies and the elimination
of electrical interconnection between redundant elements. A failure analysis
for this system is present in Table 6.3.2-7.

7.4.2 ANALYSIS

Hot shutdown is a stable unit condition, automatically reached following a unit shutdown. The hot shutdown condition can be maintained safely for an extended period of time. In the unlikely event that access to the Control Room is restricted, the unit can be safely kept at a hot shutdown until the Control Room can be re-entered.

The safety evaluation of the maintenance of a shutdown with these systems and associated instrumentation and controls has included consideration of the accident consequences that might jeopardize safe shutdown conditions. The accident consequences that are germane are those that would tend to degrade the capabilities for boration, adequate supply for auxiliary feedwater, and residual heat removal.

The results of the accident analyses are presented in Chapter 15. Of these the following produce the most severe consequences that are pertinent:

a. Uncontrolled Boron Dilution

b. Loss of Normal Feedwater

c. Loss of External Electrical Load and/or Turbine Trip

d. Loss of all A.C. Power to the Station Auxiliaries

It is shown by these analyses that safety is not adversely affected by these incidents with the associated assumptions being that the instrumentation and controls indicated in Section 7.4 are available to control and/or monitor shutdown. These available systems allow a maintenance of hot shutdown even under the accident conditions listed above which would tend toward a return to criticality or a loss of heat sink.

7.4.3 REFERENCES

a. McGuire Electrical Schematics

7.5 SAFETY RELATED DISPLAY INSTRUMENTATION

7.5.1 DESCRIPTION

Tables 7.5.1-1 and 7.5.1-2 list the information readouts provided to the operator to enable him to perform required manual safety functions, and to determine the effect of manual actions taken following a reactor trip due to a Condition II, III, or IV event, as defined in Chapter 15. The tables list the information readouts required to maintain the unit in a hot shutdown condition or to proceed to cold shutdown within the limits of the Technical Specifications. Reactivity control after Condition II and III faults is maintained by administrative sampling of the reactor coolant for boron to ensure that the concentration is sufficient to maintain the reactor subcritical. The display instrumentation needed to enable the operator to perform required manual safety functions for post-accident monitoring of Condition II and III faults involves the following seven parameters, for each one of which at least two channels of instrumentation are provided, which are presented in Table 7.5.1-1:

1. T_{cold} or T_{hot} (measured, wide range)
5 2. Pressurizer Water Level
3. System Pressure (wide range)
4. Containment Pressure
5. Steam Line Pressure
35 6. Steam Generator Water Level (narrow range)

Each of the above channels is either recorded, as shown in Table 7.5.1-1 or logged.

The display instrumentation needed to enable the operator to perform required manual safety functions for post-accident monitoring of Condition IV faults invloves the following six parameters, for each one of which at least two Q 223.6
5 channels of instrumentation are provided which are presented in Table 7.5.1-2:

1. Containment Pressure
2. RWST Water Level
3. Steam Generator Water Level (narrow range)
35 4. Steam Line Pressure
5. Pressurizer Water Level

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7.7 CONTROL SYSTEMS NOT REQUIRED FOR SAFETY

The general design objectives of the unit control systems are:

a. To establish and maintain power equilibrium between primary and secondary system during steady state unit operation;

b. To constrain operational transients so as to preclude unit trip and re-establish steady state unit operation;

c. To provide the reactor operator with monitoring instrumentation that indicates all required input and output control parameters of the systems and provides the operator the capability of assuming manual control of the system.

7.7.1 DESCRIPTION

The unit control systems described in this section perform the following functions:

a. Reactor Control System

   1. Enables the nuclear unit to accept a step load increase or decrease of 10 percent and a ramp increase or decrease of 5 percent per minute within the load range of 15 percent to 100 percent without reactor trip, steam dump, or pressurizer relief actuation, subject to possible xenon limitations.

   2. Maintains reactor coolant average temperature T_{avg} within prescribed limits by creating the bank demand signals for moving groups of full length rod cluster control assemblies during normal operation and operational transients. The T_{avg} control also supplies a signal to pressurizer water level control, and steam dump control.

b. Rod Control System

   1. Provides for reactor power modulation by manual or automatic control of full length control rod banks in a preselected sequence and for manual operation of individual banks.

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   2. Systems for Monitoring and Indicating

      Provide alarms to alert the operator if the required core reactivity shutdown margin is not available due to excessive control rod insertion.

      Display control rod position.

      Provide alarms to alert the operator in the event of control rod deviation exceeding a preset limit.

c. Unit Control System Interlocks

1. Prevent further withdrawal of the control banks when signal limits are approached that predict the approach of a DNBR limit or kw/ft limit.

2. Inhibit automatic turbine load change as required by the Nuclear Steam Supply System.

d Pressurizer Pressure Control

1. Maintains or restores the pressurizer pressure to the design pressure ± 35 psi (which is well within reactor trip and relief and safety valve actuation setpoint limits) following normal operational transients that induce pressure changes by control (manual or automatic) of heaters and spray in the pressurizer. Provides steam relief by controlling the power relief valves.

e. Pressurizer Water Level Control

1. Establishes, maintains, and restores pressurizer water level within specified limits as a function of the average coolant temperature. Changes in level are caused by coolant density changes induced by loading, operational, and unloading transients. Level changes are produced by means of charging flow control (manual or automatic) as well as by manual selection of letdown orifices. Maintaining coolant level in the pressurizer within prescribed limits by actuating the charging and letdown system thus provides control of the reactor coolant water inventory.

f. Steam Generator Water Level Control

1. Establishes and maintains the steam generator water level to within predetermined limits during normal operating transients.

2. The Steam Generator Water Level Control System also restores the
steam generator water level to within predetermined limits at
unit trip conditions. It regulates the feedwater flow rate such
35 that under operational transients the heat sink for the Reactor
Coolant System does not decrease below a minimum. Steam generator
water inventory control is manual or automatic through the use of
feedwater control valves.

g. Steam Dump Control

1. Permits the nuclear unit to accept a sudden loss of load without incurring reactor trip. Steam is dumped to the condenser and/or the atmosphere as necessary to accommodate excess power generation in the reactor during turbine load reduction transients.

2. Insures that stored energy and residual heat are removed following a reactor trip to bring the unit to equilibrium no load conditions without actuation of the steam generator safety valves.

3. Maintains the unit at no load conditions and permits a manually controlled cooldown of the unit.

h. In-Core Instrumentation

Provides information on the neutron flux distribution and on the core outlet temperatures at selected core locations.

7.7.1.1 Reactor Control System

The Reactor Control System enables the nuclear unit to follow load changes automatically including the acceptance of step load increase or decreases of 10 percent and ramp increases or decreases of 5 percent per minute within the load range of 15 percent to 100 percent without reactor trip, steam dump, or pressure relief - subject to possible xenon limitations. The system is also capable of restoring coolant average temperature to within the programmed temperature deadband following a change in load. Manual control rod operation may be performed at any time.

The Reactor Control System controls the reactor coolant average temperature by regulation of control rod bank position. The reactor coolant loop average temperatures are determined from hot leg and cold leg measurements in each reactor coolant loop. There is an average coolant temperature (T_{avg}) computed for each loop, where:

$$T_{avg} = \frac{T_{hot} + T_{cold}}{2}$$

The error between the programmed reference temperature (based on turbine impulse chamber pressure) and the highest of the average measured temperatures (which is processed through a lead-lag compensation unit) from each of the reactor coolant loops constitutes the primary control signal as shown in general on Figure 7.7.1-1 and in more detail on the functional diagrams shown in Figure 7.2.1-1, sheet 9. The system is capable of restoring coolant average temperature to the programmed value following a change in load. The programmed coolant temperature increases linearly with turbine load from zero power to the full power condition. The T_{avg} also supplies a signal to pressurizer level control and steam dump control and rod insertion limit monitoring.

The temperature channels needed to derive the temperature input signals for the reactor control system are fed from protection channels via isolation amplifiers.

An additional control input signal is derived from the reactor power versus turbine load mismatch signal. This additional control input signal improves system performance by enhancing response and reducing transient peaks.

The core axial power distribution is controlled during load follow maneuvers by changing (a manual operator action) the boron concentration in the reactor
35 coolant system. The control board $\Delta\phi$ displays (7.7.1.3.1) indicates the need for an adjustment in the axial power distribution. Adding boron to the reactor coolant will reduce T_{avg} and cause the rods (through the rod control system) to move toward the top of the core. This action will reduce power peaks in the bottom of the core. Likewise, removing boron from the reactor coolant will move the rods further into the core to control power peaks in the top of the core.

7.7.1.2 Rod Control System

7.7.1.2.1 Full Length Rod Control System

The full length Rod Control System receives rod speed and direction signals from the T_{avg} control system. The rod speed demand signal varies over the corresponding range of 3.75 to 45 inches per minute (6 to 72 steps/minute) depending on the magnitude of the input signal. The rod direction demand signal is determined by the positive or negative value of the input signal. Manual control is provided to move a control bank in or out at a prescribed fixed speed.

When the turbine load reaches approximately 15 percent of rated load, the operator may select the AUTOMATIC mode, and rod motion is then controlled by the Reactor Control System. A permissive interlock C-5 (See Table 7.7.1-1) derived from measurements of turbine impulse chamber pressure prevents automatic control when the turbine load is below 15 percent. In the AUTOMATIC mode, the rods are again withdrawn (or inserted) in a predetermined programmed sequence by the automatic programming equipment. The manual and automatic controls are further interlocked with the control interlocks (see Table 7.7.1-1).

The shutdown banks are always in the fully withdrawn position during normal operation, and are moved to this position at a constant speed by manual control prior to criticality. A reactor trip signal causes them to fall by gravity into the core. There are four shutdown banks.

The control banks are the only rods that can be manipulated under automatic control. Each control bank is divided into two groups to obtain smaller incremental reactivity changes per step. All rod cluster control assemblies in a group are electrically paralleled to move simultaneously. There is individual position indication for each rod cluster control assembly.

Power to rod drive mechanisms is supplied by two motor generator sets operating from two separate 600 volt, three-phase buses. Each generator is the synchronous type and is driven by a 150 hp induction motor. The AC power is distributed to the rod control power cabinets through the two series connected reactor trip breakers.

The variable speed rod drive programmer affords the ability to insert small amounts of reactivity at low speed to accomplish fine control of reactor coolant average temperature about a small temperature deadband, as well as furnishing control at high speed. A summary of the rod cluster control assembly sequencing characteristics is given below.

a. Two groups within the same bank are stepped such that the relative position of the groups does not differ by more than one step.

b. The control banks are programmed such that withdrawal of the banks is sequenced in the following order: control bank A, control bank B, control bank C, and control bank D. The programmed insertion sequence is the opposite of the withdrawal sequence, i.e., the last control bank withdrawn (bank D) is the first control bank inserted.

c. The control bank withdrawals are programmed such that when the first bank reaches a preset position, the second bank begins to move out simultaneously with the first bank. When the first bank reaches the top of the core, it stops, while the second bank continues to move toward its fully withdrawn position. When the second bank reaches a preset position, the third bank begins to move out, and so on. This withdrawal sequence continues until the unit reaches the desired power level. The control bank insertion sequence is the opposite.

d. Overlap between successive control banks is adjustable between 0 to 50 percent (0 to 115 steps), with an accuracy of $\pm$ 1 step.

e. Rod speeds for either shutdown banks or control banks are capable of being controlled between a minimum of 6 steps per minute and a maximum of 72 steps per minute.

7.7.1.3 Unit Control Signals for Monitoring and Indicating

7.7.1.3.1 Monitoring Functions Provided by the Nuclear Instrumentation System

The Nuclear Instrumentation System is described in Reference 2.

The power range channels are important because of their use in monitoring power distribution in the core within specified safe limits. They are used to measure power level, axial power imbalance, and radial power imbalance. These channels are capable of recording overpower excursions up to 200 percent of full power. Suitable alarms are derived from these signals as described below.

Basic power range signals are:

a. Total current from a power range detector (four such signals from separate detectors); these detectors are vertical and have an active length of 10 feet.

b. Current from the upper half of each power range detector (four such signals).

c. Current from the lower half of each power range detector (four such signals).

Derived from these basic signals are the following (including standard signal processing for calibration):

a. Indicated neutron flux (four such).

b. Indicated axial flux imbalance, derived from upper half flux minus lower half flux (four such).

Alarm functions derived are as follows:

a. Deviation (maximum minus minimum of four) in indicated nuclear power.

b. Upper radial tilt (maximum to average of four) on upper-half currents.

c. Lower radial tilt (maximum to average of four) on lower-half currents.

Provision is made to continuously record, on strip charts on the control board, the 8 ion chamber signals, i.e., upper and lower currents for each detector. Nuclear power and axial unbalance is selectable for recording as well. Indicators are provided on the control board for nuclear power and for axial power imbalance.

A comprehensive discussion of the Nuclear Instrumentation System can be found in Reference 2.

7.7.1.3.2 Control Rod Drive Position Indication System for Full and Part Length Rods

Two separate systems are provided to sense and display control rod position as described below:

a. Digital Rod Position Indication System

The digital rod position indication system is described in Reference 6. It measures the actual position of each full length rod using a detector which consists of 42 discrete coils mounted concent[illegible] with the rod drive pressure housing. The coils are located axially along the pressure housing on 3.75 inch spacing. They magnetically sense the entry and presence of the rod drive shaft through its center line. The coils are interlaced into two data channels, and are connected to the Containment electronics (Data A and B) by separate multi-conductor cables. Multiplexing is used to transmit the digital position signals from the Containment electronics to the control board display unit. The digital position signal is displayed on the main control board by light-emitting-diodes (LED) for each full length control rod. The one LED illuminated in the column shows the position for that particular rod. By employing two separate channels of information, the digital rod position indication system can continue to function (at reduced accuracy) when one channel fails.

Included in the system is a rod at bottom signal that operates a local alarm and a Control Room annunciator.

b. Demand Position Indication System

The demand position indication system counts pulses generated in the Rod Control System to provide a digital readout of the demanded bank position.

The demand position and digital rod position indication systems are separate systems; each serves as backup for the other. Operating procedures require the reactor operator to compare the demand and actual reading from the rod position indicating system so as to verify proper operation of the rod control system.

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7.7.1.3.3 Control Bank Rod Insertion Monitoring

When the reactor is critical, the normal indication of reactivity status in the core is the position of the control bank in relation to reactor power (as indicated by the Reactor Coolant System loop T) and coolant average temperature. These parameters are used to calculate insertion limits for the control banks. Two alarms are provided for each control bank.

a. The "low" alarm alerts the operator of an approach to the rod insertion limits requiring boron addition by following normal procedures with the Chemical and Volume Control System.

b. The "low-low" alarm alerts the operator to take immediate action to add boron to the Reactor Coolant System by any one of several alternate methods.

The purpose of the control bank rod insertion monitor is to give warning to the operator of excessive rod insertion. The insertion limit maintains sufficient core reactivity shutdown margin following reactor trip and provides a limit on the maximum inserted rod worth in the unlikely event of a hypothetical rod ejection, and limits rod insertion such that acceptable nuclear peaking factors are maintained. Since the amount of shutdown reactivity required for the design shutdown margin following a reactor trip increases with increasing power, the allowable rod insertion limits must be decreased (the rods must be withdrawn further) with increasing power. Two parameters which are proportional to power are used as inputs to the insertion monitor. These are the T between the hot leg and the cold leg, which is a direct function of reactor power, and T_{avg}, which is programmed as a function of power. The rod insertion monitor uses these parameters for each control rod bank as follows:

$$Z_{LL} = A(\Delta T)_{auct} + B(T_{avg})_{auct} + C$$

where

Z_{LL} = Maximum permissible insertion limit for affected control bank

$(\Delta T)_{auct}$ = Highest ΔT of all loops

$(T_{avg})_{auct}$ = Highest T_{avg} of all loops

A,B,C = Constants chosen to maintain $Z_{LL} \geq$ actual limit based on physics calculations

The control rod bank demand position (Z) is compared to Z_{LL} as follows:

If $Z - Z_{LL} \leq D$ a low alarm is actuated.

If $Z - Z_{LL} \leq E$ a low-low alarm is actuated.

Since the highest values of T_{avg} and ΔT are chosen by auctioneering, a conservatively high representation of power is used in the insertion limit calculation.

Actuation of the low alarm alerts the operator of an approach to a reduced shutdown reactivity situation. Administrative procedures require the operator to add boron through the Chemical and Volume Control System. Actuation of the low-low alarm requires the operator to initiate emergency boration procedures. The value for "E" is chosen such that the low-low alarm is normally actuated before the insertion limit is reached. The value for "D" is chosen to allow the operator to follow normal boration procedures. Figure 7.7.1-2 shows a block diagram representation of the control rod bank insertion monitor. The monitor is shown in more detail on the function diagrams shown in 7.2.1-1, sheet 9. In addition to the rod insertion monitor for the control banks, the unit computer, which monitors individual rod positions, provides an alarm system that is associated with the rod deviation alarm discussed below (7.7.1.3.4) to warn the operator if any shutdown rod cluster control assembly leaves the fully withdrawn position.

Rod insertion limits are established by:

a. Establishing the allowed rod reactivity insertion at full power consistent with the purposes given above.

b. Establishing the differential reactivity worth of the control rods when moved in normal sequence.

c. Established the change in reactivity with power level by relating power level to rod position.

d. Linearizing the resultant limit curve. All key nuclear parameters in this procedure are measured as part of the initial and periodic physics testing program.

Any unexpected change in the position of the control bank under automatic control, or a change in coolant temperature under manual control, provides a direct and immediate indication of a change in the reactivity status of the reactor. In addition, samples are taken periodically of coolant boron concentration. Variations in concentration during core life provide an additional check on the reactivity status of the reactor, including core depletion.

7.7.1.3.4 Rod Deviation Alarm

The demanded and measured rod position signals are displayed on the control board. They are also monitored by the unit computer which provides a visual printout and an audible alarm whenever an individual rod position signal deviates from the other rods in the bank by a preset limit. The alarm can be set with appropriate allowance for instrument error and within sufficiently narrow limits to preclude exceeding core design hot channel factors.

Figure 7.7.1-3 is a block diagram of the rod deviation comparator and Alarm System.

7.7.1.3.5 Rod Bottom Alarm

A rod bottom signal for the full length rods bistable in the Digital Rod Position System as described in Reference 6, is used to operate a control relay, which generates the **ROD BOTTOM ROD DROP** alarm.

7.7.1.4 <u>Unit Control System Interlocks</u>

The listing of the unit control system interlocks, along with the description of their derivations and functions, is presented in Table 7.7.1-1. It is noted that the designation numbers for these interlocks are preceeded by "C." The development of these logic functions is shown in the functional diagrams (Figure 7.2.1-1, sheets 9 to 16).

7.7.1.4.1 Rod Stops

Rod stops are provided to prevent abnormal power conditions which could result from excessive control rod withdrawal initiated by either a control system malfunction or operator violation of administrative procedures.

The C1, C2, C3, C4, and C5 control interlocks identified in Table 7.7.1-1 are rod stops. The C3 rod stop derived from overtemperature —T and the C4 rod stop, derived from overpower —T are also used for turbine runback, which is discussed below:

7.7.1.4.2 Automatic Turbine Load Runback

Automatic turbine load runback is initiated by an approach to an overpower or overtemperature condition. This prevents high power operation that might lead to an undesirable condition, which, if reached, is protected by reactor trip.

Turbine load reference reduction is initiated by either an overtemperature or overpower ΔT signal. Two-out-of-four coincidence logic is used.

A rod stop and turbine runback are initiated when

$$\Delta T > \Delta T_{rod\ stop}$$

for both the overtemperature and the over-power condition.

For either condition in general

$$\Delta T_{rod\ stop} = \Delta T_{setpoint} - B_p$$

where

$$B_p = \text{a setpoint bias}$$

where ΔT setpoint refers to the overtemperature ΔT reactor trip value and the overpower ΔT reactor trip value for the two conditions.

The turbine runback is continued until ΔT is equal to or less than $\Delta T_{rod\ stop}$. This function serves to maintain an essentially constant margin to trip.

7.7.1.4.3 Turbine Loading Stop

An interlock (C-16) is provided to limit turbine loading during a rapid return to power transient when a reduction in reactor coolant temperature is used to increase reactor power (through the negative moderator coefficient).
35 This interlock limits the drop in coolant temperature to exceed cooldown accident limits and preserves satisfactory steam generator operating conditions. Subsequent automatic turbine loading can begin after the interlock has been cleared by an increase in coolant temperature which is accomplished by reducing the boron concentration in the coolant.

7.7.1.5 Pressurizer Pressure Control

The reactor Coolant System pressure is controlled by using either the heaters (in the water region) or the spray (in the steam region) of the pressurizer plus steam relief for large transients. The electrical immersion heaters are located near the bottom of the pressurizer. A portion of the heater group is proportionally controlled to correct for small pressure variations. These variations are due to heat losses, including heat losses due to a small continuous spray. The remaining (backup) heaters are turned on when the pressurizer pressure controlled signal demands approximately 100 percent proportional heater power.

The spray nozzles are located on the top of the pressurizer. Spray is initiated when the pressure controller spray demand signal is above a given setpoint. The spray rate increases proportionally with increasing spray demand signal until it reaches a maximum value.

Steam condensed by the spray reduces the pressurizer pressure. A small continuous spray is normally maintained to reduce thermal stresses and thermal shock and to help maintain uniform water chemistry and temperature in the pressurizer.

Power relief valves limit system pressure for large positive pressure
transients. In the event of a large load reduction, not exceeding the
design unit load rejection capability, the pressurizer power operated
relief valves might be actuated for the most adverse conditions, e.g.,
the most negative Doppler coefficient, and the minimum incremental rod
worth. The relief capacity of the power operated relief valves is sized
large enough to limit the system pressure to prevent actuation of high
pressure reactor trip for the above condition. The power relief valves
also limit system pressure during cold start-up to avoid exceeding
35 reactor vessel stress limits, by opening the valves when the reactor
coolant system pressure exceeds setpoint limit and the system temperature
setpoint is below the vessel nil-ductility temperature limit.

A block diagram of the pressurizer pressure control system is shown on Figure 7.7.1-4.

7.7.1.6 <u>Pressurizer Water Level Control</u>

The pressurizer operates by maintaining a steam cushion over the reactor coolant. As the density of the reactor coolant adjusts to the various temperatures, the steam water interface moves to absorb the variations with relatively small pressure disturbances.

The water inventory in the Reactor Coolant System is maintained by the Chemical and Volume Control System. During normal plant operation, the charging flow varies to produce the flow demanded by the pressurizer water level controller. The pressurizer water level is programmed as a function of coolant average temperature, with the highest average temperature (auctioneered) being used. The pressurizer water level decreases as the load is reduced from full laod. This is a result of coolant contraction following programmed coolant temperature reduction from full power to low power. The programmed level is designed to match as nearly as possible the level changes resulting from the coolant temperature changes.

To control pressurizer water level during startup and shutdown operatins, the charging flow is manually regulated from the main Control Room.

A block diagram of the pressurizer water level control system is shown on Figure 7.7.1-5.

7.7.1.7 <u>Steam Generator Water Level Control</u>

Each steam generator is equipped with a three element feedwater flow con-
troller which maintains a programmed water level which is a function of
35 neutron flux. The three element feedwater controller regulates the feed-
water valve by continuously comparing the feedwater flow signal, the water
level signal, the programmed level and the pressure compensated steam flow
signal. In addition, by using turbine driven main feedwater pumps, the
feedwater pump speed is varied to maintain a programmed pressure differential
between the steam header and the feed pump discharge header. The speed
controller continuously compares the actual ΔP with a programmed ΔP_{ref}, which
is a linear function of steam flow. Continued delivery of feedwater to the
steam generators is required as a sink for the heat stored and generated in
the reactor following a reactor trip and turbine trip. An override signal
closes the feedwater valves when the average coolant temperature is below a

given temperature and the reactor has tripped. Manual override of the feedwater control system is available at all times. When the nuclear plant is operating at very low power levels (as during startup), the steam and feed-water flow signals will not be usable for control. Therefore, a secondary automatic control system is provided for operation at low power. This system 35 uses the steam generator water level and nuclear power signals in a feed forward control scheme to position a bypass valve which is in parallel with the main feedwater regulating valve. Switchover from the bypass feedwater control system (low power) to the main feedwater control system is initiated by the operator at approximately 25 percent power.

A block diagram of the steam generator water level control system is shown in Figures 7.7.1-6 and 7.7.1-7.

7.7.1.8 Steam Dump Control

The steam dump system is designed to accept a 100 percent loss of net load without tripping the reactor.

The automatic steam dump system is able to accommodate this abnormal load rejection and to reduce the effects of the transient imposed upon the Reactor Coolant System. By bypassing main steam directly to the condenser, an artificial load is thereby maintained on the primary system. In the event load rejection exceeds 50 percent, main steam is also dumped to the atmosphere. The Rod Control System can then reduce the reactor temperature to a new equilibrium value without causing overtemperature and/or overpressure conditions.

If the difference between the reference $T_{avg}(T_{ref})$ based on turbine impulse chamber pressure and the lead/lag compensated auctioneered T_{avg} exceeds a predetermined amount, and the interlock mentioned below is satisfied, a demand signal actuates the steam dump to maintain the Reactor Coolant System temperature within control range until a new equilibrium condition is reached.

To prevent actuation of steam dump on small load perturbations, an independent load rejection sensing circuit is provided. This circuit senses the rate of decrease in the turbine load as detected by the turbine impulse chamber pressure. It is provided to unblock the dump valves when the rate of load rejection exceeds a preset value corresponding to a 10 percent step load decrease or a sustained ramp load decrease of 5 percent/minute.

A block diagram of the steam dump control system is shown on Figure 7.7.1-8.

7.7.1.8.1 Load Rejection Steam Dump Controller

This circuit prevents large increases in reactor coolant temperature follow-
25 ing a large, sudden load decrease or a turbine trip without a reactor trip.
The error signal is a difference between the lead/lag compensated auctioneered T_{avg} and the reference T_{avg} is based on turbine impulse chamber pressure.

The T_{avg} signal is the same as that used in the Reactor Coolant System. The lead/lag compensation for the T_{avg} signal is to compensate for lags in the plant thermal response and in valve positioning. Following a sudden load decrease, T_{ref} is immediately decreased and T_{avg} tends to increase, thus generating an immediate demand signal for steam dump. Since control rods are available, in this situation steam dump terminates as the error comes within the maneuvering capability of the control rods.

7.7.1.8.2 Plant Trip Steam Dump Controller

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35 Following a reactor trip, the load rejection steam dump controller is defeated and the plant trip steam dump controller becomes active. Since control rods are not available in this situation, the demand signal is the error signal between the lead/lag compensated auctioneered T_{avg} and the no load reference T_{avg}. When the error signal exceeds a predetermined setpoint, the dump valves are tripped open in a prescribed sequence. As the error signal reduces in magnitude indicating that the Reactor Coolant System T_{avg} is being reduced toward the reference no-load value, the dump valves are modulated by the plant

trip controller to regulate the rate of removal decay heat and thus gradually establish the equilibrium hot shutdown condition.

35 Following a reactor trip the steam dump capacity requirement is only that necessary to maintain steam pressure below the steam generator relief valve setpoint (≈40 percent capacity to the condenser); therefore, only the first two groups of valves are opened. The error signal determines whether a group is to be tripped open or modulated open. In either case, they are modulated when the error is below the trip-open setpoints.

7.7.1.8.3 Steam Header Pressure Controller

Residual heat removal is maintained by the steam generator pressure controller (manually selected) which controls the amount of steam flow to the condensers. This controller operates a portion of the same steam dump valves which goes to the condensers and which are used during the initial transient following
25 a reactor trip.

7.7.1.9 In-Core Instrumentation

The In-Core Instrumentation System consists of Chromel-Alumel thermocouples at fixed core outlet positions, movable miniature neutron detectors used by the flux mapping system which can be positioned at the center of selected fuel assemblies, anywhere along the length of the fuel assembly vertical axis. The basic system for insertion of these detectors is shown in Figure 7.7.1-9. Sections 1 and 2 of Reference 5 outline the In-Core Instrumentation System in more detail.

7.7.1.9.1 Thermocouples

Chromel-Alumel thermocouples are threaded into guide tubes that penetrate the reactor vessel head through seal assemblies, and terminate at the exit flow end of the fuel assemblies. The thermocouples are provided with two primary seals, a conoseal and swage type seal from conduit to head. The thermocouples are supported in guide tubes in the upper core support assembly. Thermocouple readings are monitored by the computer with backup readout provided by a precision indicator with manual point selection located in the Control Room. Information from the in-core instrumentation is available even if the computer is not in service.

7.7.1.9.2 Movable Neutron Flux Detector Drive System

Miniature fission chamber detectors can be remotely positioned in retractable guide thimbles to provide flux mapping of the core. See Reference 5 for neutron flux detector parameters. The stainless steel detector shell is welded to the leading end of helical wrap drive cable and to stainless steel sheathed coaxial cable. The retractable thimbles, into which the miniature detectors are driven, are pushed into the reactor core through conduits which extend from the bottom of the reactor vessel down through the concrete shield area and then up to a thimble seal table.

The thimbles are closed at the leading ends, are dry inside, and serve as the pressure barrier between the reactor water pressure and the atmosphere.

Mechanical seals between the retractable thimbles and the conduits are provided at the seal line. During reactor operation, the retractable thimbles are stationary. They are extracted downward from the core during refueling to avoid interference within the core. A space above the seal line is provided for the retraction operation.

The drive system for the insertion of the miniature detectors consists basically of drive assemblies, five path rotary transfer assemblies, and ten path rotary transfer assemblies, as shown in Figure 7.7.1-9. These assemblies are described in Reference 5. The Drive System pushes hollow helical wrap drive cables into the core with the miniature detectors attached to the leading ends of the cables and small diameter sheathed coaxial cables threaded through the hollow centers back to the ends of the drive cables. Each drive assembly consists of a gear motor which pushes a helical wrap drive cable and a detector through a selective thimble path by means of a special drive box and includes a storage device that accommodates the total drive cable length.

The leakage detection and gas purge provisions are discussed in Reference 5.

Manual isolation valves (one for each thimble) are provided for closing the thimbles. When closed, the valve forms a 2500 psig barrier. The manual isolation valves are not designed to isolate a thimble while a detector/drive cable is inserted into the thimble. The detector/drive cable must be retracted to a position above the isolation valve prior to closing the valve.

A small leak would probably not prevent access to the isolation valves and thus a leaking thimble could be isolated during a hot shutdown.

A large leak might require cold shutdown for access to the isolation valve.

7.7.1.9.3 Control and Readout Description

The control and readout system provides means for inserting the miniature neutron detectors into the reactor core and withdrawing the detectors while plotting neutron flux versus detector position. The control system consists of two sections, one physically mounted with the drive units, and the other contained in the Control Room. Limit switches in each transfer device provide feedback of path selection operation. Each gear box drives an encoder for position feedback. One five path operation selector is provided for each drive unit to insert the detector in one of five functional modes of operation. A ten path rotary transfer assembly is a transfer device that is used to route a detector into any one of up to ten selectable paths. A common path is provided to permit cross calibration of the detectors.

The Control Room contains the necessary equipment for control, position indication, and flux recording for each detector. Additional panels are provided for such features as drive motor controls, core path selector switches, plotting and gain controls.

Flux mapping is accomplished by selecting (by panel switches) flux thimbles in given fuel assemblies at various core quadrant locations. The detectors are driven to the top of the core and stopped automatically. An X-Y plot (position versus flux level) is initiated with the slow withdrawal of the detectors through the core from the top to a point below the bottom of the core. In

7.7.1.14.3 Alarms

Annunciator alarms for filter malfunction, high temperature and fire alarm are provided.

9 Alarm bells are provided at each end of the fuel pool to indicate loss of exhaust fan operation. Q310.13

7.7.1.14.4 Bypass

A dampered bypass is provided around the fuel pool exhaust filter train for use during normal operation. Bypass position status is indicated on the
20 HVAC panel in the Control Room.

Refer to Table 9.4.2-2, page 4, "Exceptions and Comments," C-2-i, for a complete description of the filter bypass.

7.7.2 ANALYSIS

The Unit Control Systems are designed to assure high reliability in any anticipated operational occurrences. Equipment used in these systems is designed and constructed with a high level of reliability.

Proper positioning of the control rods is monitored in the Control Room by bank arrangements of the individual position columns for each rod cluster control assembly. A rod deviation alarm alerts the operator of a deviation of one rod cluster control assembly from the other rack in that bank position. There are also insertion limit monitors with visual and audible annunciation. A rod bottom alarm signal is provided to the Control Room for each full length rod cluster control assembly. Four excore long ion chambers also detect asymmetrical flux distribution indicative of rod misalignment.

Overall reactivity control is achieved by the combination of soluble boron and rod cluster control assemblies. Long term regulation of core reactivity is accomplished by adjusting the concentration of boric acid in the reactor coolant. Short term reactivity control for power changes is accomplished by the unit control systems which automatically move rod cluster control assemblies. The input signals to the unit control systems include neutron flux, coolant temperature, and turbine load.

The axial core power distribution is controlled by moving the control rods through changes in reactor coolant system boron concentration. Adding
35 boron causes the rods to move out thereby reducing the amount of power in the bottom of the core, this allows power to redistribute toward the top of the core. Reducing the boron concentration causes the rods to move into the core thereby reducing the power in the top of the core, the result redistributes power towards the bottom of the core.

7.7.2.2 Response Considerations of Reactivity

Reactor shutdown with control rods is completely independent of the control functions since the trip breakers interrupt power to the full length rod drive mechanisms regardless of existing control signals. The design is such that the system can withstand accidental withdrawal of control groups or unplanned dilution of soluble boron without exceeding acceptable fuel design limits. The design meets the requirements of criteria 25 of the 1971 GDC.

No single electrical or mechanical failure in the Rod Control System can cause the accidental withdrawal of a single rod cluster control assembly from the partially inserted bank at full power operation. The operator can deliberately withdraw a single rod cluster control assembly in the control bank; this feature is necessary in order to retrieve a rod, should one be accidentally dropped. In the extremely unlikely event of simultaneous electrical failures which could result in single RCCA withdrawal, rod deviation is displayed on the unit annunciator, and the individual rod position readouts indicate the relative positions of the rods in the bank. Withdrawal of a single rod cluster control assembly by operator action, whether deliberate or by a combination of errors, results in activation of the same alarm and the same visual indications.

Each bank of control and shutdown rods in the system is divided into two groups of up to 4 or 5 mechanisms each. The rods comprising a group operate in parallel through multiplexing thyristors. The two groups in a bank move sequentially such that the first group is always within one step of the second group in the bank. A definite schedule of actuation or deactuation of the stationary gripper, moveable gripper, and lift coils of a mechanism is required to withdraw the rod cluster control assembly attached to the mechanism. Since the four stationary grippers, moveable gripper, and lift coils associated with the rod cluster control assemblies of a rod group are driven in parallel, any single failure which could cause rod withdrawal would affect a minimum of one group of rod cluster control assemblies. Mechanical failures are in the direction of insertion, or immobility.

The identified multiple failure involving the least number of components consists of open circuit failure of the proper two-out-of sixteen wires connected to the gate of the lift coil thyristors. The probability of open wire (or terminal) failure is 0.016×10^{-6} per hour by MIL-HDB217A. These wire failures would have to be accompanied by failure, or disregard, of the indications mentioned above. The probability of this occurrence is therefore too low to be significant.

Concerning the human element, to erroneously withdraw a single rod cluster control assembly, the operator has to improperly set the BANK SELECTOR switch, the LIFT COIL DISCONNECT switches, and the IN-OUT-HOLD switch. In addition, the three indications have to be disregarded or ineffective. Such series of errors requires a complete lack of understanding and administrative control. A probability number cannot be assigned to a series of errors such as these.

The Rod Position Indication System provides direct visual displays of each control rod assembly position. The unit computer alarm is actuated for deviation of rods from their banks. In addition a rod insertion limit monitor provides an audible and visual alarm to warn the operator that an abnormal condition is approaching due to dilution. The low-low insertion limit alarm alerts the operator to follow emergency boration procedures. The facility reactivity control systems are such that acceptable fuel damage limits are not exceeded even in the event of a single malfunction of either system.

An important feature of the Rod Control System is that insertion is provided by gravity fall of the rods.

In all analyses involving reactor trip, the single, highest worth rod cluster control assembly is postulated to remain untripped in its full out position.

One way of detecting a stuck control rod assembly is available from the actual rod position information displayed on the control board. The control
35 board position readouts, one for each full length rod, give the operator the actual position of the rod in steps. The indications are grouped by banks (e.g., CONTROL BANK A, CONTROL BANK B, etc.) to indicate to the operator the deviation of one rod with respect to other rods in a bank. This serves as a means to identify rod deviation.

The computer monitors the actual position of all rods. Should a rod be misaligned from the other rods in that bank by more than 15 inches, the rod deviation alarm is actuated.

Misaligned rod cluster control assemblies are also detected and alarmed in the Control Room via the flux tilt monitoring portion of the Nuclear Instrumentation System which is independent of the unit computer.

Isolated signals derived from the Nuclear Instrumentation System are compared with one another to determine if a preset amount of deviation of average power level has occurred. Should such a deviation occur, the comparator output operates a bistable unit to actuate a control board annunciator. This alarm alerts the operator to a power imbalance caused by a misaligned rod. By use of individual rod position readouts, the operator can determine the deviating control rod and take corrective action. The design of the plant control systems meets the requirements of criteria 23 of 10CFR50 Appendix A.

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The Boron Recycle System can compensate for all xenon burnout reactivity transients without exception.

The Rod Control System can compensate for xenon burnout reactivity transients over the allowed range of rod travel. Xenon burnout transients of larger magnitude must be accommodated by boration or by reactor trip (which eliminates the burnout).

The Boron Recycle System is not used to compensate for the reactivity effects of fuel/water temperature changes accompanying power level changes.

The Rod Control System can compensate for the reactivity effect of fuel/water temperature changes accompanying power level changes over the full range from full load to no load at the design maximum load uprate.

Automatic control of the rods is limited to the range of approximately 15 percent to 100 percent of rating.

The Boron Recycle System maintains the reactor in the cold shutdown state irrespective of the disposition of the control rods.

7.7.2.3 Step Load Changes Without Steam Dump

The Reactor Control System restores equilibrium conditions, without a trip, following a plus or minus 10 percent step change in load demand, over the 15 to 100 percent power range for automatic control. Steam dump is blocked for load decrease less than or equal to 10 percent. A load demand greater than full power is prohibited by the turbine control load limit devices.

The Reactor Control System minimizes the reactor coolant average temperature deviation during the transient within a given value and restores average temperature to the programmed setpoint. Excessive pressurizer pressure variations are prevented by using spray and heaters and power relief valves in the pressurizer.

The control system must limit nuclear power overshoot to acceptable values following a 10 percent increase in load to 100 percent.

7.7.2.4 Loading and Unloading

Ramp loading and unloading of 5 percent per minute can be accepted over the 15 to 100 percent power range under automatic control without tripping the unit. The function of the control system is to maintain the coolant average temperature as a function of turbine-generator load.

The coolant average temperature increases during loading and causes a continuous insurge to the pressurizer as a result of coolant expansion. The sprays limit the resulting pressure increase. Conversely, as the coolant average temperature is decreasing during unloading, there is a continuous outsurge from the pressurizer resulting from coolant contraction. The pressurizer water level is programmed such that the water level is above the setpoint for heater cutout during the loading and unloading transients. The primary concern during loading is to limit the overshoot in nuclear power and to provide sufficient margin in the overtemperature ΔT setpoint.

The automatic load controls are designed to adjust the unit generation to match load requirements within the limits of the unit capability and licensed rating.

35 During rapid loading transients, a drop in reactor coolant temperature is

sometimes used to increase core power. This mode of operation is applied when the control rods are not inserted deep enough into the core to supply all the reactivity requirements of the rapid load increase (the boron control system is relatively ineffective for rapid power charges). The reduction in temperature is initiated by continued turbine loading past the point where the control rods are completely withdrawn from the core. The temperature drop is recovered and nominal conditions restored by a boron dilution operation.

Excessive drops in coolant temperature are prevented by interlock C-16. This interlock circuit monitors the auctioneered low coolant temperature indications and the programmed reference temperature which is a function of turbine impulse pressure and causes a turbine loading stop when the decreased temperature reaches the setpoints.

The core axial power distribution is controlled during the reduced temperature return to power by placing the control rods in the manual mode when the $\Delta\phi$ operating limits are approached. Placing the rods in manual will stop further changes in $\Delta\phi$ and it will also initiate the required drop in coolant temperature. Normally power distribution control is not required during a rapid power increase and the rods will proceed, under the automatic rod control system, to the top of the core. The bite position is reestablished at the end of the transient by decreasing the coolant boron concentration.

7.7.2.5 <u>Load Rejection Furnished By Steam Dump System</u>

35 When a load rejection occurs and the difference between the required temperature setpoint of the Reactor Coolant System and the actual average temperature exceeds a predetermined amount, a signal actuates the steam dump to maintain the Reactor Coolant System temperature within control range until a new equilibrium condition is reached.

The reactor power is reduced at a rate consistent with the capability of the Rod Control System. Reduction of the reactor power is automatic. The steam dump flow reduction is as fast as rod cluster control assemblies are capable of inserting negative react vity.

The Rod Control System can then reduce the reactor temperature to a new equilibrium value without causing overtemperature and/or overpressure conditions. The steam dump steam flow capacity is 85 percent of full load steam flow at full load steam pressure.

The steam dump flow reduces proportionally as the control rods act to reduce the average coolant temperature. The artificial load is therefore removed as the coolant average temperature is restored to its programmed equilibrium value.

The dump valves are modulated by the reactor coolant average temperature signal. The required number of steam dump valves can be tripped quickly to stroke full open or modulate, depending upon the magnitude of the temperature error signal resulting from loss of load.

7.7.2.6 Reactor Trip

25 The unit is operated with a programmed average temperature as a function of load, with the full load average temperature significantly greater than the equivalent saturation pressure of the safety valve setpoint. The thermal capacity of the Reactor Coolant System is greater than that of the secondary system, and because the full load average temperature is greater than the no load temperature, a heat sink is required to remove heat stored in the reactor coolant to prevent actuation of steam generator safety valves for a trip from full power. This heat sink is provided by the combination of controlled release of steam to the condenser and by makeup of cold feedwater to the steam generators.

The Steam Dump System is controlled by the reactor coolant average temperature signal whose setpoint values are programmed as a function of turbine load. Actuation of the steam dump is rapid to prevent actuation of the steam generator safety valves. With the dump valves open, the average coolant temperature starts to reduce quickly to the no load setpoint. A direct feedback of temperature acts to proportionally close the valves to minimize the total amount of steam which is bypassed.

25 Following a reactor trip, the feedwater flow is cut off when the average coolant temperature decreases below a given temperature or when the steam generator water level reaches a given high level.

Additional feedwater makeup is then controlled manually to restore and maintain steam generator water level while assuring that the reactor coolant temperature is at the desired value. Residual heat removal is maintained by the steam header pressure controller (manually selected) which controls the amount of steam flow to the condensers. This controller operates a portion of the same steam dump valves to the condensers which are used during
25 the initial transient following a reactor trip.

The pressurizer pressure and level fall rapidly during the transient because of coolant contraction.

25

If heaters become uncovered following the trip, the Chemical and Volume Control System provides full charging flow to restore water level in the pressurizer. Heaters are then turned on to restore pressurizer pressure to normal.

The steam dump and feedwater control systems are designed to prevent the average coolant temperature from falling below the programmed no load temperature following the trip to ensure adequate reactivity shutdown margin.

7.7.3 REFERENCES

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4. Loving, J. J, In-Core Instrumentation (Flux-Mapping System and Thermocouples), Westinghouse Electric Corporation, WCAP-7607, July, 1971.

5. Blanchard, A. E. and Calpin, J. E., Digital Rod Position Indication, Westinghouse Electric Corporation, WCAP-8014, December, 1972.

6. McGuire Electrical Schematics

Table 7.2.1-1 (Page 2 of 4)

List of Reactor Trips

Reactor Trip	Coincident Logic	Interlocks	Comments
7. Overpower ΔT	2/4	No interlocks	
8. Pressurizer low pressure	2/4	Interlocked with P-7	Blocked below P-7
9. Pressurizer high pressure	2/4	No interlocks	
10. Pressurizer high water level	2/3	Interlocked with P-7	Blocked below P-7
11. Low reactor coolant flow	2/3 in any loop	Interlocked with P-7 and P-8	Low flow in one loop will cause a reactor trip when above P-8 and a low flow in two loops will cause a reactor trip when above P-7. Blocked below P-7
35 12. Reactor coolant pump under voltage	2/4	Interlocked with P-7	Low voltage to RCP motors permitted below P-7

Table 7.2.1-1 (Page 3 of 4)

List of Reactor Trips

	Reactor Trip	Coincident Logic	Interlocks	Comments
	13. Reactor coolant pump bus underfrequency	2/4	Interlocked with P-7	Under frequency on 2 buses will trip all reactor coolant pump breakers and cause reactor trip; reactor trip blocked below P-7.
35	14. Low-low steam generator water level	2/4 in any loop	No interlocks	
	15. Safety injection signal	Coincident with actuation of safety injection	No interlocks	(See Section 7.3 for Engineered Safety Features actuation conditions)
25				

Table 7.2.1-1 (Page 4 of 4)

List of Reactor Trips

Reactor Trip	Coincident Logic	Interlocks	Comments
35 16. Manual	1/2	No interlocks	

35

Table 7.2.1-2 (1 of 2)

Protection System Interlocks

Designation	Derivation	Function
I	POWER ESCALATION PERMISSIVES	
P-6	1/2 Neutron flux (intermediate-range) above setpoint	Allows manual block of source range reactor trip
	2/2 Neutron flux (intermediate-range) below setpoint	Defeats the block of source range reactor trip
P-10	2/4 Neutron flux (power-range) above setpoint	Allows manual block of power range (low setpoint) reactor trip
		Allows manual block of intermediate range reactor trip and intermediate range rod stops (C-1)
		Blocks source range reactor trip (de-energizes source range high voltage for equipment protection)
	3/4 Neutron flux (power-range) below setpoint	Defeats the block of power range (low setpoint) reactor trip
		Defeats the block of intermediate range reactor trip and intermediate range rod stops (C-1)
		Input to P-7
II	BLOCKS OF REACTOR TRIPS	
P-7 5 25	3/4 Neutron flux (power-range) below setpoint (from P-10) and 2/2 Turbine impulse chamber pressure below setpoint (from P-13)	Absence of signal blocks reactor trip on: Low flow in more than one loop, under voltage, underfrequency, pressurizer low pressure, and pressurizer high level

Table 7.2.1-1 (Page 4 of 4)

List of Reactor Trips

Reactor Trip	Coincident Logic	Interlocks	Comments
35 16. Manual	1/2	No interlocks	

35

Table 7.2.1-2 (1 of 2)

Protection System Interlocks

	Designation	Derivation	Function
	I	POWER ESCALATION PERMISSIVES	
	P-6	1/2 Neutron flux (intermediate-range) above setpoint	Allows manual block of source range reactor trip
		2/2 Neutron flux (intermediate-range) below setpoint	Defeats the block of source range reactor trip
	P-10	2/4 Neutron flux (power-range) above setpoint	Allows manual block of power range (low setpoint) reactor trip
			Allows manual block of intermediate range reactor trip and intermediate range rod stops (C-1)
			Blocks source range reactor trip (de-energizes source range high voltage for equipment protection)
		3/4 Neutron flux (power-range) below setpoint	Defeats the block of power range (low setpoint) reactor trip
			Defeats the block of intermediate range reactor trip and intermediate range rod stops (C-1)
			Input to P-7
	II	BLOCKS OF REACTOR TRIPS	
5 25	P-7	3/4 Neutron flux (power-range) below setpoint (from P-10) and 2/2 Turbine impulse chamber pressure below setpoint (from P-13)	Absence of signal blocks reactor trip on: Low flow in more than one loop, under voltage, underfrequency, pressurizer low pressure, and pressurizer high level

Table 7.2.1-2 (2 of 2)

Protection System Interlocks

	Designation	Derivation	Function
	P-8	2/4 Neutron flux (power-range) above setpoint	Absence of signal blocks reactor trip on low flow
35 20	P-13	2/2 Turbine impulse chamber pressure below setpoint	Input to P-7

Table 7.2.1-3

Reactor Protection System Instrument Accuracies

Rev.	Reactor Trip Signal	System Accuracy	See Note
	1. Power-range high neutron flux	±1 percent of full power	
	2. Intermediate-range high neutron flux	±5 percent of full scale ±1 percent of full scale from 10^{-4} to 10^{-3} amperes	(a) (a)
	3. Source-range high neutron flux	±5 percent of full scale	(a)
	4. Power-range high positive neutron flux rate	±5 percent	(a)
	5. Power-range high negative neutron flux rate	±5 percent	(a)
	6. Overtemperature ΔT	±3.2°F	
	7. Overpower ΔT	±2.7°F	
	8. Pressurizer low pressure	±18 psi	
	9. Pressurizer high pressure	±14 psi	
	10. Pressurizer high water level	±2.3 percent of full range Δp between taps at design temperature and pressure.	
	11. Low reactor coolant flow	±2.5 percent of full flow within range of 70 percent to 100 percent of full flow	(a)
35 5	12. Reactor coolant pump under voltage	±5 percent of rated voltage	
	13. Reactor coolant pump bus underfrequency	±0.1 Hz	
35	14. Low-low steam generator water level	±2.3 percent of Δp signal over pressure range of 700 to 1200 psig	

35 (a) Reproducibility

Table 7.2.1-4 (1 of 5)

Reactor Trip Correlation

	Trip (a)	Accident (b)	Tech. Spec. (c)
21	1) Power-Range High Neutron Flux Trip (Low Setpoint)	1) Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From A Subcritical Condition (15.2.1)	2.2
		2) Excessive Heat Removal Due to Feedwater System Malfunctions (15.2.10)	
		3) Rupture of a Control Rod Drive Mechanism Housing (15.4.6)	
21	2) Power-Range High Neutron Flux Trip (High Setpoint)	1) Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From A Subcritical Condition (15.2.1)	2.2
		2) Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (15.2.2)	
		3) Startup of an Inactive Reactor Coolant Loop (15.2.6)	
		4) Excessive Heat Removal Due to Feedwater System Malfunctions (15.2.10)	
		5) Excessive Load Increase Incident (15.2.11)	
		6) Accidental Depressurization of the Main Steam System (15.2.13)	
		7) Major Secondary System Pipe Rupture (15.4.2)	
		8) Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection) (15.4.6)	

Table 7.2.1-4 (4 of 5)

Reactor Trip Correlation

	Trip (a)	Accident (b)	Tech. Spec. (c)
21	11) Pressurizer High Water Level Trip	1) Uncontrolled Rod Cluster Control Assembly Bank at Power (15.2.2)	2.2
		2) Loss of External Electrical Load and/or Turbine Trip (15.2.7)	
21	12) Low Reactor Coolant Flow	1) Partial Loss of Forced Reactor Coolant Flow (15.2.5)	2.2
		2) Loss of Off-Site Power to the Station Auxiliaries (Station Blackout) (15.2.9)	
		3) Complete Loss of Forced Reactor Coolant Flow (15.3.4)	
21 35 5	13) Reactor Coolant Pump Under voltage Trip	Complete Loss of Forced Reactor Coolant Flow (15.3.4)	2.2
21	14) Reactor Coolant Pump Bus Underfrequency Trip	Complete Loss of Forced Reactor Coolant Flow (15.3.4)	2.2
35	15) Low-low Steam Generator Water Level Trip	Loss of Normal Feedwater (15.2.8)	2.2
35 25	16) Turbine Trip upon a Reactor Trip	1) Loss of Off-Site Power to the Station Auxiliaries (Station Blackout) (15.2.9)	2.2
		2) Excessive Heat Removal due to Feedwater System Malfunction	

Table 7.2.1-4 (5 of 5)

Reactor Trip Correlation

	Trip (a)	Accident (b)	Tech. Spec. (c)
25			
35	17) Safety Injection Signal Actuation Trip	Accidental Depressurization of the Main Steam System (15.2.13)	see note e
35	18) Manual Trip	Available for all Accidents (Chapter 15)	see note d

TABLE 7.3.1-1

Instrumentation Operating Condition For Engineered Safety Features

	NO.	FUNCTIONAL UNIT	NO. OF CHANNELS	NO. OF CHANNELS TO TRIP
	1	SAFETY INJECTION		
	a.	Manual	2	1
	b.	High Containment Pressure	3	2
35	c.	Low Steam Line Pressure	3/Steam line	2/Steam line in any steam line.
	d.	Pressurizer Low Pressure*	4	2
35				
	2	CONTAINMENT SPRAY		
	a.	Manual	2	1
	b.	Containment Pressure High-High	4	2

* Permissible bypass if reactor coolant pressure less than 1900 psig.

TABLE 7.3.1-2

Instrument Operating Conditions For Isolation Functions

NO.	FUNCTIONAL UNIT	NO. OF CHANNELS	NO. OF CHANNELS TO TRIP
1.	CONTAINMENT ISOLATION		
a.	Automatic Safety Injection (Phase A)	See Item No. 1 (b) through (d) of Table 7.3.1-1	
b.	Containment Pressure (Phase B)	See Item No. 2 (b) of Table 7.3.1-1	
c.	Manual Phase A Phase B	 2 See Item No. 2 (a) of Table 7.3.1-1	 1
2.	STEAM LINE ISOLATION		
35 a.	Low Steam Line Pressure	See Item No. 1 (c) of Table 7.3.1-1	
b.	Containment Pressure High-High	See Item No. 2 (b) of Table 7.3.1-1	
c.	Manual	1/loop	1/loop
35 d.	High Negative Steam Line Pressure Rate	3/Steam line in any steam line	2/Steam line
3.	FEEDWATER LINE ISOLATION		
a.	Safety Injection	See Item No. 1 of Table 7.3.1-1	

TABLE 7.3.1-3 (1 of 2)

Interlocks For Engineered Safety Features Actuation System

	Designation	Input	Function Performed
	P-4	Reactor trip	Actuates turbine trip
			Closes main feedwater falves on T_{avg} below setpoint
			Prevents opening of main feedwater valves which were closed by safety injection or high steam generator water level
			Allows manual reset/block of the automatic reactuation of safety injection
35			Blocks steam dump control via load rejection T_{avg} controller.
			Makes steam dump valves available for either tripping or modulation
		Reactor not tripped	Defeats the manual reset/block preventing automatic reactuation of safety injection
35			Blocks steam dump control via plant trip T_{avg} controller.
	P-11	2/3 Pressurizer pressure below setpoint	Allows manual block of safety injection actuation on low pressurizer pressure signal.
35			Allows manual block of safety injection actuation and steam line isolation on low compensated steam line pressure signal and allows steam line isolation on high steam line negative pressure rate.

TABLE 7.3.1-3 (2 of 2)

Interlocks For Engineered Safety Features Actuation System

	Designation	Input	Function Performed
35		2/3 Pressurizer pressure above setpoint	Defeats manual block of safety injection actuation and steam line isolation on low steam line pressure and defeats steam line isolation on high steam-line negative pressure rate.
			Reinstates automatically safety injection and steamline isolation on low steam line pressure and automatically blocks steam line negative pressure rate.
	P-12	2/4 T_{avg} below setpoint	Allows manual bypass of steam dump block for the cooldown valves only
35			Blocks steam dump
		3/4 T_{avg} above setpoint	Defeats the manual bypass of steam dump block
	P-14	2/3 Steam generator water level above setpoint on any steam generator	Closes all feedwater control valves
35			Trips all main feedwater pumps which closes the pump discharge valves.
			Actuates turbine trip.

35

Table 7.4.1-1

Auxiliary Shutdown Control Panel
Controls and Indicators Available for Hot Shutdown

Indicators

Pressurizer Level
Pressurizer Pressure
11 Reactor Coolant Loop D Hot Leg Temperature

Controls

Reciprocating Charging Pump
Nuclear Service Water Pump 1A
Nuclear Service Water Pump 1B
Component Cooling Water Pump 1A1
Component Cooling Water Pump 1A2
Component Cooling Water Pump 1B1
Component Cooling Water Pump 1B2
Pressurizer Heater Backup Group '1A'
Pressurizer Heater Backup Group '1B'
Boric Acid Transfer Pump 1A
Boric Acid Transfer Pump 1B
Letdown Orifice Isolation Valve - 1NV457A
Letdown Orifice Isolation Valve - 1NV458A
Letdown Orifice Isolation Valve - 1NV459A

11

Centrifugal Charging Pump 1A
Centrifugal Charging Pump 1B
Boric Acid Charging Pumps Valve - 1NV265B
Auxiliary Spray Supply to Pressurizer Isolation Valve - 1NV21A
11 NV Supply to NC Loop 4 Isolation Valve - 1NV16A
NV Supply to NC Loop 1 Isolation Valve - 1NV13B
Pressurizer #1 Power Operated Safety Relief Valve - 1NC34A
19 Pressurizer #1 Power Operated Safety Relief Valve - 1NC32B
Pressurizer #1 Power Operated Safety Relief Valve - 1NC36B
NC Loop 3 Supply to Excess Letdown Hx. #1 Isolation Valve - 1NV24B
NC Loop 3 Supply to Excess Letdown Hx. #1 Isolation Valve - 1NV25B
NC Letdown Isolation To Regenerative Hx. #1 Valve - 1NV1A
11 NC Letdown Isolation To Regenerative Hx. #1 Valve - 1NV2A
BA To BA Blender Control Valve - 1NV267A
Excess Letdown Hx. #1 Tube Outlet Control Valve - 1NV26
Regenerative Hx. #1 Tube Inlet Control Valve - 1NV241

Note: While not used for hot shutdown, RHR controls as described in 7.4.1.5.1.6, have been provided on this panel for the operators convenience.

Control Room Indicators and/or Recorders Available to the Operator to

Monitor Significant Unit Parameters During Normal Operation

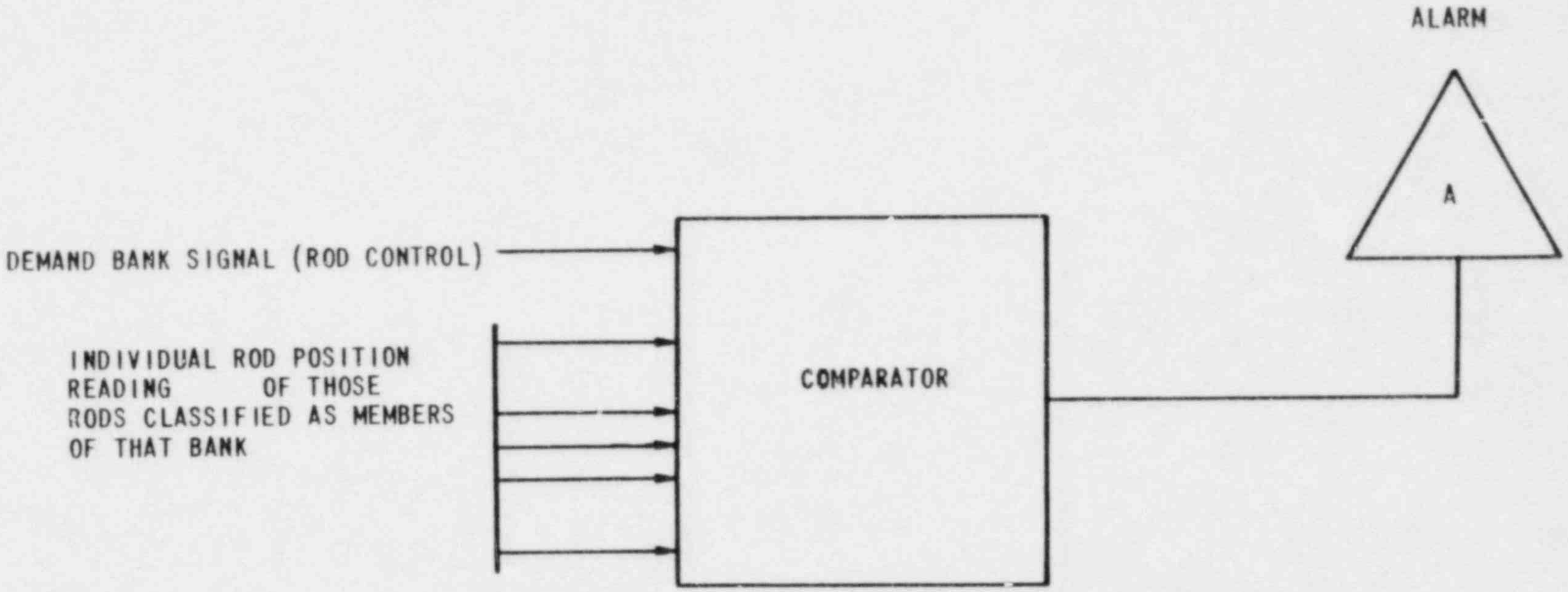
Parameter	No. of Channels Available	Range	Indicated Accuracy(1)	Indicator/ Recorder	Location	Notes
3. $T_{reference}$	1	540 to 590°F	± 4°F	The one channel is recorded.	Control Board	
4. Control Rod Position						If system not available, borate and sample accordingly.
a. Number of steps of demanded rod withdrawal	1/group	0 to 230 steps	± 1 step	Each group is indicated during rod motion.	Control Board	These signals are used in conjunction with the measured position signals (4c) to detect deviation of any individual rod from the demanded position. A deviation will actuate an alarm and annunciator.
b. Full length rod measured position	1 for each rod	0 to 228 steps	± 3 steps at full accuracy ± 6 steps at 1/2 accuracy	Each rod position is indicated.	Control Board	

35

Control Room Indicators and/or Recorders Available to the Operator to

Monitor Significant Unit Parameters During Normal Operation

	Parameter	No. of Channels Available	Range	Indicated Accuracy(1)	Indicator/ Recorder	Location	Notes
5.	Control rod bank measured	4	0 to 230 steps	± 2.5% of total bank travel	All 4 control rod bank positions are recorded along with the low-low limit alarm for each bank.	Control Board	1. One channel for each control rod 2. An alarm and annunciator is actuated when the last rod control bank to be withdrawn reaches the withdrawal limit, when any rod control bank reaches the low insertion limit and when any rod control bank reaches the low-low insertion limit
	CONTAINMENT SYSTEM						
8	Containment Pressure	4	0-115% of design pressure	± 3.5% of full scale	All 4 channels indicated, one is recorded	Control Board	
	FEEDWATER AND STEAM SYSTEMS						
1.	Auxiliary feed-water flow	1/Feed line	50 to 800 gpm	3.5%	All channels indicated.	Control Board	One channel to measure the flow to each steam generator
2.	Steam generator level (narrow range)	3/Steam generator	+7 to -5 feet from nominal full load level	± 4% of ΔP (hot)	All channels indicated. The channels used for control are recorded.	Control Board	

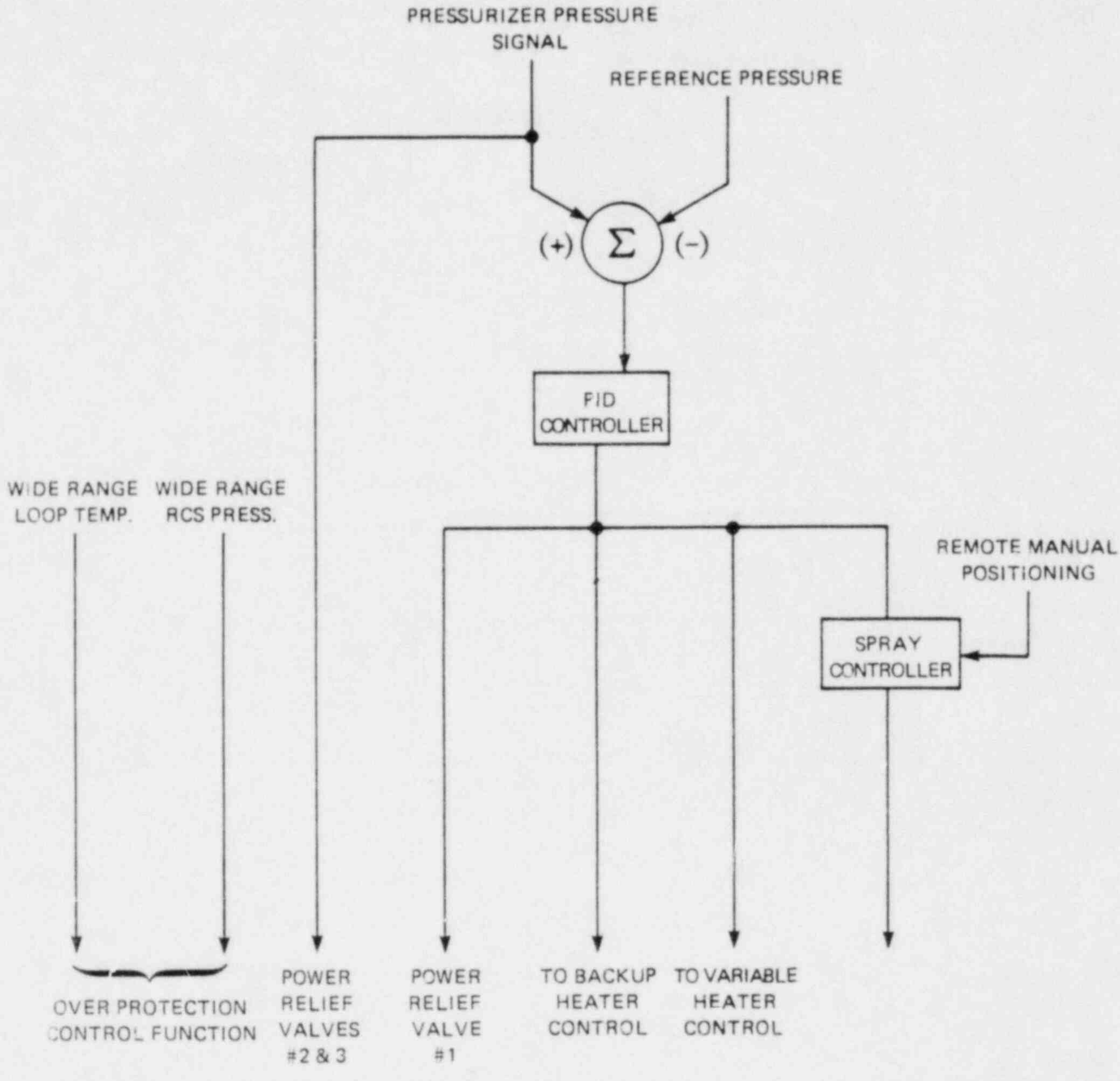


NOTE:
1. DIGITAL OR ANALOG SIGNALS MAY BE USED FOR THE COMPARATOR COMPUTER IMPUTS.
2. THE COMPARATOR WILL ENERGIZE THE ALARM IF THERE EXISTS A POSITION DIFFERENCE GREATER THAN A PRESENT LIMIT BETWEEN ANY INDIVIDUAL ROD AND THE DEMAND BANK SIGNAL.
3. COMPARISON IS INDIVIDUALLY DONE FOR ALL CONTROL BANKS.

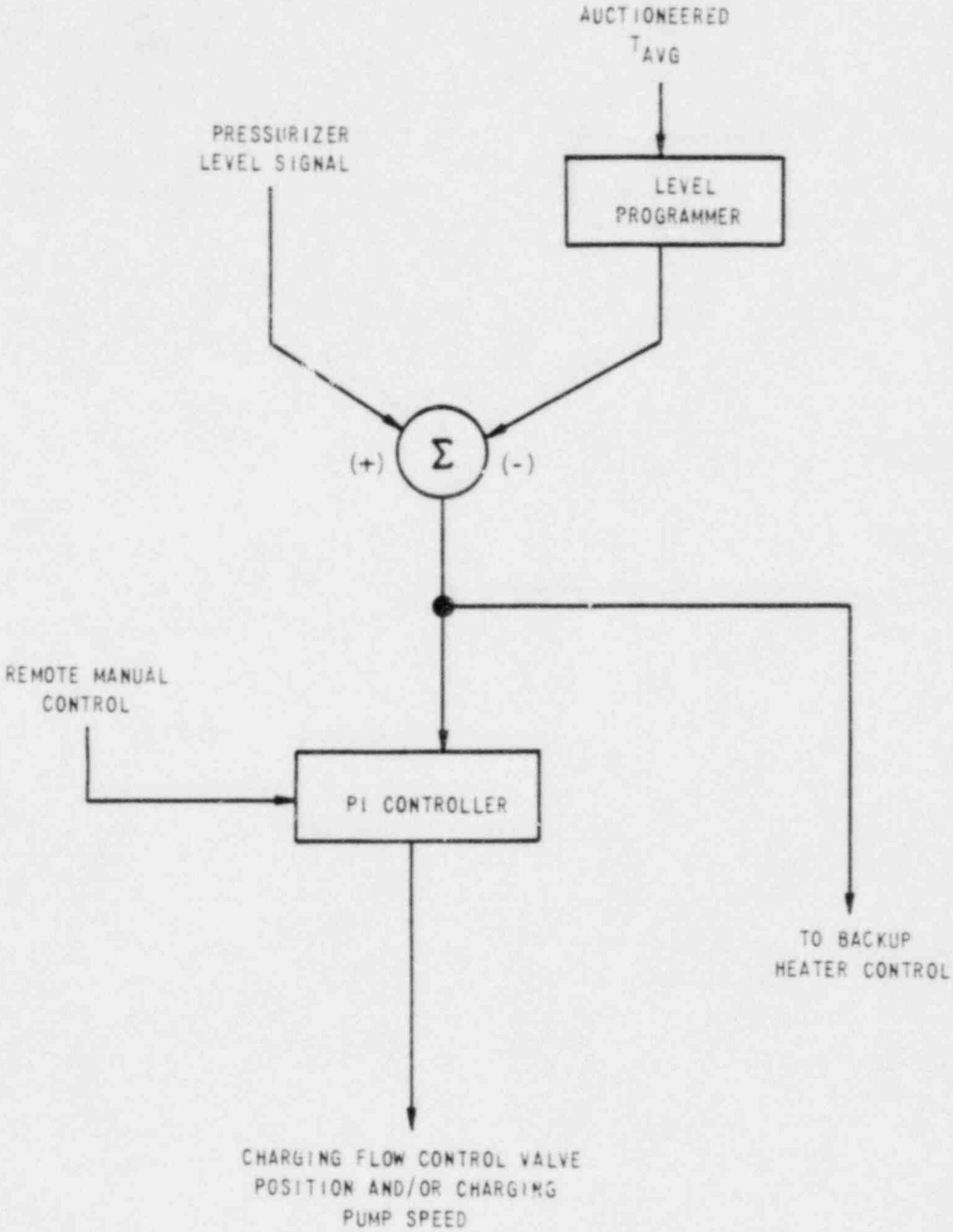
ROD DEVIATION COMPARATOR

McGUIRE NUCLEAR STATION

Figure 7.7.1-3



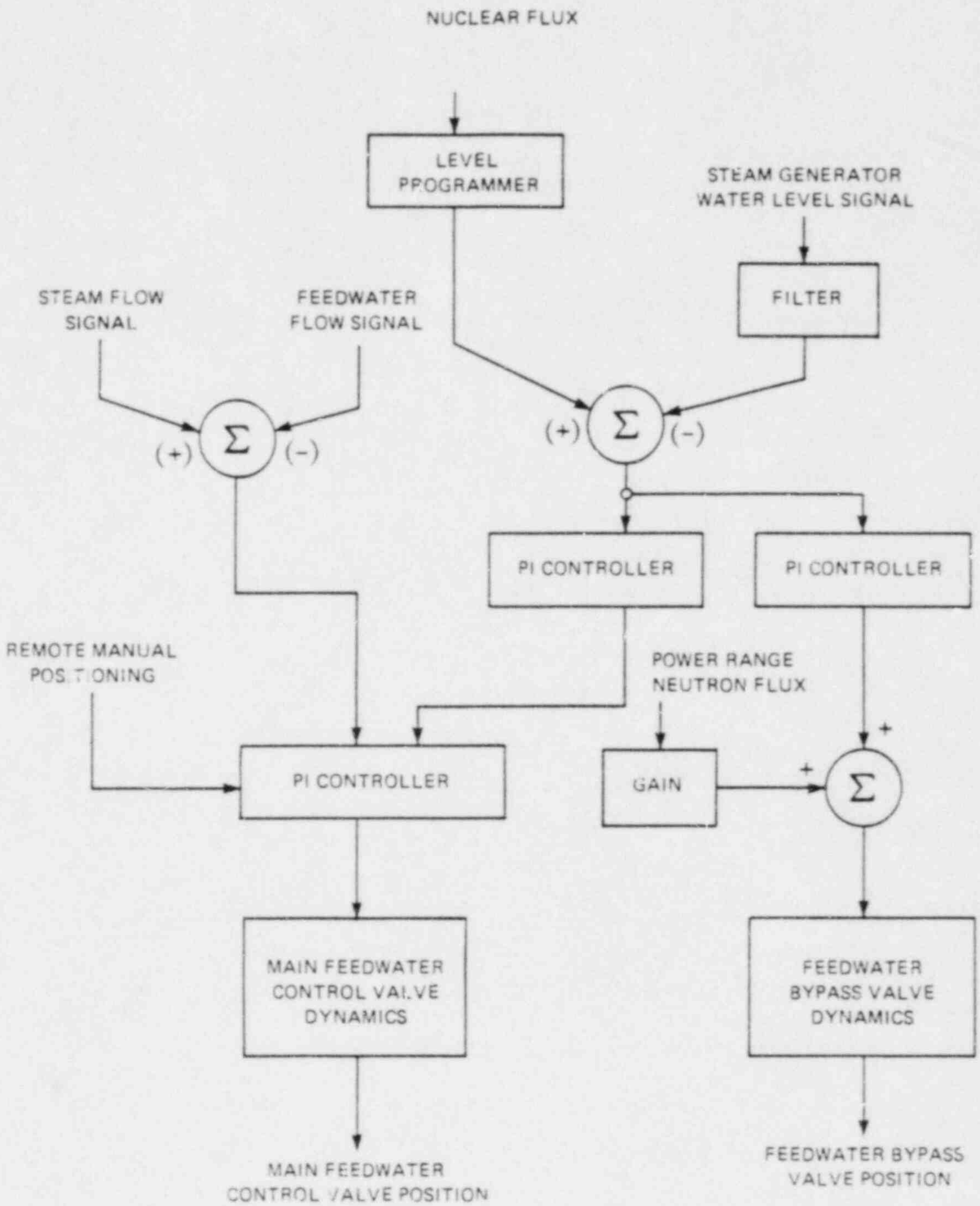
BLOCK DIAGRAM OF PRESSURIZER
PRESSURE CONTROL SYSTEM
McGUIRE NUCLEAR STATION
Figure 7.7.1-4
Revision 35



BLOCK DIAGRAM OF PRESSURIZER
LEVEL CONTROL SYSTEM

McGUIRE NUCLEAR STATION

Figure 7.7.1-5



BLOCK DIAGRAM OF STEAM GENERATOR WATER LEVEL CONTROL SYSTEM

McGUIRE NUCLEAR STATION

Figure 7.7.1-6

Revision 35

35 Table 9.3.1-3 (Page 1 of 3)

Unit 1 Valves Aligned to Blackout Air Supply

	Number	Location	Blackout Air Header Alignment	Valve Name
26				
	1CA36	AB	B	Aux FDWP No 1 Disch to Stm Gen 1D Control
	1CA40	AB	B	Aux FDWP 1B Disch to Stm Gen 1D Control
	1CA44	AB	B	Aux FDWP 1B Disch to Stm Gen 1C Control
	1CA48	AB	B	Aux FDWP No 1 Disch to Stm Gen 1C Control
	1CA52	AB	A	Aux FDWP No 1 Disch to Stm Gen 1B Control
	1CA56	AB	A	Aux FDWP 1A Disch to Stm Gen 1B Control
	1CA60	AB	A	Aux FDWP 1A disch to Stm Gen 1A Control
	1CA64	AB	A	Aux FDWP No 1 Disch to Stm Gen 1A Control
20	1NC32B	RB	B	Pressurizer No 1 Power Operated Safety Relief
	1NC34A	RB	A	Pressurizer No 1 Power Operated Safety Relief
	1NC36B	RB	B	Pressurizer No 1 Power Operated Safety Relief
	1NI23A	AB	A	BIT Recirc Auto Isolation
	1NI24B	AB	B	BIT Recirc Auto Isolation
	1NI41A	AB	A	Boron Inj Recirc Pumps Discharge Auto Block
	1NV1A	RB	A	NC Letdown Isol to Regenerative HX No 1
	1NV2A	RB	A	NC Letdown Isol to Regenerative HX No 1
	1NV13B	RB	B	NV Supply to NC Loop 1 Isolation
	1NV16A	RB	A	NV Supply to NC Loop 4 Isolation
	1NV21A	RB	A	NV Aux Spray Supply to Pressurizer Isolation
	1NV24B	RB	B	NC Loop 3 Supply to Excess Letdown HX No 1 Isolation
	1NV25B	RB	B	NC Loop 3 Supply to Excess Letdown HX No 1 Isolation

35 Table 9.3.1-3 (Page 2 of 3)

Rev.	Number	Location	Blackout Air Header Alignment	Valve Name
26				
	1NV26	RB	B	Excess Letdown HX No 1 Tube Outlet Control
	1NV124	AB	A	Low Pressure Letdown Control
	1NV137A	AB	A	NC Filters Outlet Three Way Control
	1NV238	AB	A	Centrifugal Charging Pumps Disch Control
	1NV241	AB	B	Regenerative HX No 1 Tube Inlet Control
	1NV267A	AB	A	Boric Acid to Boric Acid Blender Control
20	1NV457A	RB	A	Letdown Orifice 1C Outlet Containment Isolation
	1NV458A	RB	A	Letdown Orifice 1B Outlet Containment Isolation
	1NV459A	RB	A	Letdown Orifice 1A Outlet Containment Isolation
	1RN442A	AB	A	Control Room Air Conditioning Condenser A Control No 1
	1RV445A	AB	A	Control Room Air Conditioning Condenser A Control No 2
	1RN457B	AB	B	Control Room Air Conditioning Condenser B Control No 1
	1RN460B	AB	B	Control Room Air Conditioning Condenser B Control No 2
	1SM1	DH	B	Main Steam 1D Isolation
	1SM3	DH	A	Main Steam 1C Isolation
	1SM5	DH	B	Main Steam 1B Isolation
	1SM7	DH	A	Main Steam 1A Isolation
	1SV1	DH	A	Main Steam 1D Power Operated Relief
26	1SV7	DH	B	Main Steam 1C Power Operated Relief
	1SV13	DH	B	Main Steam 1B Power Operated Relief
	1SV19	DH	A	Main Steam 1A Power Operated Relief
	1RV79A	AB	A	Upper Cont. Vent. Unit Supply Cont. Isolation
35	1RV80B	RB	B	Upper Cont. Vent Unit Supply Cont. Isolation
	1RV101A	RB	A	Upper Cont. Vent Unit Discharge Cont. Isolation
	1RV102B	AB	B	Upper Cont. Vent Unit Discharge Cont. Isolation

Table 9.3.1-3 (Page 3 of 3)

Number	Location	Blackout Air Header Alighment	Valve Name
IRV32A	AB	A	Lower Cont. Vent Unit Supply Cont. Isolation
IRV33B	RB	B	Lower Cont. Vent. Unit Supply Cont. Isolation
IRV76A	RB	A	Lower Cont. Vent. Unit Discharge Cont. Isolation
IRV77B	AB	B	Lower Cont. Vent. Unit Discharge Cont. Isolation
IRF821A	AB	A	Cont. Fire Protection Supply Cont. Isolation
IRF832A	AB	A	Cont. Fire Protection Supply Cont. Isolation

Table 9.3.4-1

Chemical and Volume Control System Design Parameters (Per Unit)

General

Seal water supply flow rate, for four reactor coolant pumps, nominal, gpm	32
Seal water return flow rate, for four reactor coolant pumps, nominal, gpm	12
Letdown Flow:	
Normal, gpm	75
Maximum purification, gpm	120
Charging Flow (excludes seal water):	
Normal, gpm	55
Maximum prufication, gpm	100
Temperature of letdown reactor coolant entering system, F	557.3
Temperature of charging flow directed to Reactor Coolant System, F	516
Temperature of effluent directed to Boron Recycle System, F	115
Centrifugal charging pump bypass flow (each), gpm	60
Amount of 4% boric acid solution required to meet cold shutdown requirements at the end of a core cycle with the most reactive control rod stuck out of the core, gallons	14,700
Maximum pressurization required for hydrostatic testing of Reactor Coolant System, psig	3107

Table 10.3.2-1

Main Steam Supply System
Main Steam Line Safety Valves

Number of main steam lines	4
Number of valves per main steam line	5
Total number of safety valves	20

Design Data For Valves In Each Main Steam Line

	Valve No.	Set Pressure (psig)	Flow (lb/hr)
35	1	1170	608,832
	2	1190	616,511
	3	1205	907,165
	4	1220	918,325
35	5	1225	925,765
		Total per line	3,976,698

Total capacity for four lines lb/hr = 15,906,392

Table 10.4.1-1

Main Condenser Performance And Data

Characteristics	Shell A	Shell B	Shell C
Total heat load, Btu/hr	2.5589×10^9	2.5589×10^9	2.5589×10^9
Design absolute pressure in condensing zone in. Hg (with 60°F circulating water temperature at inlet to shells)	1.46	1.46	1.46
Maximum absolute pressure in condensing zone in. Hg (with 79°F circulating water temperature at inlet to shells)	2.40	2.40	2.40
Circulating water flow, GPM	319,867	319,867	319,867
Average velocity in tube, ft/sec	7.0	7.0	7.0
Effective surface area, ft^2	245,370	245,370	245,370
Cleanliness factor, %	95	95	95
Tube outside diameter, in.	1.0	1.0	1.0
Tube BWG	22	22	22
Tube overall length, ft	45.01	45.01	45.01
Number of Tubes: 304 Stainless Steel	20,944	20,944	20,944
Condensate stored at normal operating level, gal.	56,667	56,667	56,667

c. Level Instrumentation

OWMLT5000-Laundry and Hot Shower Tank Level-This instrument indicates liquid level in the laundry and hot shower tank, both at the WPS panel and locally. The instrument also provides high and low level alarms at the system panel and interlocks the laundry and hot shower pump with the tank level, so that the pump automatically shuts off when the level falls below a predetermined value. Then the floor drain tank pump is taking suction from the Laundry 4 and Hot Shower Tank, the floor drain tank pump will be automatically shut off on low level.

OWMLT5050-Floor Drain Tank Level-This instrument indicates liquid level in the floor drain tank, both at the system panel and locally, the instrument also provides high and low level alarms at the system panel, and interlocks the floor drain tank pump with the tank level, so that the pump automatically shuts off when the level falls below a predetermined value. The Laundry and 4 Hot Shower Pump is automatically shut off on low level when it is aligned to the floor drain tank.

OWMLT5260-Mixing and Settling Tank Level-This instrument is used to control valve 1WM172 on the pump discharge and to provide local indication. The mixing and settling tank pump is shutoff automatically on low level.

OWMLT5090-Waste Monitor Tank A Level-OWMLT5100 Waste Monitor Tank B Level-These instruments indicate liquid level in the waste monitor tanks, provide high and low level alarms on the WPS Panel, and interlock the waste monitor pumps with tank level, so that the pumps shut off automatically if the level falls below a predetermined value. Level indicators are located on the WPS 35 panel.

d. Flow Instrumentation

OWMFT5130, 1WMFT5140-Waste Monitor Tank A Pump Flow, Waste Monitor Tank B Pump Flow-These instruments indicate waste monitor pump flow on the system panel.

OWMPG5250-Waste Flow to Mixing and Settling Tank-This instrument provides local indication of waste flow to this tank.

Floor Drain Tank Subsystem

a. Pressure Instrumentation

30 OWMPG5070 and OWMPG5350-Floor Drain Tank Filter Differential Pressure-These instruments provide local indication of the differential pressure across the floor tank filter. The ΔP across the filter at full flow may be used to determine filter cleanliness.

OWMPG5060 Floor Drain Tank Pump Discharge Pressure-This instrument provides local indication of the discharge pressure of the floor drain tank pump.

b. Temperature Instrumentation

4

OWMTE5330 Floor Drain Tank Temperature-This instrument provides a computer alarm on high temperature.

Ventilation Unit Condensate Drain Tank Subsystem

a. Pressure Instrumentation

1WLPG5610, 1WLPG5630-Vent Unit Condensate Drain Tank Pump Discharge Pressure-This instrumentation provides local indication of the discharge pressure of the pumps.

4 b. Level Instrumentation

1WLLT5590-Vent Unit Condensate Drain Tank Level-This instrument indicates liquid level of the ventilation unit condensate drain tank in the control room. The instrument interlocks the ventilation unit condensate drain tank pump with the tank level, so that the pump automatically shuts off when the level falls below a predetermined value.

11.2.4 OPERATING PROCEDURES

The Liquid Waste Recycle System and the Liquid Waste Monitor and Disposal System are operated manually except for some functions of the reactor coolant drain tank subsystem and the mixing and settling tank operation. The system includes adequate control equipment to protect the system components and instrumentation and alarm functions to provide operator information to assure proper system operatior

11.2.4.1 Normal Operation

Operation of the system is essentially the same during all phases of normal reactor operation; the only differences are in the load on the system. The followina sections discuss the operation of the system in performing its

Radiation detectors are located in the waste effluent paths and provide back-up to laboratory analysis for the control of releases by supplying interlocks to automatically terminate releases at predetermined radioactivity concentrations.

The parameter measured (gross gamma, gross beta, or specific isotope) and the type of sensor selected for each location are chosen to provide information to the operator consistent with the radioactive isotope or combination of isotopes that are most indicative of the status of the unit, or that are expected to be potentially limiting. Other factors which influence the detector type and the parameter measured include response time, required sensitivity, background, and the availability of equipment to measure the minute radioactive concentrations consistent with regulatory limits. Dilution factors and/or integrated sampling are required to approach these sensitivities, particularly for airborne particulate, iodine, and liquid effluents.

The overall range of the process radiation monitoring instrumentation encompasses the full range of radioactive concentrations expected including postulated loss of coolant accident concentrations. For some applications, the magnitude of these ranges require dual instrumentation with widely different sensitivity characteristics. Where more than one sensor is necessary to insure coverage of the full range, overlap of ranges is provided.

An analog display which indicates the activity being sensed is provided in the Control Room for each monitoring channel. Analog outputs from all monitoring channels are also recorded on multipoint recorders located in the Control Room.

Control Room alarms are incorporated for annunciating high radioactivity from all monitoring channels, for annunciating loss of sample flow to detectors where off stream monitoring is incorporated, and for annunciating failed or abnormal monitor operation such as torn or depleted filter paper on airborne particulate monitors. The signal conditioning and readout package for each monitor channel includes a high level contact output, the setpoint for which is adjustable over the full range of the instrument. Basis for setpoints of each monitoring channel are included with the description of that channel.

In addition to providing high alarm information, interlocks are pro from monitors sensing radioactivity in the unit vents, waste gas effluent waste liquid effluent, steam generator blowdown and recycle, steam generator sampling, component cooling water, and Control Room ventilation. The interlocks are obtained in the same manner as the alarms with identical setpoint capability.

11.4.2.1 Liquid Monitoring

Table 11.4.2-1, liquid process radiation monitoring equipment, is a summary of continuous monitoring equipment. These monitors provide indication of radioactive concentrations within closed piping systems and provide an alarm for high concentrations in these systems. The systems monitored include those which are expected to be contaminated with radioactive fission or corrosion products and those systems which would become contaminated due to component failure such as steam generator or heat exchanger tube leaks. A description of each monitor is provided on a per unit basis. The total number of monitors for the station is indicated on the referenced table.

11.4.2.1.1 Reactor Coolant Monitor

A process radiation monitor sampler assembly consisting of a lead shielded sodium iodide (NaI) scintillation crystal, a photomultiplier tube, a pre-amplifier, and an electrically positioned check source is designed to continuously sense gross gamma in a sample of reactor coolant. Signal conditioning, counting, indicating, recording, and alarming equipment are located in the Control Room. The sampler assembly is located in an accessable area of the Auxiliary Building. To permit decay of activity not indicative of fuel clad integrity, primarily N^{16}, a delay of approximately one minute is incorporated in the sample transport from the reactor to the detector. The range of the instrument is approximately 1×10^{-2} µCi/ml to 1×10^{3} µCi/ml. This range is compatible with the calculated activities of 5×10^{-2} µCi/ml for corrosion products and 125 µCi/ml for one percent fuel defects.

Abnormal conditions of high activity or loss of sample flow are alarmed in the Control Room. The setpoint for high activity is adjustable over the full range of the instrument. Within limits of coolant activity established in the Technical Specifications, the high activity setpoint is adjusted to alarm a significant change in reactor coolant activity. Depending upon the magnitude of the change, the operator can verify the change by laboratory analysis and/or reduce reactor power. The loss of sample flow alarm initiates operator action to determine the cause for the loss and to re-establish the reactor coolant sample flow to the detector.

11.4.2.1.2 Conventional Waste Water Treatment System Monitor

The conventional waste water treatment system monitor continuously monitors the liquids discharged from the turbine building for activity prior to being treated in the conventional waste water treatment system. In the absence of a primary to secondary leak, the activity of liquids discharged from the turbine building is essentially background level and is verified by grab sampling routines. For a known primary to secondary leak, all contaminated materials are treated by other systems designed for this purpose. This monitor will essentially be reading background level at all times and is provided to give additional assurance of proper waste management by alarming
12 any detectable contamination. Q010.6

The monitor consists of an off line sample changer surrounded by 7-inches of 4 π lead shielding with a 2" x 2" NaI scintillation crystal and photo-multiplier tube. This arrangement provides high sensivity to gamma radiation with minimal interference from background sources. Concentrations of
35 1×10^{-6} µCi/ml are detectable based on I^{131}. Indications provided in the Control Room consists of six decade readout, annunciation on loss of sample flow or high activity level, and a recorded activity level. The receipt of a high activity alarm will alert the operator to an abnormal condition.
12 Evaluation of the source of the activity is then made to determine whether additional processing will be necessary.

academic or related technical training on a one-for-one, time basis. The Operating Engineer shall hold a Senior Reactor Operator license.

4 (f) Performance Engineer

The Performance Engineer shall have a minimum of a Bachelor's degree in
35 engineering or the physical sciences and two years of responsible nuclear power plant experience. The Performance Engineer or the Reactor Engineer shall have two years of experience in such areas as reactor physics, core measurements, core heat transfer, and core physics testing programs.

The Chemist shall have a minimum of five years of experience in chemistry, of which a minimum of one year shall be in radiochemistry. A minimum of two years of this five years of experience should be fulfilled by academic or related technical training. A maximum of four years of this five years of experience may be fulfilled by academic or related technical training.

(h) Health Physics Supervisor

The Health Physics Supervisor shall have a minimum of five years of experience in radiation protection at a nuclear facility. A minimum of two years of this five years of experience should be related technical training. A maximum of four years of this five years of experience may be fulfilled by academic or related technical training.

(i) Maintenance Engineer

4 A Maintenance Engineer shall have a high school diploma, or equivalent, and a minimum of four years of experience in maintenance activities. A maximum of three years of this four years of experience may be fulfilled by academic or related technical training.

(j) Instrument and Control Engineer

The Instrument and Control Engineer shall have a minimum of five years of experience in instrumentation and control of which a minimum of six months shall be in nuclear instrumentation and control. A minimum of two years of this five years of experience should be fulfilled by academic or related technical training. A maximum of four years of this five years of experience may be fulfilled by academic or related technical training.

(k) Shift Supervisor

A Shift Supervisor shall have the same qualifications as the Operating Engineer.

(l) Assistant Shift Supervisor

An Assistant Shift Supervisor shall have the same qualifications as a Shift Supervisor.

(m) Operators

Operators to be licensed by the NRC shall have a high school diploma, or equivalent, and two years of nuclear or fossil station experience, of which a minimum of one year shall be nuclear station experience. In order to be acceptable for full responsibility in a job, they shall hold an NRC Reactor Operator license.

Operators, whether or not they are to be licensed by the NRC, should have a high school diploma, or equivalent, and should possess a high degree of manual dexterity and mature judgment.

(n) Technicians

Technicians in responsible positions shall have a minimum of two years of experience in their specialty. These personnel should have a minimum of one year of related technical training in addition to their experience.

(o) Maintenance Personnel

Maintenance personnel in responsible positions shall have a minimum of three years of experience in one or more crafts. They should possess a high degree of manual dexterity and ability, and should be capable of learning and applying basic skills in maintenance operations.

13.1.3.2 Qualifications of Station Personnel

Mr. Maurice D. McIntosh, station Manager, held positions as Engineering Trainee, Junior Engineer, and Assistant Plant Engineer at a fossil-fired steam station after joining Duke in 1963. He was named Operating Engineer of Oconee Nuclear Station in 1968 following completion of one year of graduate study in Nuclear Engineering at North Carolina State University. He and other Oconee supervisors conducted classes for the "cold license" trainees for the six-month period immediately prior to six months work experience at an operating reactor. After reporting to the Oconee site in 1969, Mr. McIntosh was intimately involved in the startup and operation of the Oconee Nuclear Station until being named Acting Superintendent of McGuire Nuclear Station in 1973. In 1974 he was named station Manager for McGuire Nuclear Station. He received a BSME from N. C. State in 1962 and a Professional Degree in Nuclear Engineering from N. C. State in 1968. Mr. McIntosh held a Senior Reactor Operator license on Oconee Units 1 and 2.

Mr. George W. Cage, Operating Superintendent, joined Duke in 1967 following five years of experience in the U. S. Navy and six years of experience at the CVTR. In the U. S. Navy, as a First Class Petty Officer and qualified reactor operator, he directed men in startups, checkout and maintenance of reactor
6 plant controls and instrumentation. His experience at CVTR includes almost two years as a reactor technician and over four years as a Shift Supervisor. He held one of the first "hot licenses" and later a Senior Operator License at the CVTR. Prior to being named Operating Engineer in 1973, Mr. Cage was Assistant Operating Engineer and Training Shift Supervisor at the Oconee

Table 14.1.1-1

Preoperational and Startup Test Schedule

		Unit 1	Unit 2
	Begin Preoperational Testing	Aug 1, 1976	May 1, 1978
24	Begin Hot Functional Testing	Nov 1, 1978	Jul 1, 1980
	Initial Fuel Loading	Jan 1, 1979	Sep 1, 1980
	Moveable Incore Detector Functional Test		
	Incore Thermocouple Functional Test		
	Incore Thermocouple and RTD Cross-Calibration		
	Rod Position Indication Alignment		
20	Rod Cluster Control Assembly Drop Time Test		
	Rod Control System Alignment Test		
	Full-Length Rod Drive Mechanism Timing Test		
35			
	Reactor Coolant System Flow Test		
	Reactor Coolant System Flow Coastdown Test		
20	RTD Bypass Flow Verification		
	Reactor Protection System Trip Circuits Test		
	Initial Criticality	Feb 1, 1979	Oct 1, 1980
24	Begin Low Power Testing	Feb 1, 1979	Oct 1, 1980
	Begin Power Ascension Testing	Feb 7, 1979	Oct 7, 1980
	Commercial Operation	Jul 1, 1979	Mar 1, 1981

19 Table 14.1.3-1 (Page 1)

Testing Prior to Initial Fuel Loading

CONTENTS

19 Table 14.1.3-1 (Page 20)

PRESSURIZER FUNCTIONAL TEST

Abstract

Purpose

To establish the continuous spray flow rate and to determine the effectiveness of the pressurizer normal control spray and of the pressurizer heaters.

Prerequisites

The Reactor Coolant System is in a hot condition. The Reactor Coolant System is lined up for normal operation in accordance with applicable operating procedures. All reactor coolant pumps are operating. Each bank of pressurizer heaters is operable.

Test Method

While maintaining pressurizer level constant, spray bypass valves are adjusted until a minimum flow is achieved which maintains less than a 200°F temperature difference between the spray line and the Reactor Coolant System, and pressurizer heater cycling is minimized.

To determine pressurizer heater and spray capability, all pressurizer spray valves are closed. All pressurizer heaters are then energized and the time to reach a 2300 psig system pressure is measured and recorded. Continuous (bypass) spray valves are then returned to their previously determined setting and full spray is initiated through each spray valve individually and in parallel. Pressure versus time is recorded for each transient. The transient is terminated at a Reactor Coolant System pressure of 2000 psig by shutting the spray valves.

Acceptance Criteria

35 For setting of continuous spray flow, the flow through each bypass valve is such that the temperature difference between the spray line and the Reactor Coolant System is less than 200°F.

For spray and heater response tests, the response to induced transients is within the band assumed in the FSAR Safety Analysis.

19 Table 14.1.3-1 (Page 21)

PRESSURIZER RELIEF TANK FUNCTIONAL TEST

Abstract

Purpose

To demonstrate the functional performance of the pressurizer relief tank and associated equipment.

Prerequisites

This test is performed after the hydrostatic leak test of the Reactor Coolant System and prior to the start of the initial unit heatup. Support systems and components supplied by the pressurizer relief tank must be available to the extent necessary to demonstrate pressurizer relief tank performance. The pressurizer relief tank is ready for service and empty. Associated instrumentation and control equipment checkout has been completed.

Test Method

The pressurizer relief tank is isolated, filled and pressurized. Data are recorded during level and pressure increases. Associated instrumentation and control equipment setpoints are verified and/or adjusted as necessary. The tank is drained and backfilled using nitrogen as a cover gas.

Acceptance Criteria

5 The level and pressure alarms and cover gas system operate at the setpoints designated in the test documents. The pressurizer relief tank spray flow is verified to meet design requirements. Automatic pressure regulating valves and valve interlocks are verified to function properly.

Table 14.1.4-1 (Page 1 of 35)

Initial Startup Testing

CONTENTS

10 Table 14.1.4-1 (Page 2 of 35)

INITIAL FUEL LOADING

Abstract

Purpose

To accomplish initial fuel loading in a safe and orderly manner.

Prerequisites

Testing prior to initial fuel loading is completed sufficiently to demonstrate the operability of required systems and components. Temporary and permanent source range channels are operable. At least one path for boron addition to Reactor Coolant System is available. Uniform boron concentration in the
5 Reactor Coolant System is maintained by recirculation with at least one Resi-
30 dual Heat Removal Pump and is sufficient to assure $K_{eff} \leq 0.95$ during fuel loading. Containment intergrity is established in accordance with station
5 operating license.

Test Method

Fuel and, where appropriate, fuel inserts are inserted into the reactor vessel in accordance with the prespecified loading sequence. Neutron count rate is monitored on temporary and permanent source range detectors. Core reactivity is monitored through plots of inverse neutron count rate ratio.

Acceptance Criteria

The core is assembled in accordance with the prespecified configuration.

FULL-LENGTH ROD DRIVE MECHANISM TIMING TEST
Abstract

Purpose

To demonstrate proper operation and timing of each rod drive mechanism.

Prerequisites

The reactor vessel upper internals are installed, the reactor vessel head is installed with studs tensioned, each full-length rod is latched and the Reactor Coolant System is filled and vented. The reactor is in the cold or hot shutdown condition, as dictated by the test requirements. Containment integrity is established in accordance with applicable Technical Specifications. Cold condition Rod Control System alignment has been completed.

Test Method

With the reactor in the cold shutdown condition, the timing for each slave cycler is set, measured and reset as necessary. Each full-length rod drive mechanism is manually operated with a rod cluster control assembly attached, checking the latching and releasing features of each. The test is repeated for each rod drive mechanism with the reactor in the hot shutdown condition.

Acceptance Criteria

The final settings for each slave cycler with the reactor in the hot shutdown condition are in accordance with the rod drive mechanism design requirements. The free latching and releasing of each rod drive mechanism is verified under both cold and hot conditions.

(Deleted)

10 Table 14.1.4-1 (Page 29 of 35)

UNIT LOAD TRANSIENT TEST

Abstract

Purpose

To demonstrate satisfactory unit response to a 10 percent load change.

Prerequisites

The various control systems have been tested and are in automatic. All pressurizer and main steam relief and safety valves are operable. The control rods are in the maneuvering band for the power level existing at the commencement of the test. Unit conditions are stabilized and all pertinent parameters to be measured are connected to high speed recorders.

Test Method

Output is manually reduced at a rate sufficient to simulate a step load change equivalent to approximately a 10 percent load decrease. After stabilization of all systems, output is manually increased at a rate sufficient to simulate a step load change equivalent to approximately a 10 percent load increase. Pertinent parameters affected by a load change are measured and recorded. At various power levels, as required by the test procedure, the test is repeated.

Acceptance Criteria

Neither the turbine nor the reactor trips, and no initiation of safety injection is experienced. No pressurizer or main steam relief or safety valves lift. No operator action is required to restore conditions to steady state. Parameters affected by the load change do not incur sustained or divergent oscillations.

(Deleted)

SHUTDOWN FROM OUTSIDE CONTROL ROOM TEST

Abstract

Purpose

To demonstrate the capability to shutdown the unit from outside the control room.

Prerequisites

Various control systems are functioning properly. Unit generator output $\geq$10%.

Test Method

Evacuation of the main control room is simulated by dispatching personnel to their assigned stations while additional operators occupy the control room to observe unit behavior. The reactor is tripped at the local reactor trip switchgear. The unit is maintained in the hot standby condition by manipulation of local controls and observation of local indications.

Acceptance Criteria

The reactor trips. The turbine generator trips. A stable hot standby condition is established and maintained.

Table 14.1.4-2

Power Ascension Test Program

	Test	0%	>10%*	30%	50%	75%	90%	100%
10								
	Zero Power Physics Test	X						
	Rod Control Cluster at Power Test			X				
	Pressurizer Pressure & Level Control System Test			X				
24	Radiation Shielding Survey	X		X	X	X		X
	Nuclear Instrumentation Initial Calibration			X	X	X	X	X
	Unit Load Transient Test			X		X		X
	Effluent Radiation Monitor Test				X			X
	Power Coefficient & Power Defect Measurement			X	X	X		X
	Below Bank Rod Test				X			
	Pseudo Rod Ejection Test			X				
	Incore & Nuclear Instrumentation Systems Detector Correlation					X		
	Unit Load Steady State	X		X	X	X	X	X
	Unit Loss of Electrical Load Test				X			X
35								
	Turbine Trip Test							X
	Dynamic Rod Drop Test				X			
	Core Power Distribution	X		X	X	X	X	X
	Loss of Offsite Power Test		X					
10	Shutdown from Outside Control Room Test		X					

*Generator Load

X - Indicates test to be performed

X - Indicates test the successful completion of which is a pre-requisite for increasing power to the next testing plateau. Otherwise, a safety analysis will be performed for any discrepancies prior to the power increase.

For reactivity insertion rates between $\sim 2.5 \times 10^{-4}$ δK/sec and $\sim 4 \times 10^{-5}$ σK/sec the effectiveness of the Overtemperature ΔT trip increases (in terms of increased minimum DNB ratio) due to the fact that with lower insertion rates the power increase rate is slower, the rate of rise of average coolant temperature is slower and the lead-lag compensation provided can increasingly account for the coolant system thermal capacity lag.

4. For reactivity insertion rates less than $\sim 4 \times 10^{-5}$ δK/sec, the rise in reactor coolant temperature is sufficiently high so that the steam generator safety valve setpoint is reached prior to trip. Opening of these valves, which act as an additional heat load on the RCS, sharply decreases the rate of rise of Reactor Coolant System average temperature. This decrease in rate of rise of the average coolant system temperature during the transient is accentuated by the lead-lag compensation causing the Overtemperature ΔT trip setpoint to be reached later with resulting lower minimum DNB ratios.

For transients initiated from higher power levels (for example, See Figure 15.2.2-5) this effect, described in 4. above, which results in the sharp peak in minimum DNB ratio at $\sim 4 \times 10^{-5}$ δK/sec, does not occur since the steam generator safety valves are never actuated prior to trip.

Figures 15.2.2-5, 15.2.2-6 and 15.2.2-7 illustrate minimum DNBR calculated
7 for minimum and maximum reactivity feedback.

15.2.2.3 Conclusions

The high neutron flux and overtemperature ΔT trip channels provide adequate protection over the entire range of possible reactivity insertion rates, i.e., the minimum value of DNBR is always larger than 1.30.

15.2.3 ROD CLUSTER CONTROL ASSEMBLY MISALIGNMENT

15.2.3.1 Identification of Causes and Accident Description

Rod cluster control assembly misalignment accidents include:

1. A dropped full-length assembly;
2. A dropped full-length assembly bank:
3. Statically misaligned full length assembly (See Table 15.2.3-1).

35

7 Each rod cluster control assembly has a position indicator channel which displays position of the assembly. The displays of assembly positions

are grouped for the operator's convenience. Fully inserted assemblies are further indicated by a rod bottom light. Group demand position is also indicated. The full length assemblies are always moved in preselected banks and the banks are always moved in the same preselected sequence.

A dropped assembly or assembly banks are detected by:

1. Sudden drop in the core power level is seen by the Nuclear Instrumentation System;

2. Asymmetric power distribution as seen on out of core neutron detectors or core exit thermocouples;

3. Rod bottom light(s);

4. Rod deviation alarm;

5. Rod position indication.

Misaligned assemblies are detected by;

1. Asymmetric power distribution as seen on out of core neutron detectors or core exit thermocouples;

2. Rod deviation alarm;

3. Rod position indicators.

The resolution of the rod position indicator channel is ± 5 percent of span (± 7.2 inches). Deviation of any assembly from its group by twice this distance (10 percent of span, or 14.4 inches) will not cause power distributions worse than the design limits. The deviation alarm alerts the operator to rod deviation with respect to group demand position in excess of 10 percent os span. If the rod deviation alarm is not operable, the operator is required to take action as required by the Technical Specifications.

If one or more rod position indicator channels should be out of service, detailed operating instructions shall be followed to assure the alignment of the non-indicated assemblies. These operating instructions require selected pairs of core exit thermocouples to be monitored in a prescribed time sequence and following significant motion of the non-indicated assemblies. The operating instructions also call for the use of moveable in-core neutron detectors to confirm core exit thermocouple indication of assembly misalignment.

15.2.3.2 Analysis of Effects and Consequences

Method of Analysis

Steady state power distributions are analyzed for this event using the TURTLE[5] code. The peaking factors calculated by TURTLE are then used

NRC position transmitted by letter of July 14, 1977 from Karl Kniel, Chief, Light Water Reactors Branch No. 2, Division of Project Management

Class 1E Equipment Qualification (Outside Containment)

With regard to all Class 1E equipment located outside the containment building, we require assurance that the environment is maintained within the temperature range for which the equipment is qualified to operate. In those locations where the temperature could exceed that for which the Class 1E equipment is qualified, the Staff requires that the applicant provide a temperature monitoring system. The system should at a minimum meet the following requirements:

a. The control room should receive an alarm when the temperature range has been exceeded. This alarm should be provided by instrumentation which

    1. is of high quality

    2. is checked to verify its functional capability by plant technical specification requirements, and

    3. is powered from a continuous power source or is redundant with separate channels and power sources.

b. The operator should have a method of maintaining a continuous record of the temperature during the time that the temperature range is exceeded.

Based on the monitoring system the applicant shall report the occurrence of the temperature exceeding the equipment qualification range as an abnormal occurrence to the NRC. In addition to this, the applicant shall provide results of an analysis to demonstrate that the excess temperature has not degraded the involved Class 1E equipment below an acceptable level for continued plant operation.

Response:

For plant areas containing Class 1E equipment where the area temperature could be postulated to exceed the design temperature of the equipment in that area, a temperature monitoring system will be provided. This temperature monitoring system will provide an alarm in the Control Room when the temperature in the monitored area exceeds a preset value. Upon receipt of a high temperature alarm, the temperature in the alarmed area will be recorded periodically, either manually or automatically during the time that the area temperature is above the alarm setpoint.

This system will be comprised of high quality components and will have provisions for verifying its functional capability. The temperature monitoring system will be powered from a continuous power source (i.e., the battery-backed plant auxiliary control power system described in Section 8.3.2.1.3).

Table S2-1 is a listing of Class 1E equipment located in areas outside the containment where the area temperature could be postulated to exceed the equipment design temperature. These areas which will be monitored were identified based on the following criteria:

1. Those areas served by redundant safety-related ventilation systems, assume the loss of one train of powered ventilation.

2. Those areas served by a single, train-related safety-related ventilation system, assume a complete loss of the powered ventilation train.

27 3. Those areas served by a non safety-related ventilation system, assume a total loss of powered ventilation.

4. For 1, 2, and 3 (above), the design climatic conditions for the site were obtained from ASHRAE, Handbook of Fundamentals, 1972 Edition, Table of Climatic Conditions for the United States and Canada (Charlotte, North Carolina).

In the event that the temperature in a monitored area exceeds the design temperature of the Class 1E equipment in that area, a report will be filed with the NRC. This report will contain an analysis of the effects of the excess temperature on the Class 1E equipment in the alarmed area.

This temperature monitoring system for the plant areas identified in
35 Table S2-1 is scheduled to be installed four months after Unit 1 fuel loading.

NRC REQUEST FOR INFORMATION TRANSMITTED
BY LETTER OF NOVEMBER 6, 1978 FROM ROBERT L. BAER,
CHIEF, LIGHT WATER REACTORS BRANCH NUMBER 2
DIVISION OF PROJECT MANAGEMENT

I. Provide an evaluation of the exfiltration that can occur from the fuel storage building under fuel handling accident conditions. Your evaluation should include a calculation of the expected flow of air through the paneled areas of each side of the building, and the resulting negative pressure, as a function of wind speed. Determine the wind speed at which air will begin to flow out of the building, and the flow rate as a function of wind speed. The degree of mixing assumed in the building should be justified. If complete mixing in the fuel building is assumed, this exfiltration flow will contain activity at the same concentration as the air which reaches the filters in the exhaust system. This fact can then be used to apportion the fractions of filtered and unfiltered release which is occurring at each wind speed. The atmospheric dispersion factor may be considered inversely proportional to the wind speed, starting from the value used in the base case analysis, which is usually 1 m/sec.

II. In order that we may evaluate your analysis, and perform independent analyses as necessary, provide the following information:

- Information and considerations for evaluating fuel handling accidents outside containment.
- Area of panel siding on each of four faces of building exposed to winds.
- Area of panel siding not exposed to winds (connected to other buildings).
- The pressures that can be expected in connected buildings.
- Size and locations of doors and major penetrations.
- Flow rate of exhaust air passing through the area occupied by the fuel pool.
- Differences between total supply air and total exhaust air flow for fuel building.
- Readable drawing showing the location and layout of the building ventilation system and important building features.
- Leakage test data as follows:

    - Wind speed and direction with respect to building during tests.

    - Negative pressure measured in building for two different supply air flows.

    - Locations of pressure reference points, instrument type and accuracy.

For a spectrum of wind speeds (1 to 10 m/sec) having velocity pressure P_v:

- Positive fraction of velocity pressure assumed for windward paneled areas.
- Negative fraction of velocity pressure assumed for leeward paneled areas.
- Negative fraction of velocity pressure assumed for paneled areas parallel to winds.

- Area assumed to have higher negative pressures to allow for unusual wind pressure patterns, and the pressure assumed.

Response:

I. An analysis of the effects of leakage from the fuel storage building for a postulated fuel handling accident was performed. The results of this analysis are shown below for three different wind speeds.

WIND SPEED (meters/sec)	DOSE (REM)	
	WHOLE BODY	THYROID
1	4.0	52
5	4.0	10.4
10	4.0	6.75

From this analysis it is apparent that the improved atmosphere dispersion more than compensates for the increased building leakage at higher wind speeds. Figure 1 shows the calculated building leakage as a function of wind speed. Complete mixing was assumed in the building for calculational purposes. This is a conservative assumption since most of the activity would remain in the vicinity of the fuel pool and would be filtered prior to release.

II. Other information:

1. Area of panel siding on each of four faces exposed to winds --
   North - 3480 ft^2
   South - 0
   East - 2175 ft^2
   West - 2667 ft^2 (Prevailing wind direction)

2. Area of panel siding not exposed to winds (connected to other buildings) --
   None

3. Pressures that can be expected in connected buildings --
   Slightly negative (assumed 0 psig in analysis)

4. Size and locations of doors and major penetrations --
   Roll-up door on north face of building - 18.5' x 21'

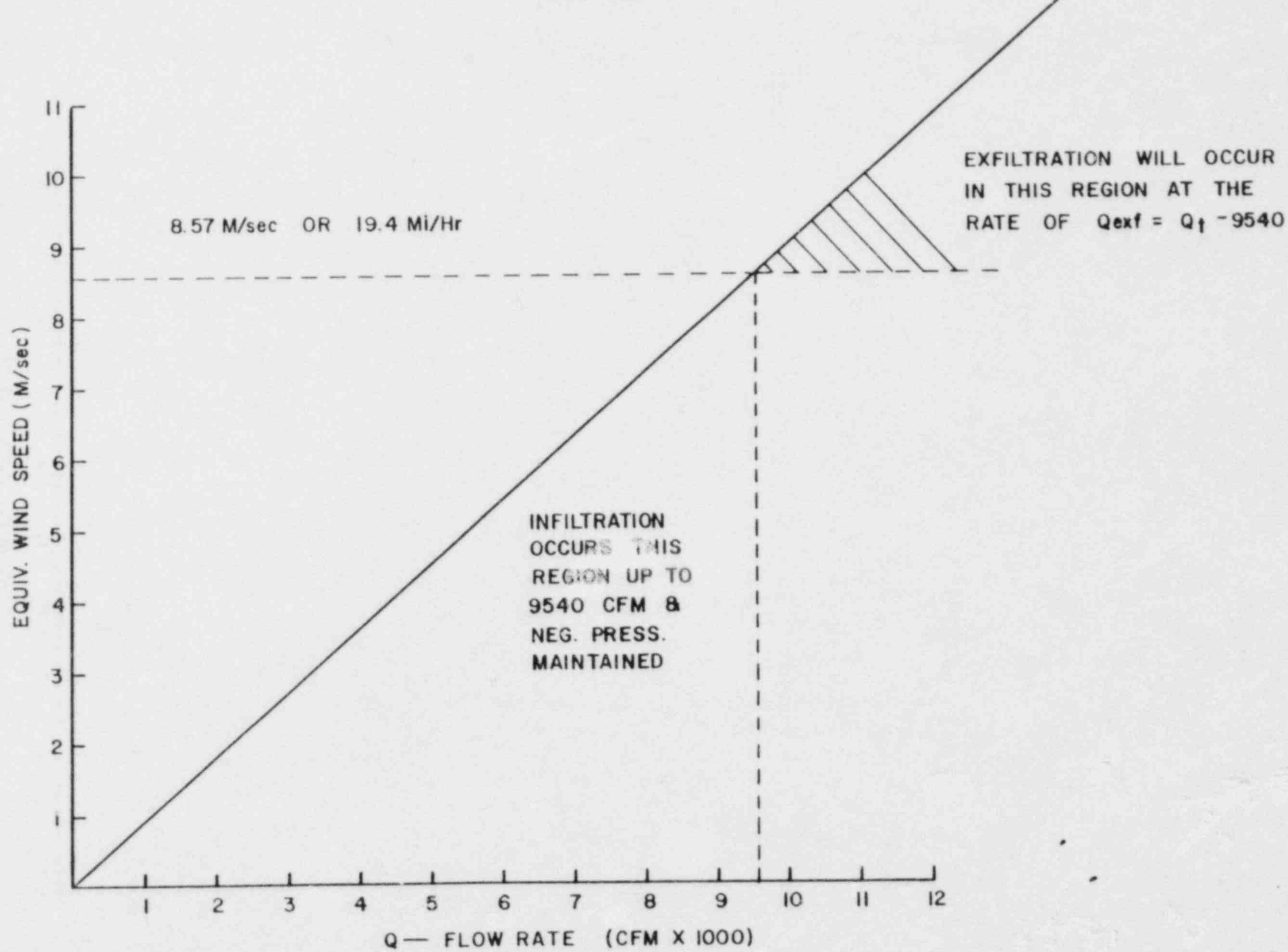
5. Flow rate of exhaust air passing through the area occupied by the fuel pool --
   35,675 cfm

6. Difference between total supply air and total exhaust air flow for fuel building --
   9540 cfm

7. Readable drawing showing the location and layout of the building ventilation system and important building features --
   See FSAR figures 9.1.1-1, 9.1.1-2 & 9.4.2-3

8. Leakage test data --
   Wind speed and direction -
   10 mph, 220°
   Negative pressure measured in building for two different supply air flows
   - .03" with 26,135 cfm supply, 35,675 cfm exhaust
   - .36" with ø cfm supply, 35,675 cfm exhaust
   Location of pressure reference points, instrument type and accuracy -
   Measurements taken with an incline manometer (0-1" W.G. range) along west side of building.
   Velocity pressure factors (from ANSI A58.1-1972) -
   +.8 - windward panels
   -.6 - leeward panels
   -.7 - parallel panels
   All areas in building assumed tube at same pressure (-.03")

FIGURE I


EXFILTRATION WILL OCCUR IN THIS REGION AT THE RATE OF Qexf = Qt -9540
8.57 M/sec OR 19.4 Mi/Hr
INFILTRATION OCCURS THIS REGION UP TO 9540 CFM & NEG. PRESS. MAINTAINED
EQUIV. WIND SPEED (M/sec)
Q — FLOW RATE (CFM X 1000)
1
2
3
4
5
6
7
8
9
10
11
12