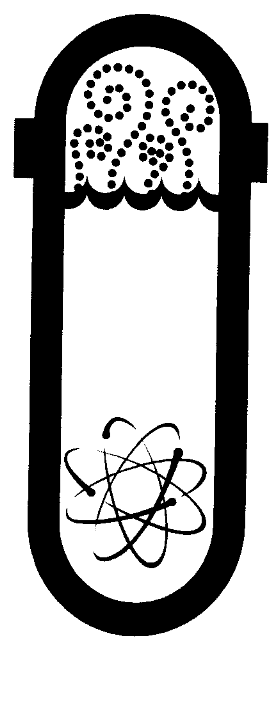
GE Nuclear Energy

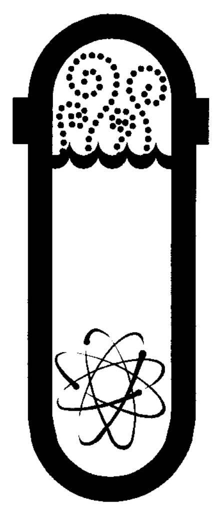
BWR/6
General Description of a Boiling Water Reactor



60140

GE Nuclear Energy

BWR/6
General Description of a Boiling Water Reactor



60149

NOTICE

(Compiled and edited by Proposal Engineering, Nuclear Power Systems Engineering Department.)

CONTENTS

CONTENTS (Continued)

TABLES

ILLUSTRATIONS

ILLUSTRATIONS (Continued)

ILLUSTRATIONS (Continued)

Section 1
Introduction

BWR BACKGROUND

In this era of greater ecological emphasis, dwindling fossil fuel supply, and increased electric power demand, it becomes necessary to use to the fullest extent possible all technological advances available to us. Nuclear fuel — replacing coal, oil, and gas — provides the most economical, the most reliable, and the most stabilized power source of the era.

General Electric selected the boiling water reactor (BWR) as the most promising nuclear power concept because of its inherent advantages in control and design simplicity, and established an atomic power equipment business in 1955 to offer it commercially. Aside from its heat source, the BWR generation cycle is substantially similar to that found in fossil-fueled power plants.

The beginning of the General Electric BWR Product Line was the Vallecitos BWR in 1957, which was located in Vallecitos, California. This 1000-psi (6900 kPa) reactor powered a 5-MWe generator and provided power to the Pacific Gas & Electric Company grid through 1963. A major extrapolation from that first test facility is the Dresden 1 plant, located near Morris, Illinois. Construction of this 180-MWe plant began in 1959, with commercial power production achieved in 1961.

The value of the BWR concept has been proven by the BWR designs General Electric has evolved over the past decade. The Oyster Creek plant, located in Toms River, New Jersey, incorporated the BWR/2, the first modern light water reactor. The BWR/2 was first offered in 1963, and had an output of well over 500 MWe. Then came BWR/3, at the twin Dresden 2 and 3 units, at 800 MWe. Shortly thereafter, the Browns Ferry BWR/4 units at 1100 MWe located in Decatur, Alabama, were announced, followed by the Zimmer Class BWR/5 in 1969 located in Moscow, Ohio. And in 1972, BWR/6 with ratings up to 1390 MWe was introduced as the sixth stage in the design evolution of the BWR. In terms of product refinement, the progressive development from BWR/1 through BWR/6 has been marked by a disciplined, evolutionary, building-block approach, supported at each step by exhaustive research and testing, and underscored by adequate design margins at each stage. Today, the selection and refinement of standard modules serves as the basis for General Electric's design standardization activities. Since the Vallecitos BWR, utilities throughout the world are committed to over 87 GE-BWR's, of which 36 are currently in operation. A summary of the BWR design progress is presented in Table 1-1.

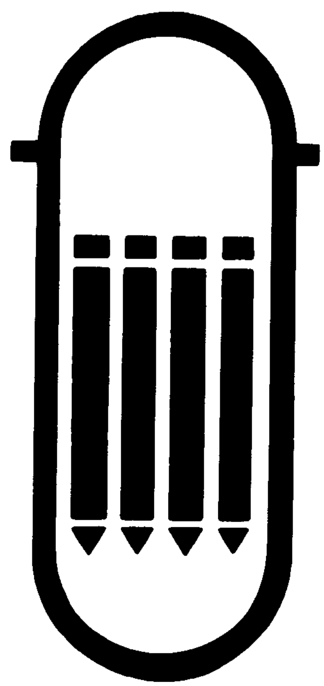


Table 1-1
EVOLUTION OF THE GENERAL ELECTRIC BWR

Product Line Number	Year of Introduction	Characteristic Plants
BWR/1	1955	Dresden 1, Big Rock Point, Humboldt Bay, KRB Initial commercial BWR's First internal steam separation
BWR/2	1963	Oyster Creek Plants purchased solely on economics Large direct cycle
BWR/3	1965	Dresden 2 First jet pump application Improved ECCS: spray and flood
BWR/4	1966	Browns Ferry Increased power density (20%)
BWR/5	1969	Zimmer Improved ECCS systems Valve flow control
BWR/6	1972	BWR/6 8 by 8 fuel bundle Improved jet pumps and steam separators Added fuel bundles, increased output Reduced fuel duty: 13.4 kW/ft (44 kW/m) Improved ECCS performance Improved licensability Solid-state nuclear system protection system Compacted control room

BWR/6 PRODUCT LINE

This document describes the latest BWR Product Line, the BWR/6. The improvements in performance, reliability, and licensability represent a logical extension of the values which traditionally have characterized BWR Product Lines. This latest product improvement is again a well-balanced combination of extensive GE development work, generating experience obtained from BWR plants, and retention of proven standard modules to strengthen the BWR reliability, licensability, and standardization.

The BWR/6 Product Line is capable of producing 20% more power from the same size pressure vessels as used in the BWR/5 Product Line, without increasing the size of the respective buildings or supporting systems. Power output capabilities range from approximately 600 MWe to 1400 MWe gross. Principal design features include:

- Compact jet pumps with increased coolant circulation capability.
- Increased capacity from steam separators and dryers.
- More fuel bundles in standard pressure vessels and improvements in reactor internals arrangement.
- Smaller-diameter fuel rods, longer in active fuel length and arranged in 8 by 8 bundles within the same external outline as the previous 7 by 7 design. This lowers the kilowatt rating per length of fuel and permits increased heat output per bundle.
- Improved control and instrumentation systems incorporating the latest solid-state electronics technology.
- Improved operator-machine interface systems for better control of the plant.

MARK III CONTAINMENT

Also described in this document is the Mark III reference containment design for the BWR/6 Product Line. The Mark III provides a number of innovative features over previous containment designs:

- Reduced overall reactor building height
- Improved seismic response
- Lower containment design pressure
- Improved accessibility for installation and inspection of nuclear boiler piping and equipment
- Improved pipe whip design
- Improved licensing features
- Shorter construction schedules
- Construction cost savings

In short, Mark III employs the construction simplicity of a dry containment while retaining the proven safety and low pressure advantages of a pressure suppression type containment.

SUMMARY DESCRIPTION

The direct cycle boiling water reactor nuclear system (Figure 1-1) is a steam generation and steam utilization system consisting of a nuclear core located inside a reactor vessel and a conventional turbine-generator and feedwater supply system. Associated with the nuclear core are auxiliary systems to accommodate the operational and safeguard requirements and necessary controls and instrumentation. Water is circulated through the reactor core, producing saturated steam which is separated from recirculation water, dried in the top of the vessel, and directed to the steam turbine-generator. The turbine employs a conventional regenerative cycle with condenser deaeration and condensate demineralization.

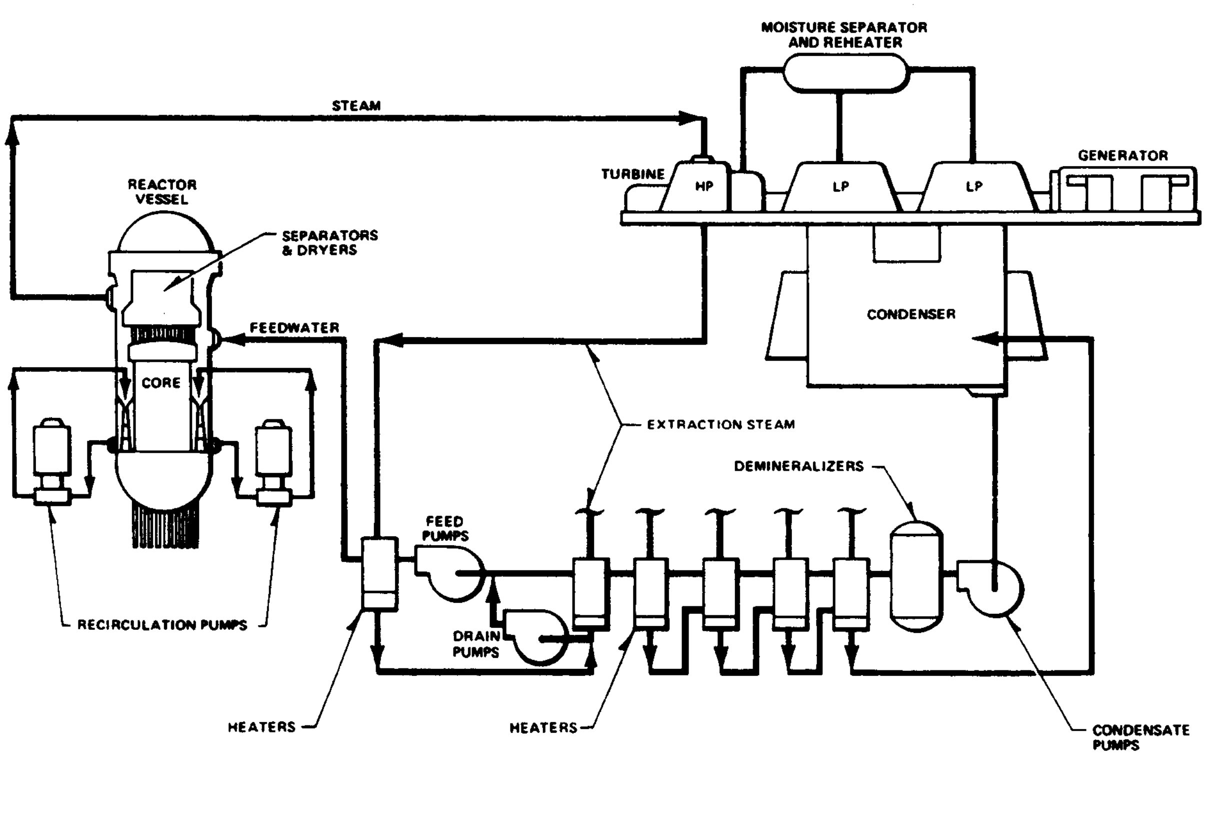


Figure 1-1. Direct Cycle Reactor System

The steam produced by the nuclear core is, of course, radioactive. The radioactivity is primarily N-16, a very short-lived isotope (7 seconds half-life) so that the radioactivity of the steam exists from the reactor vessel only during power generation. Extensive generating experience has fully demonstrated that shutdown maintenance on a BWR turbine, condensate, and feedwater components can be performed essentially as at a fossil-fuel plant. Carryover of long-lived radioactive particles by the steam supply to the turbine and condensate system is virtually nonexistent. More than 700 billion kWh of successful operating experience and numerous refueling and maintenance outages in plants using this direct-cycle approach support the soundness of General Electric's choice of the BWR.

The nuclear core, the source of the heat, consists of fuel assemblies and control rods contained within the reactor vessel and cooled by the recirculating water system. A 1220-MWe BWR/6 core consists of 748 fuel assemblies and 177 control rod assemblies forming a core array about 16 feet (4.9m) in diameter and 14 feet (4.3m) high. (See Section 3 for a detailed description of the BWR core.) The power level is maintained or adjusted by positioning control rods up and down within the core. The BWR core power level is further adjustable by changing the recirculation flow rate through the core without changing control rod position. This unique BWR feature helps achieve the superior load-following capability of the BWR.

The BWR is the only light water reactor system that employs bottom-entry control rods (see Section 2). Bottom-entry and bottom-mounted control rod drives

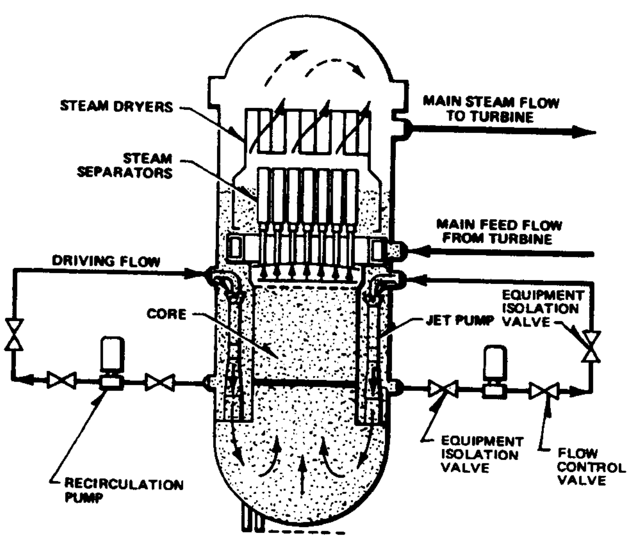


Figure 1-2. Steam and Recirculation Water Flow Paths

allow refueling without removal of control rods and drives, and allow drive testing with an open vessel prior to initial fuel loading or at each refueling operation. The hydraulic control rod drive system, which incorporates mechanical locking of the rod at the selected position, provides positive driving and positioning of the control rods. Rapid control rod insertion is accomplished by pressurized accumulators that provide a rod insertion force far greater than any gravity or mechanical system.

The BWR requires substantially lower primary coolant flow through the core than pressurized water reactors (PWR's). The core flow of a BWR is the sum of the feedwater flow and the recirculation flow (typical of any boiler).

An important and unique feature of the GE-BWR Product Line is the application of jet pumps inside the reactor vessel. These pumps generate about two-thirds of the recirculation flow within the reactor vessel. The external recirculation loop is small and compact in comparison with PWR's, a feature that contributes to the inherent ease of maintenance of the BWR (see Figure 1-2). Recirculation pump power of the BWR is but a small fraction of the PWR pumping power (see Section 2). The jet pumps also contribute to the inherent safety of the BWR design under loss-of-coolant emergency conditions. Like most boilers, the BWR can deliver at least 10% power in a natural recirculation mode without operation of the recirculation pumps.

The BWR operates at constant pressure and maintains constant steam pressure similar to most fossil boilers. Most important, the BWR operates at steam pressure of about half the primary system pressure of a PWR, while producing steam of equal pressure and quality. The lower pressure and compact system is not only easier and less time-consuming to install, but also the lower pressure operation permits application of fast-acting and effective emergency core cooling systems (ECCS), described in Section 4.

The integration of the turbine pressure regulator and control system with the reactor water recirculation flow control system permits automated changes in steam flow to accommodate varying load demands on the turbine. Power changes of up to 25% of rated power can be accomplished automatically by recirculation flow control alone, thus providing automatic load-following capability for the BWR without altering control rod settings.

The nuclear boiler system is supported by the specialized functions of its auxiliary systems which are described in Section 4.

Several auxiliary systems are used for normal plant operation:

- Reactor water cleanup (RWCU) system
- Shutdown cooling function of residual heat removal (RHR) system
- Fuel building and containment pools cooling and filtering system
- Closed cooling water system for reactor service
- Radioactive waste treatment system

The following auxiliary systems are used as backup (standby) or emergency systems:

- Standby liquid control (SBLC) system
- Reactor core isolation cooling (RCIC) system
- Residual heat removal (RHR) system
  - Low pressure coolant injection (LPCI) function
  - Steam condensing function
  - Containment spray function
  - Suppression pool cooling function
- High pressure core spray (HPCS) system
- Low pressure core spray (LPCS) system
- Automatic depressurization function

Additional sections of this manual provide descriptions of other supporting systems and programs.

Section 2

Nuclear Boiler Assembly

INTRODUCTION

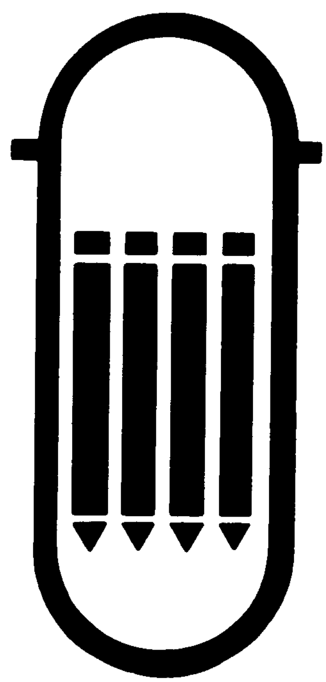
The nuclear boiler assembly consists of the equipment and instrumentation necessary to produce, contain, and control the steam power required by the turbine-generator. The principal components of the nuclear boiler are:

- **Reactor Vessel and Internals** — Reactor pressure vessel, jet pumps for reactor water recirculation, steam separators and dryers, core spray and feedwater spargers, and core support structure.
- **Reactor Water Recirculation System** — Pumps; control and equipment isolation valves; piping and its suspension devices, restraints, and suppressors; used in providing and controlling core flow.
- **Main Steam Lines** — Safety/relief and containment isolation valves; piping up to and including outboard containment isolation valve, and its restraints, suppressors, and guides.
- **Control Rod Drive System** — Control rods, control rod drive mechanisms, and hydraulic system for insertion and withdrawal of the control rods.
- **Nuclear Fuel and Instrumentation** are the subjects of separate sections of this manual (Sections 3 and 6).

REACTOR ASSEMBLY

The reactor assembly (Figure 2-1) consists of the reactor vessel, its internal components of the core, shroud, top guide assembly, core plate assembly, steam separator and dryer assemblies, and jet pumps. Also included in the reactor assembly are the control rods, control rod drive housings, and the control rod drives. Each fuel assembly that makes up the core rests on an orificed fuel support mounted on top of the control rod guide tubes. Each guide tube, with its fuel support piece, bears the weight of four assemblies and is supported by a control rod drive penetration nozzle in the bottom head of the reactor vessel. The core plate provides lateral guidance at the top of each control rod guide tube. The top guide provides lateral support for the top of each fuel assembly.

Control rods occupy alternate spaces between fuel assemblies and may be withdrawn into the guide tubes below the core during plant operation. The rods are coupled to control rod drives mounted within housings which are welded to the bottom head of the reactor vessel. The bottom-entry drives do not interfere with refueling operations. A flanged joint is used at the bottom of each housing for ease of removal and maintenance of the rod drive assembly.



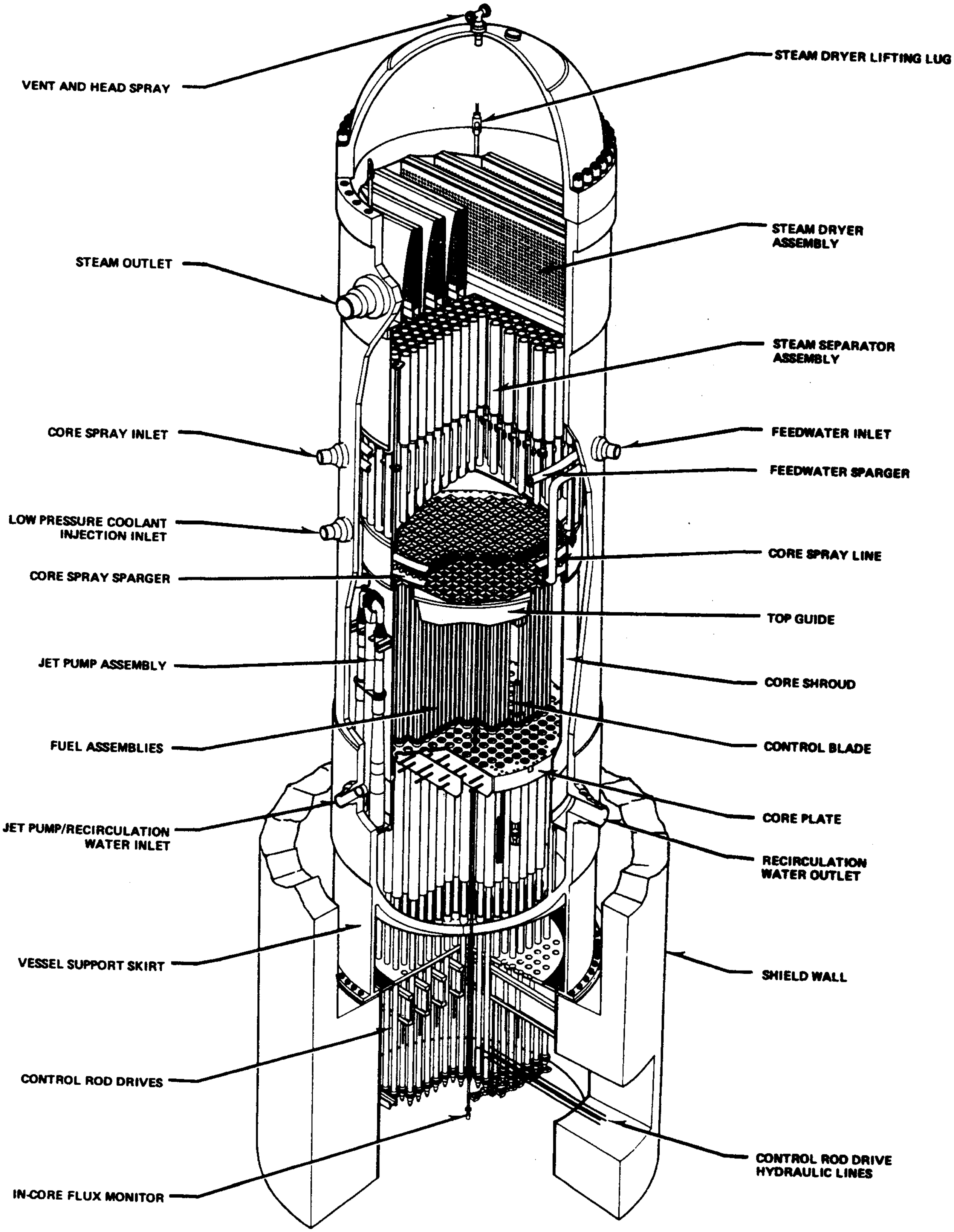


Figure 2-1. Reactor Assembly

Except for the Zircaloy in the reactor core, these reactor internals are stainless steel or other corrosion-resistant alloys. All major internal components of the reactor can be removed except the jet pump diffusers, the core shroud, the jet pump and high pressure coolant injection inlet piping. The removal of the top guide assembly and the core plate assembly is a major task and it is not expected that these components would require removal during the life of the plant. The removal of other components such as fuel assemblies, in-core assemblies, control rods, and fuel support pieces, is performed on a routine basis.

Reactor Vessel

The reactor vessel is a pressure vessel with a single full diameter removable head. The base material of the vessel is low alloy steel which is clad on the interior except for nozzles with stainless steel weld overlay to provide the necessary resistance to corrosion. Since the vessel head is exposed to a saturated steam environment throughout its operating lifetime, stainless steel cladding is not used over its interior surfaces.

Fine-grained steels and advanced fabrication techniques are selected to maximize structural integrity of the vessel. BWR vessels have the lowest neutron exposure of any light water reactor. Furthermore, the annulus space which carries recirculating water and feedwater downward between the core shroud and the vessel reduces radiation experienced by the vessel wall material. Vessel material surveillance samples are located within the vessel to enable periodic monitoring of exposure and material properties. Provisions are made for irradiating both tensile and impact specimens for a program of monitoring and evaluating radiation induced changes in vessel. Such programs have been conducted in most General Electric designed power reactors and considerable data have been accumulated on the performance of vessel materials after irradiation. The initial selection of high-quality materials, coupled with a continuing evaluation program, permits the vessel to meet the requirements of operability and safety throughout its design lifetime.

The vessel head closure seal consists of two concentric metal O-rings. This seal system has been demonstrated to perform without detectable leakage at all operating conditions. These conditions include cold hydrostatic testing, heating and cooling, and power operation. To monitor seal integrity, a leak detection system is used as discussed in Section 6.

Vessel supports, internal supports, their attachments, and adjacent shell sections are designed to take combined loads, including control rod drive reactions, earthquake loads, and jet reaction thrusts. The vessel is mounted on a supporting skirt which is bolted to a concrete and steel cylindrical vessel pedestal which is integral with the reactor building foundation.

Many features have been incorporated in the design of the vessel and its associated piping to simplify the refueling operation. Steam outlet lines are welded to the vessel body, thereby eliminating the need to break flanged joints in the steam lines when removing the head for refueling. Another design feature is the seal between the vessel and the surrounding drywell, which permits flooding of the space (reactor well) above the vessel.

Core Shroud

The shroud is a cylindrical, stainless steel structure which surrounds the core and provides a barrier to separate the upward flow through the core from the downward flow in the annulus. A flange at the top of the shroud mates with a flange on the top guide which in turn mates with a flange on the steam separator assembly to form the core discharge plenum. The jet pump discharge diffusers penetrate the peripheral shelf of the shroud support below the core elevation to introduce the coolant into the inlet plenum. The peripheral shelf of the shroud support is welded to the vessel wall to prevent the jet pump outlet flow from bypassing the core and to form a chamber around the core which can be reflooded in the event of a loss-of-coolant accident. The shroud support carries the weight of the shroud, the steam separators, the jet pump system, and the seismic and pressure loads both in normal and fault conditions of operation.

Two ring spargers, one for low pressure core spray and the other for high pressure core spray, are mounted inside the core shroud in the space between the top of the core and the steam separator base. The core spray ring spargers are provided with spray nozzles for the injection of cooling water. The core spray spargers and nozzles do not interfere with the installation or removal of fuel from the core. A nozzle for the injection of neutron absorber (sodium pentaborate) solution is mounted below the core in the region of the recirculation inlet plenum. The core sprays and standby liquid control systems are described in detail in Section 4.

Steam Separator Assembly

The steam separator assembly consists of a domed base on top of which is welded an array of standpipes with a three-stage steam separator located at the top of each standpipe. The steam separator assembly rests on the top flange of the core shroud and forms the cover of the core discharge plenum region. The seal between the separator assembly and core shroud flanges is metal-to-metal contact and does not require a gasket or other replacement sealing devices. The separator assembly is bolted to the core shroud flange, by long holddown bolts which, for ease of removal, extend above the separators. During installation, the separator base is aligned on the core shroud flange with guide rods and finally positioned with locating pins. The objective of the long-bolt design is to provide direct access to the bolts during reactor refueling operations with minimum-depth underwater tool manipulation during the removal and installation of the assemblies. It is not necessary to engage threads in making up the shroud head. A tee-bolt engages in the top guide and its nut is tightened to only nominal torque. Final loading is established through differential expansion of the bolt and compression sleeve. The fixed axial flow type steam separators have no moving parts and are made of stainless steel. In each separator, the steam-water mixture rising through the standpipe impinges on vanes which give the mixture a spin to establish a vortex wherein the centrifugal forces separate the water from the steam in each of three stages. Steam leaves the separator at the top and passes into the wet steam plenum below the dryer. The separated water exits from the lower end of each stage of the separator and enters the pool that surrounds the standpipes to join the downcomer annulus flow. An internal steam separator schematic is shown in Figure 2-2.

Steam Dryer Assembly

The steam dryer assembly is mounted in the reactor vessel above the steam separator assembly and forms the top and sides of the wet steam plenum. Vertical guides on the inside of the vessel provide alignment for the dryer assembly during installation. The dryer assembly is supported by pads extending inward from the vessel wall and is held down in position during operation by the vessel head. Steam from the separators flows upward and outward through the drying vanes. These vanes are attached to a top and bottom supporting member forming a rigid, integral unit. Moisture is removed and carried by a system of troughs and drains to the pool surrounding the separators and then into the recirculation downcomer annulus between the core shroud and reactor vessel wall. A schematic of a typical steam dryer panel is shown in Figure 2-3.

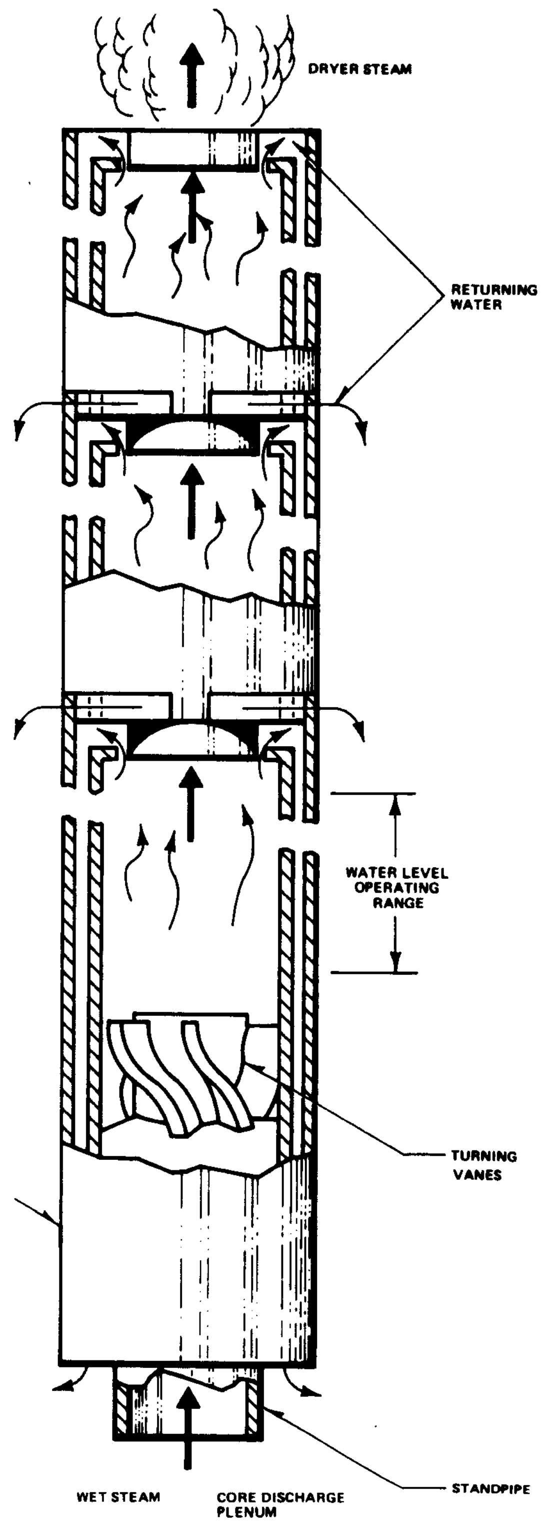


Figure 2-2. Internal Steam Separator

CONTROL ROD DRIVE BLOWOUT RESTRAINT STRUCTURE

The control rod drive blowout restraint structure is an engineered safeguard feature to prevent a nuclear transient in the highly unlikely event that there is a control rod drive housing failure which may cause a control rod to be expelled from the core.

The blowout restraint structure consists of restraint beams and restraint hardware. The restraint beams are located in the horizontal plane immediately below the reactor vessel bottom head between the rows of control rod drive housings and secured at their ends to the reactor vessel pedestal. The restraint hardware consists of support bars, grid plates, and grid clamps bolted together into a structure immediately below the mounting flanges of the control rod drives. The structure is hung from the restraint beams by hanger rods and their disc springs. Loads resulting from a blowout are transmitted to two grid plates, two support bars, four hanger rods and their disc springs, and two adjacent restraint beams.

The control rod drive housings are also restrained with respect to lateral seismic loadings by sets of adjustable spacer blocks (which are bolted to every other control drive housing flange) and by a beam attached to the reactor support pedestal.

REACTOR WATER RECIRCULATION SYSTEM

The function of the reactor water recirculation system is to circulate the required coolant through the reactor core. The system consists of two loops external to the reactor vessel, each containing a pump with a directly coupled water-cooled (air-water) motor, a flow control valve, and two shutoff valves.

High-performance jet pumps located within the reactor vessel are used in the recirculation system. The jet pumps, which have no moving parts, provide a continuous internal circulation path for a major portion of the core coolant flow. The system was first incorporated in the Dresden 2 plant design and since has been a standard part of the General Electric BWR product line.

Flow Path

The jet pump recirculation system (Figure 2-4) provides forced circulation flow through the reactor core.

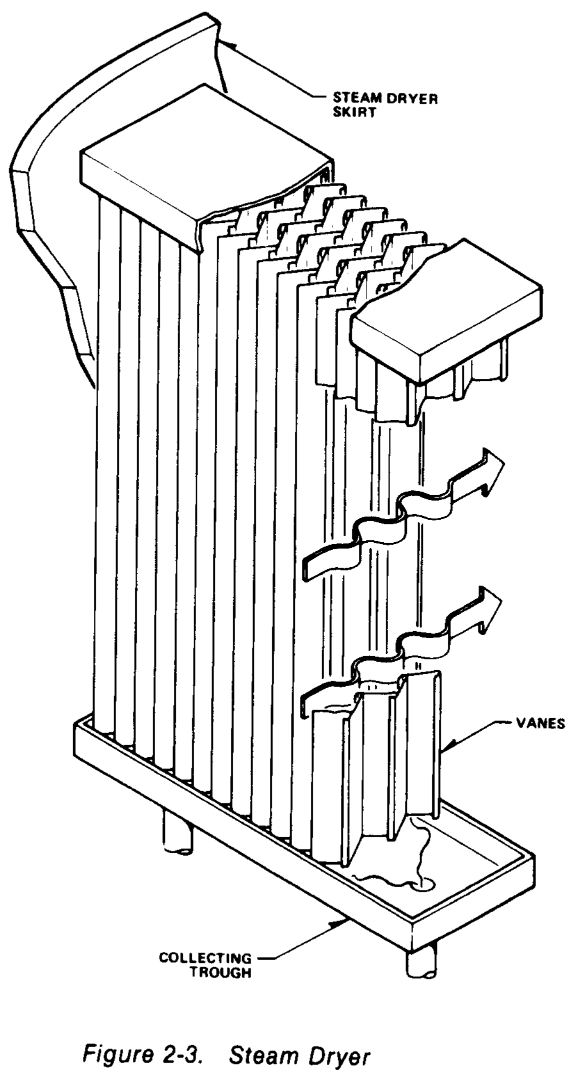


Figure 2-3. Steam Dryer

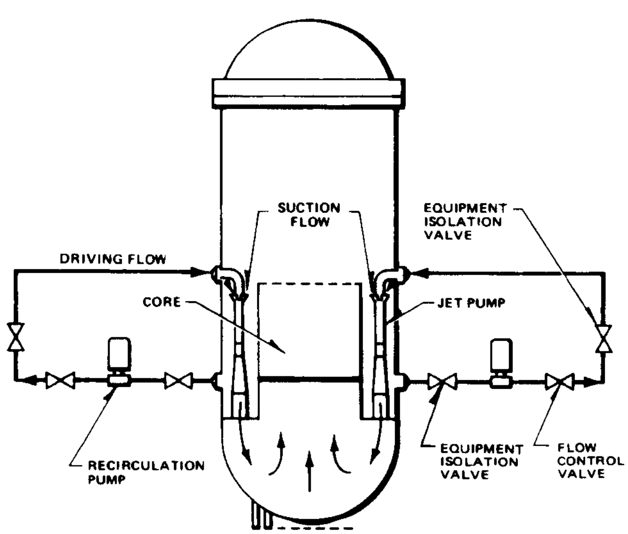


Figure 2-4. Jet Pump Recirculation System

The recirculation pumps take suction from the downward flow in the annulus between the core shroud and the vessel wall. Approximately one-third of the core flow is taken from the vessel through the two recirculation nozzles. There, it is pumped at a higher pressure, distributed through a manifold to which a number of riser pipes are connected, and returned to the vessel inlet nozzles. This flow is discharged from the jet pump nozzle into the initial stage of the jet pump throat where, due to a momentum exchange process, it induces surrounding water in the downcomer region to be drawn into the jet pump throat where these two flows mix and then diffuse in the diffuser, to be finally discharged into the lower core plenum.

The jet pump diffusers are welded into openings in the core shroud support shelf which forms a barrier between the lower plenum and the suction side of the jet pump. The flow of water turns upward, where it flows between the control rod drive guide tubes and enters into the fuel support where the flow is individually directed to each fuel bundle through the nosepiece. Orifices in each fuel support piece provide the desired flow distribution among the fuel assemblies. The coolant water passes along the individual fuel rods inside

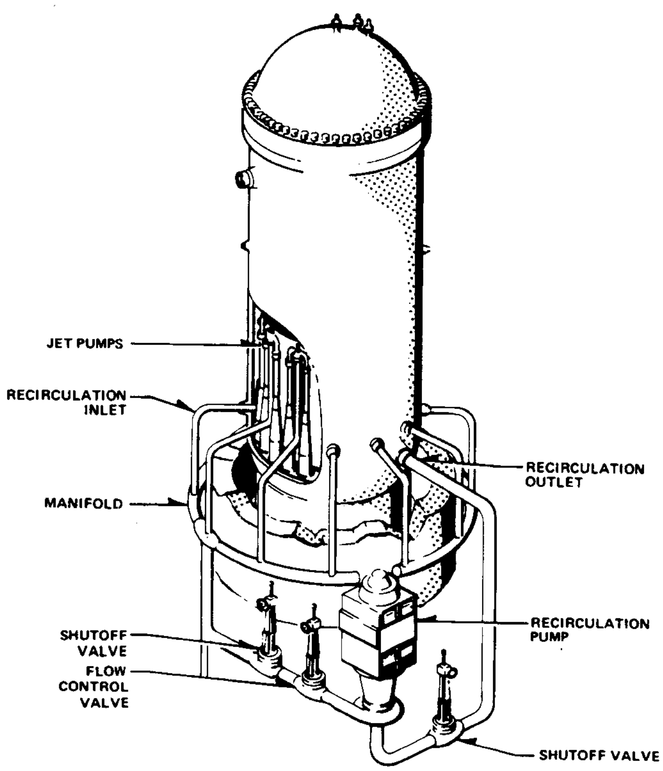


Figure 2-5. BWR Vessel Arrangement for Jet Pump Recirculation System

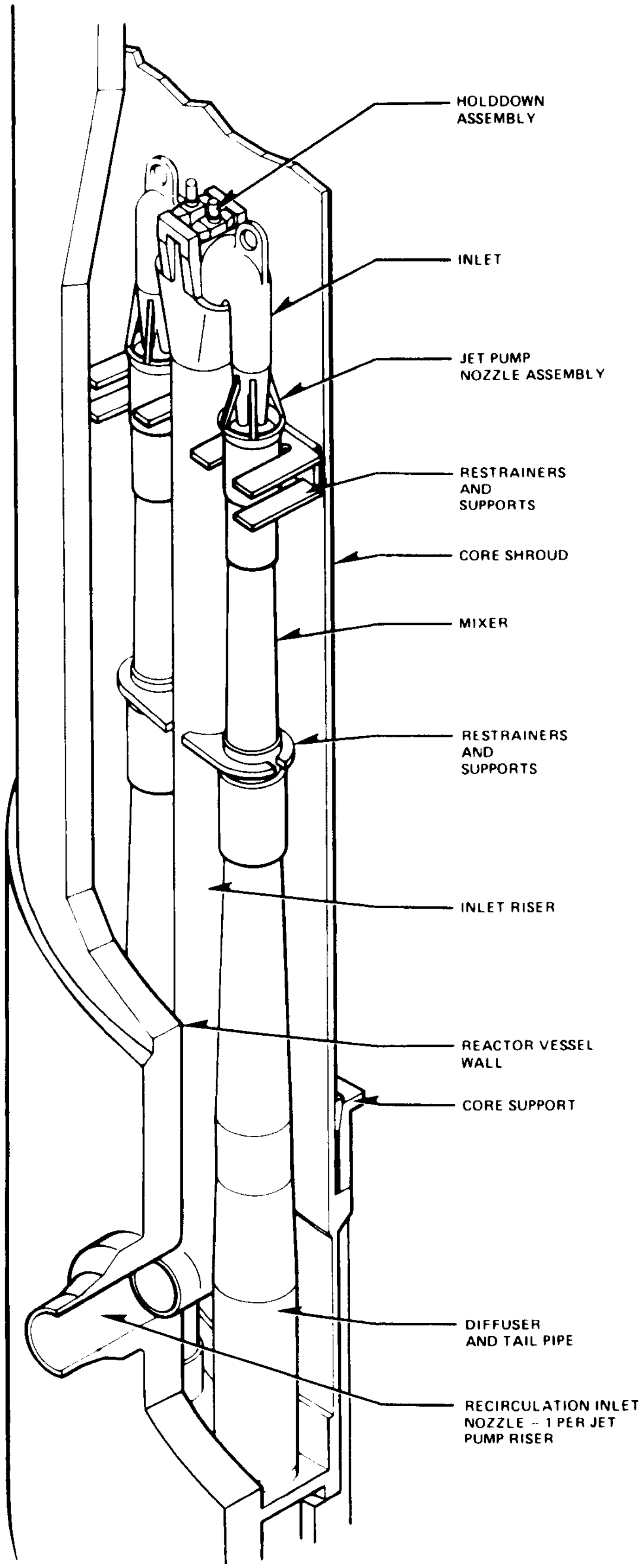


Figure 2-6. Jet Pump Assembly

the fuel channel where it is heated and becomes a two-phase, steam-water mixture. The steam-water mixture enters a plenum located directly above the core and bounded by the separator dome which opens to the separator array of fixed steam separators. The steam is separated from the water and passes through a dryer where any remaining water is removed. The saturated steam leaves the vessel through steam line nozzles located near the top of the vessel body and is piped to the turbine. Water collected in the support tray of the dryer is routed through drain lines, joins the water leaving the separators, and flows downward in the annulus between the core shroud and the vessel wall. Feedwater is added to the system through spargers located above the annulus and joins the downward flow of water. A portion of this downward flow enters the jet pumps and the remainder exits from the vessel as recirculation flow.

Jet Pump Applications In BWR's

The jet pumps are located in the annular region between the core shroud and the vessel inner wall (Figure 2-5). Each pair of jet pumps is supplied driving flow from a single riser pipe: these risers have individual vessel penetrations and receive flow from one of two external manifolds. Driving flow to each distribution manifold is furnished by its associated centrifugal pump. The recirculation system includes 16 to 24 jet pumps, depending on the size of the nuclear boiler system.

Description

Each jet pump assembly (Figure 2-6) is composed of two jet pumps and contains no moving parts. Each jet pump consists of an inlet mixer, a nozzle assembly with five discharge ports, and a diffuser.

The inlet mixer assembly, a replaceable component, is a constant-diameter section of pipe with a diffuser entrance section at the lower end and the drive nozzle at the upper end. The nozzle assembly can be removed by disconnecting the removable split flange.

The jet pump diffuser is a gradual conical section terminating in a straight cylindrical section at the lower end which is welded into the shroud support.

The overall length of the jet pumps is approximately 19 feet (5.8m).

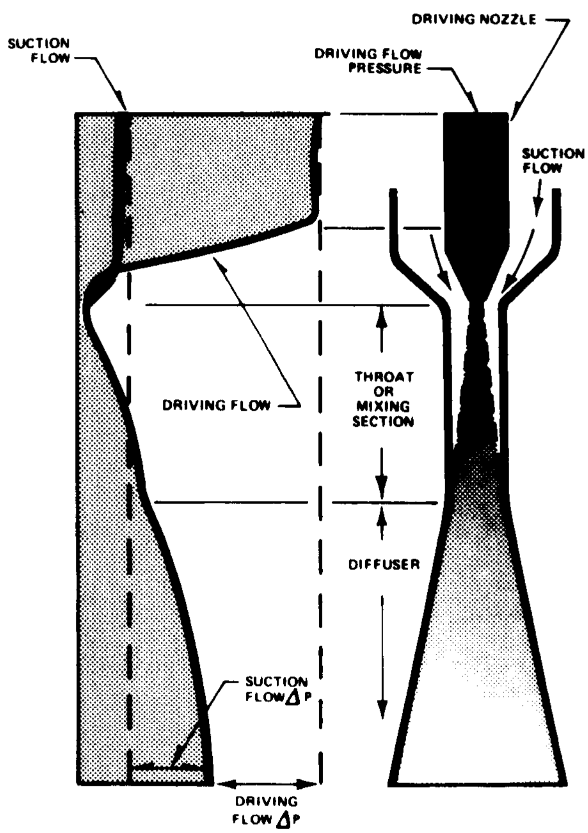


Figure 2-7. Jet Pump Principle

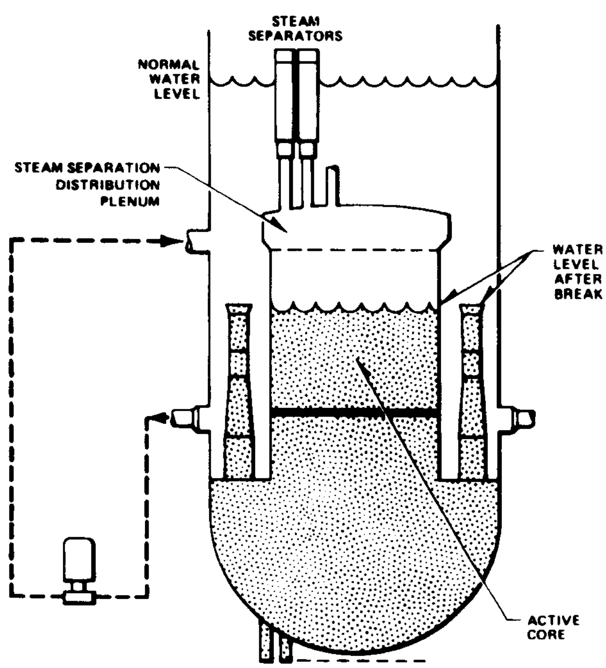


Figure 2-8. Core Submersion Capability of Jet Pump System

Instrumentation monitors jet pump flow passages to ascertain their individual and collective flow rates under varying operating conditions.

Jet Pump Operating Principle

The driving flow enters the nozzle section at a high pressure and is accelerated to a high velocity because of the constriction at the nozzle outlet. The suction flow enters at a low pressure which is further reduced as the flow is accelerated through the converging suction inlet nozzle. These two streams merge in the mixing section where a pressure rise occurs because of velocity profile rearrangement and the momentum transfer caused by the mixing. The rate of pressure rise decreases near the end of the mixing section because mixing is essentially completed. A diffuser is located downstream from the mixing section to slow the relatively high velocity mixed streams. This converts the dynamic head into static head. These processes are illustrated in Figure 2-7.

The jet pump system readily accommodates the full spectrum of flow rates required for load following.

Jet Pump Safety Features

The safety feature of post-accident core flooding capability with a jet pump design is depicted in Figure 2-8. This design allows flooding at no less than two-thirds of the core height. There is no recirculation line break which can prevent reflooding of the core to the level at the top of the jet pump.

Pumps and Motors

The reactor recirculation pumps are of the vertical, centrifugal, mechanical seal type and are constructed of stainless steel. The pumps operate at 25% of rated speed during startup and are powered from a low-frequency motor-generator set. Following startup, the pumps operate at constant speed and are powered from auxiliary power.

The pump shaft seal assembly consists of multiple mechanical seals built into a cartridge or cartridges which can be readily replaced with spare cartridges without removing the motor from the pump. Each of the seals carries an equal portion of the total pressure differential and each seal is capable of sealing against maximum pump operating pressure. A throttle bushing located in the pump casing minimizes leakage in the unlikely event of a gross failure of all shaft seals. The temperature of the seal cavity is controlled by cooling coils which circulate water from the closed cooling water system for reactor service. The temperatures of the cavity and cooling water for each pump are recorded, and on high temperature activate an alarm in the control room.

The drive motor for each pump is a vertical, water-cooled (air-water heat exchanger), totally enclosed, three-phase, squirrel-cage induction motor designed to operate at constant speed. Cooling water to the air-water cooler for motor windings cooling and through coils in the bearing oil reservoir for motor bearing cooling is provided from the closed cooling water system, described in Section 4. Temperature recorders and high-temperature alarms are located in the control room for motor windings, bearing oil reservoirs, and cooling water.

Valves and Piping

The recirculation loop piping is of welded construction. The piping, associated valves, and pumps are hung using constant-support hangers, thereby minimizing resultant stresses at the point of attachment to the reactor vessel. All recirculation piping is restrained to prevent pipe whipping as a result of jet action forces that might arise if a pipe break were to occur.

The shutoff and bypass valves are motor-operated gate valves and the flow control valve is a ball-type with electro-hydraulic actuator.

The flow control valve is on the discharge side of the pump. One shutoff valve is on the suction side of the pump, the other is downstream of the flow control valve. This allows maintenance in parallel with the refueling operation. No special reactor pressure vessel water level considerations are necessary. The stainless steel valves have double sets of valve stem packing to provide a highly reliable seal.

MAIN STEAM LINES

Steam exits from the vessel several feet below the reactor vessel flange through four nozzles. Carbon steel steam lines are welded to the vessel nozzles, and run parallel to the vertical axis of the vessel, downward to the elevation where they emerge horizontally from the containment. Two air-operated isolation valves are installed on each steam line, one inboard and one outboard of the containment penetration. The safety/relief valves are flange connected to the main steam line for ease of removal for test and maintenance.

A flow-restricting nozzle is included in each steam line as an additional engineered safeguard to protect against a rapid uncovering of the core in case of a break of a main steam line.

Safety/Relief Valves

The safety/relief valves are dual-function valves discharging directly to the pressure suppression pool. The safety function provides protection against overpressure of the reactor primary system. The relief function provides power-actuated valve opening to depressurize the reactor primary system. The valves are sized to accommodate the most severe of the following two pressurization transient cases determined by analysis:

- Turbine trip from turbine design power, failure of direct scram on turbine stop valve closure, failure of the steam bypass system, and reactor scrams from an indirect scram, or
- Closure of all main steam line isolation valves, failure of direct scram based on valve position switches, and reactor scrams from an indirect scram.

For the safety function, the valves open at spring set point pressure and close when inlet pressure falls to 96% of spring set point pressure.

For the pressure-relief function, the valves are power-actuated manually from the control room or power-actuated automatically upon high pressure. Each valve is supplied by separate power circuits. Valves which are power-actuated automatically upon high pressure close when pressure falls to a preset closing pressure. The pressure-relief function set point is below that for operation of valves for the safety (spring-actuated) function. By operating at the lower set point for the pressure-relief function, the reclosing set point of the valves provides a higher differential shutoff pressure than a spring reset valve, assuring leak tightness of the valves. The pressure-relief function may be used (in the event the main condenser is not available as a heat sink after reactor shutdown) to release steam generated by core decay heat until the residual heat removal system steam condensing function is initiated.

To limit the cycling of relief valves to one valve subsequent to their initial actuation during a main steam line isolation event, two valves (one a backup to the other) have the feature of automatically changing normal set pressures (opening and closing) following their initial actuation at normal set pressures to a lower level, thereby limiting the pressure cycles to a level where the other relief valves will not reopen. In conjunction with these two valves, the set pressure for the closing of the other valves is changed automatically which allows for them to stay open longer before closing to accommodate pressure swings. Manual valve operation and resetting of valve set pressure to their normal levels following the transient is by the control room operator.

Selected safety/relief valves are associated with the automatic depressurization of the primary system under assumed loss-of-coolant accident conditions. These valves have two independent logic channels powered from different power sources, either of which can initiate depressurization. The valves open automatically and remain open until the pressure falls to a preset closure pressure. These valves open automatically upon signals of high drywell pressure and low reactor water level and confirmation of one low pressure coolant injection function of the residual heat removal system or low pressure core spray system running. Initiation signals need not be simultaneous. The valves remain open until the primary system pressure is reduced to a point where the low pressure coolant injection function of the residual heat removal system and/or the low pressure core spray system can adequately cool the core. The initiation of automatic depressurization is delayed from 90 to 120 seconds to allow the operator to terminate the initiation should the high pressure core spray system initiation and acceptable reactor vessel level have been confirmed.

The valves used for automatic depressurization can be manually power actuated to open at any pressure. The signal for manual power actuation is from redundant control room switches from different power sources.

In the unlikely event that the residual heat removal shutdown suction line is unavailable during reactor shutdown to cool reactor water and during the period when the low pressure coolant injection function of the residual heat removal system and/or the low pressure core spray system pumps are injecting water into the reactor vessel, safety/relief valves used for automatic depressurization can be used to pass water from the reactor vessel to the suppression pool via valve discharge lines. For this to occur, the reactor vessel floods to a level above the vessel main steam line nozzles. Selected safety/relief valves are opened from the control room to pass reactor water to the suppression pool. See Section 4 for further descriptions of automatic depressurization.

Isolation Valves

Each steam line has two containment isolation valves, one inside and one outside the containment barrier. The isolation valves are spring loaded, pneumatic piston-operated globe valves designed to fail closed on loss of pneumatic pressure or loss of power to the pilot valves. Each valve has an air accumulator to assist in the closure of the valve upon loss of the air supply, electrical power to the pilot valves, and failure of the loaded spring. Each valve has an independent position switch initiating a signal into the reactor protection system scram trip circuit when the valve closes.

The isolation valves close upon (a) low water level in the reactor vessel, (b) high radiation from the steam line, (c) high flow rate in the main steam line, (d) low pressure at inlet to the turbine, (e) high ambient and differential steam line tunnel temperature (outside the containment), (f) low condenser vacuum (unless procedurally bypassed), and (g) high turbine building temperature. The signal for closure comes from two independent channels; each channel has two independent tripping sensors for each measured variable. Once isolation is initiated, valves continue to close and cannot be opened except by manual means. The valves may each be operated by independent remote-manual switches located in the control room. Lights in the control room indicate the position of the valve.

A shutoff valve is used in each steam line outboard of the external containment isolation valve and functions as a backup to the isolation valve. The shutoff valve is part of leakage control system to prevent possible release of nuclear steam which could leak through the main steam containment isolation valves following a loss-of-coolant event. Independent containment inboard and containment outboard divisions are used to establish a pressurized barrier between the containment barrier and the environs. Outleakage is effectively eliminated and inleakage is directed into the containment from the pressurized volume. Both divisions are powered from auxiliary and standby ac power. While either of the two divisions is sufficient to establish the necessary pressure barrier, both are initiated in the control room by a remote manual switch after it has been determined that a loss-of-coolant event has occurred. The system will not actually initiate unless the pressure levels of the air supply and the reactor vessel are within the permissive interlock set points.

The main steam line isolation valves remain closed if the steam line pressure is greater than the air pressure interlock set point. When the interlock is cleared, air is admitted to raise the pressure in the main steam lines to a predetermined level to establish the containment pressure boundary.

CONTROL ROD DRIVE SYSTEM

Description

Positive core reactivity control is maintained by the use of movable control rods interspersed throughout the core. These control rods thus control the overall reactor power level and provide the principal means of quickly and safely shutting down the reactor. The rods are moved vertically by hydraulically actuated, locking piston-type drive mechanisms. The drive mechanisms perform both a positioning and latching function, and a scram function with the latter overriding any other signal. The drive mechanisms are bottom-entry, upward-scramming drives which are mounted on a flanged housing on the reactor vessel bottom head. Here they cause no interference during refueling and yet they are readily accessible for inspection and servicing. Hydraulic connections to the drive mechanism are made at ports in the face of the housing flange.

The control rod drive system consists of a number of locking-piston control rod drive mechanisms, a hydraulic control unit for each drive mechanism, a hydraulic power supply for the entire system, and instrumentation and controls with necessary interconnections (Figure 2-9). The locking-piston-type control rod drive mechanism is a double-acting hydraulic piston which uses condensate water as the operating fluid. Accumulators provide additional energy for scram. An index tube and piston, coupled to the control rod, are locked at fixed increments by a collet mechanism. The collet fingers engage notches in the index tube to prevent unintentional withdrawal of the control rod, but without restricting insertion. The drive mechanism can position the rods at intermediate increments over the entire core length. The control rod can be uncoupled from below the vesel without removing the reactor vessel head, or with the vessel head removed for refueling, without removing the drive mechanism. Some of the advantages of the botttom-mounted drive arrangement are as follows:

- The drives do not interfere with refueling and are operative even when the head is removed from the reactor. Furthermore, this location makes them more accessible for inspection and servicing. Such an arrangement makes maximum use of the water in the reactor as a neutron shield, while yielding the least possible neutron exposure to drive components.

- The locking piston drive provides the highest scram forces and operating force margins of all known types of drive mechanisms. This provides high operational reliability, particularly in the scram function.

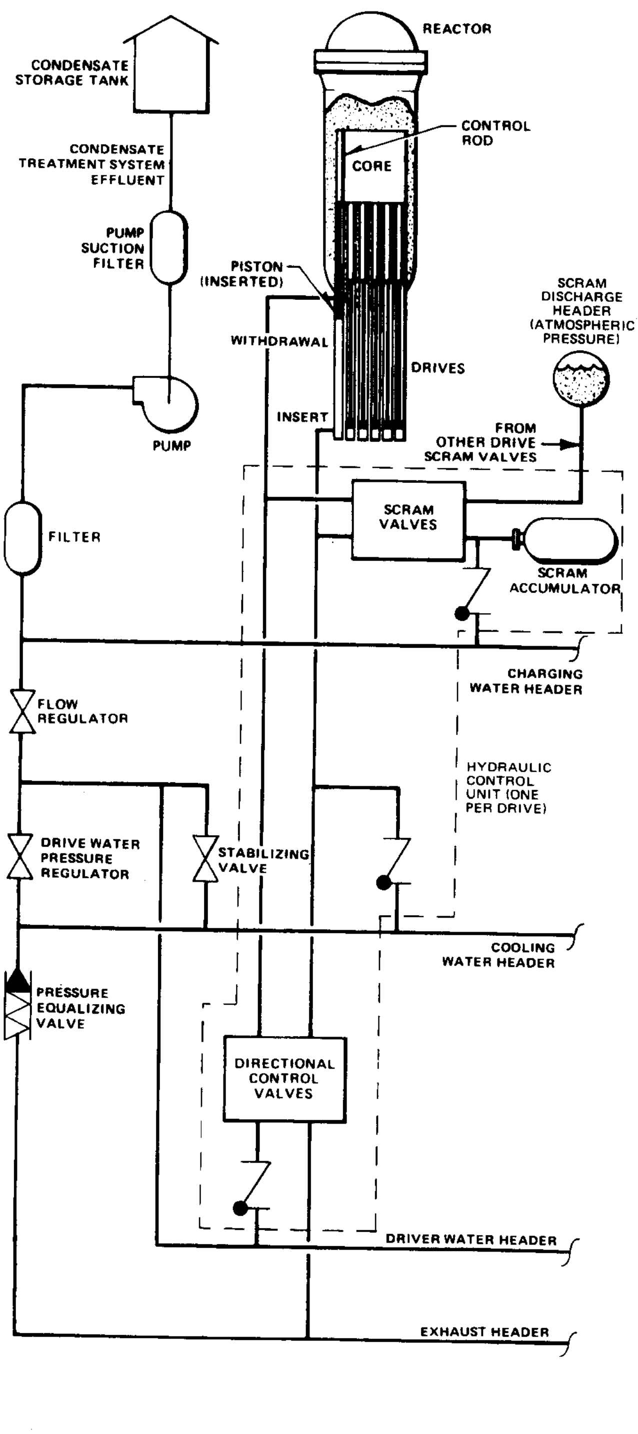


Figure 2-9. Basic Control Rod Drive System

- The use of water of reactor quality as the operating fluid eliminates the need for special hydraulic systems, with their inherent leakage and maintenance problems.
- The continuous in-flow of high-purity water through the drives minimizes the contamination deposits within the drives from foreign material that may be in the reactor vessel.
- By using high-purity water as the operating fluid, the drives can use simple internal piston seals which allow leakage into the reactor vessel. Dynamic shaft or push rod seals and their attendant systems and wear problems are eliminated.
- Control rod entry from below the core provides the best axial flux shaping and resultant fuel economy for the boiling water reactor.

The number of drives supplied with a reactor is selected to give the optimum power distribution in the core and to give the operator the maximum degree of control flexibility during startup, maneuvering, and flux shaping.

Design Background

Locking piston drives have been extensively and successfully used in boiling water reactors since 1959 and have accumulated millions of hours of operating time during this period. A total of 354 hydraulic control rod drives had been manufactured prior to the initiation of efforts to produce a standard drive. These original drives are still in service in six operating plants.

The current design is essentially the "third generation" of the locking piston drive concept. It is a standard model with identical units being supplied to General Electric boiling water reactors now operating or under construction. There are more than 4800 drives of standard design now operating on reactors in commercial service and more than 5000 in various stages of manufacture, installation, and testing.

As of early 1980, the 4800 drives operating in 40 reactors in commercial service represented in excess of 28,000 drive-years of operating experience. These totals represent the largest totals of any control rod drive design used in the electrical power-producing industry.

High reliability has been demonstrated by these drives in plants using the standard model; there have been no forced outages caused by control rod drive mechanisms after the beginning of commercial service.

Operating experience has also demonstrated that the design life of the major and expendable parts is being met.

Standardization of control rod drives provides the following benefits:

- It has made it possible to concentrate effort on features affecting reliability and maintainability.
- It has allowed extensive and thorough evaluation through tests such as those described in the following paragraphs.
- It ensures the ready availability of spare and replacement parts in the future.

Test Background

In addition to the extensive experience record accumulated with the manufacture and operation of 4800 drives now in operation, all drives are subjected to extensive manufacturing and preoperational tests.

Factory Quality Control Tests — Each drive mechanism, including the sampling assigned for prototype testing, is subjected to a standard quality control test at reactor pressure conditions. Drive shim motion and latching, including proper position indication, are verified. After completion of this test, drive friction is determined to ensure proper operation and to verify proper alignment and concentricity. Each drive is then subjected to cold scram tests to determine scram times at various reactor pressures and to verify stepping performance.

In addition to normal dimensional inspection, material verification, heat-treatment control, weld inspection, etc., standard quality control tests and examinations include:

- Hydrostatic testing of each drive is performed to verify pressure welds in accordance with ASME Code.
- Electrical components are tested for electrical continuity and resistance to ground.
- All drive parts which cannot be visually inspected for dirt are flushed with filtered water (no foreign material is permissible in effluent water) at high velocity.
- Seal leakage tests are performed to demonstrate proper drive assembly.

To preserve their factory condition, CRD's are shipped to the site in watertight reusable metal containers.

Reactor Preoperational Tests — After the drives are installed in the reactor, a final series of tests is performed to make final adjustments in the hydraulic system and to ensure proper installation of both the hydraulic system and the drives. Readings and observations taken at this time constitute the basis for evaluating any performance changes in actual reactor service and their relationship to maintenance life.

CONTROL ROD DRIVE HYDRAULIC SUPPLY SYSTEM

The control rod drive hydraulic supply system, located in the fuel building, supplies pressurized demineralized water to the control rod drive system to provide hydraulic operating pressure and cooling water for the drive mechanisms. The system is made up of high-head, low-flow pumps and the necessary piping, filters, control valves, and instrumentation. Two 100% capacity pumps are used, one of which is a standby spare. The pumps take suction from the condensate treatment system effluent.

Pressure Control

Condensate treatment system effluent is pumped to a pressure of 1800 psi (12,400 kPa) by a multistage centrifugal pump with a full-capacity standby pump available. After filtering, water approximately 1750 psi (12,000 kPa) charges the scram accumulators. The flow regulator automatically maintains a constant flow to the system. During normal operation periods when rod drive movement is not required, the bypass flow provides cooling for each of the drives by way of the cooling water header, with the excess water and exhaust water generated by normal drive motion going to the reactor vessel by reverse flow through deenergized insert exhaust solenoid valves in hydraulic control units of non-moving valves. The cooling pressure regulator valve is manually adjusted to give the desired pressure at the cooling water header to maintain proper cooling of the drive mechanisms.

The drive pressure regulator valve maintains the correct pressure on this drive water header for control rod positioning. It is also manually adjustable. During periods when a drive mechanism is in motion, a stabilizing valve is automatically closed to reduce the bypass flow by the amount required to move the drive, thus maintaining the pressure balance of the system.

System Operation

Basic components of the hydraulic system for controlling the drive mechanism during positioning and scram operation are shown in Figure 2-10. All components shown are typical for each rod drive.

The main movable element of the system consists of the main drive piston, the index tube and the control rod coupled to the index tube. The movable element is held in any chosen position by a collet which engages one of several notches in the index tube. Gravity holds the tube notch against the latch since the entire mechanism is essentially at reactor pressure. The control rod is moved by applying a pressure greater than reactor pressure to either the top or bottom of the main drive piston. When the reactor protection system calls for a reactor scram, all control rods are driven into the core at the maximum rate of speed.

When a scram signal is initiated, control air is vented from the scram valves, allowing them to open by spring action. Opening the exhaust valve reduces the pressure above the main drive piston to atmospheric pressure and opening the inlet valve applies the accumulator pressure of 1750 psi (1200 kPa) to the bottom of the piston. Since the notches in the index tube are tapered on the lower edge, the latch is forced open by cam action, allowing the index tube to move upward without restriction under influence of the high pressure differential across the piston. As the drive moves upward and accumulator pressure reduces to reactor pressure, the ball check valve opens, letting reactor water complete the scram action. If reactor pressure is low, such as during startup, the accumulator will fully insert the rod in the required time without assistance from reactor pressure.

ROD CONTROL AND INFORMATION

The primary purpose of the rod control and information function is to effect control rod motion as requested by the operator. It displays all information which is relevant to the movement of rods. In addition to enabling the operator to move rods, the function also enforces adherence to operating restrictions which limit the consequences of a potential rod drop accident. At higher power levels it limits rod movements so that rods cannot be withdrawn to the point of generating excessive heat flux in the fuel. Unit conditions are considered in determining which restrictions are applied to a given rod movement request.

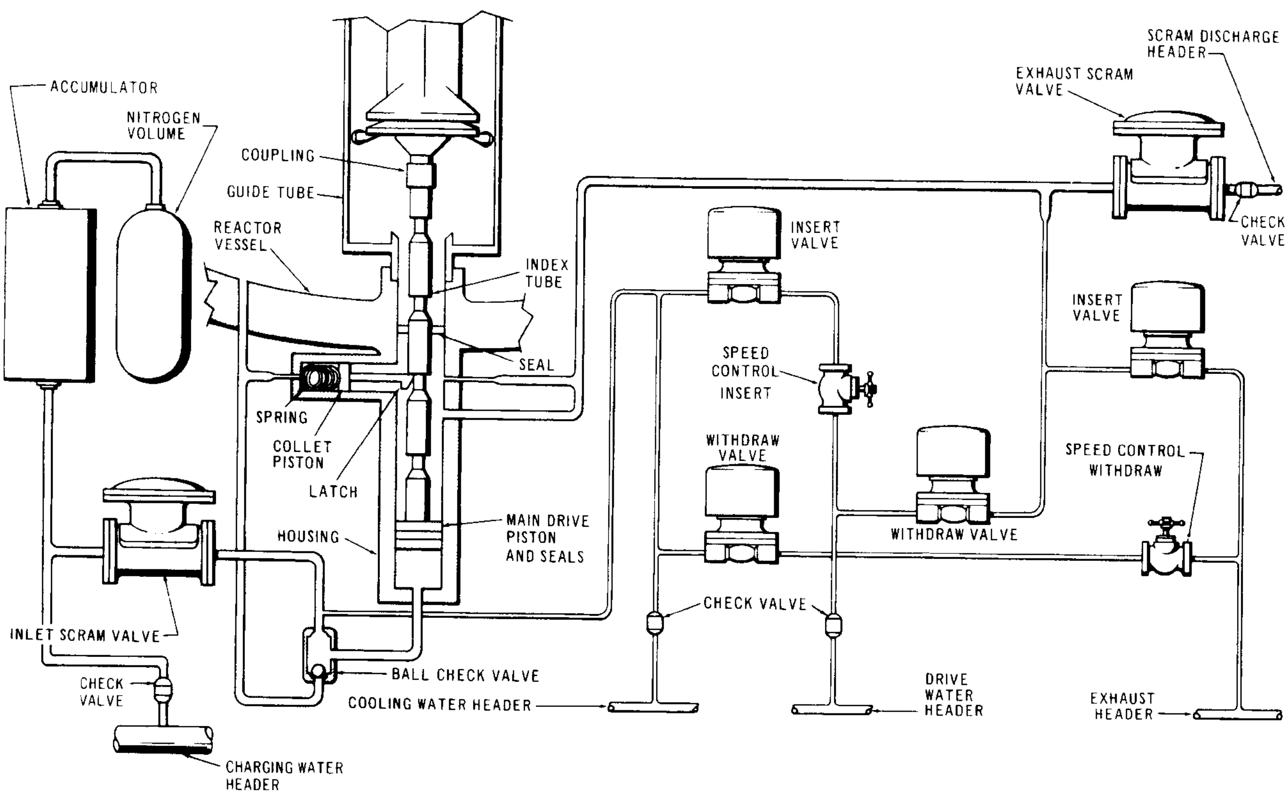


Figure 2-10. Locking Piston Drive System

Position

Rod position is sensed by a series of sealed glass reed switches contained within a tube inside the drive piston. Two switches are spaced every 3 inches with each of the dual switches feeding a separate channel. These signals are multiplexed inside the containment and transmitted to the control room. The rod position information function decodes these data and makes them available to other parts of the rod control and information function, to the Unit process computer and to the operator. An invalid input caused by a failed reed switch is detected and made known to the operator who can take remedial action. The status of the scram valves and accumulators on the hydraulic control unit is monitored and these data are available to the operator and the computer.

Ganged Rods

The speed and capacity of the rod control and information function permit the control of more than one rod at a time. Up to four rods can be operated simultaneously. The position of each rod in a gang is monitored. In the continuous movement ganged rod mode, the rods move together so that all rods are within two notches of all other rods in the gang. Ganged rod operation effects significant time saving during hot restart and other large control maneuvers.

Operation

The following description of the operation of the rod control and information function follows the flow of data as the operator selects and drives a rod. Prior to the operator selecting a rod, certain data are being transferred. Rod position information is one parameter which is being continuously monitored. Starting at the control rod drives, rod data are taken from the magnetic reed switches of the rod position probes. These data are duplicated (i.e., two switches per position). These raw data are wired to a position multiplexer which is located inside the containment. Data are transmitted to the scan control portion of the rod position information function which makes them available to other functions as required. The scan control is also capable of recognizing an invalid input which would result from a failed reed switch.

When the operator selects a rod (or rods) the rod interface function transforms that request into suitable electronic form and transmits the request word to the rod activity control function. (Note that this is done for two parallel channels.) The rod activity control function looks at the present control rod pattern as provided by the rod position information function, and also looks at the Unit status (i.e., mode and power level). It then examines its pattern control memory to ascertain whether movement of the requested rod is allowed under existing conditions. If motion is permitted, the rod activity control function then generates a command word and transmits it to the rod gang drive function. If the two rod activity control channels agree with each other, the rod gang drive function passes the command word through to the transponder cards on the individual control rod drive hydraulic control units. The transponder on the selected drive acknowledges that the command word has been received and acted upon. As the rod moves, the rod position and information function transmits its position to the rod activity control function, and then to the rod gang drive function and to the operator.

Control Rod Withdrawal Features

Control rods cannot be withdrawn when any input from redundant sources indicates a block condition. When a block condition occurs, the rod settle cycle is initiated to avoid spurious rod drift indications.

Continuous rod withdrawal may be effected only by deliberate manipulation of two independent control switches by the operator.

Interlocks prevent more than one control rod from being withdrawn from its fully inserted position at any time during refueling.

Drive Design

The mechanical arrangement of the drive mechanism itself is illustrated in Figure 2-11 which shows the important elements in the drive unit. In comparing this with Figure 2-10, the following should be noted:

- The area above the main piston is actually inside the index tube. The moving piston is hollow and moves in an annulus between the stationary cylinder and the stationary piston tube.
- The inside of the stationary piston tube is connected to the scram hydraulic passage. It is connected to the area above the piston by orifices in the piston tube. These orifices are progressively cut off by the main piston seals at the end of the scram stroke to decelerate the rod.
- The latch mechanism is made of six collet fingers attached to an annular collet piston operating in an annular cylinder surrounding the index tube. The

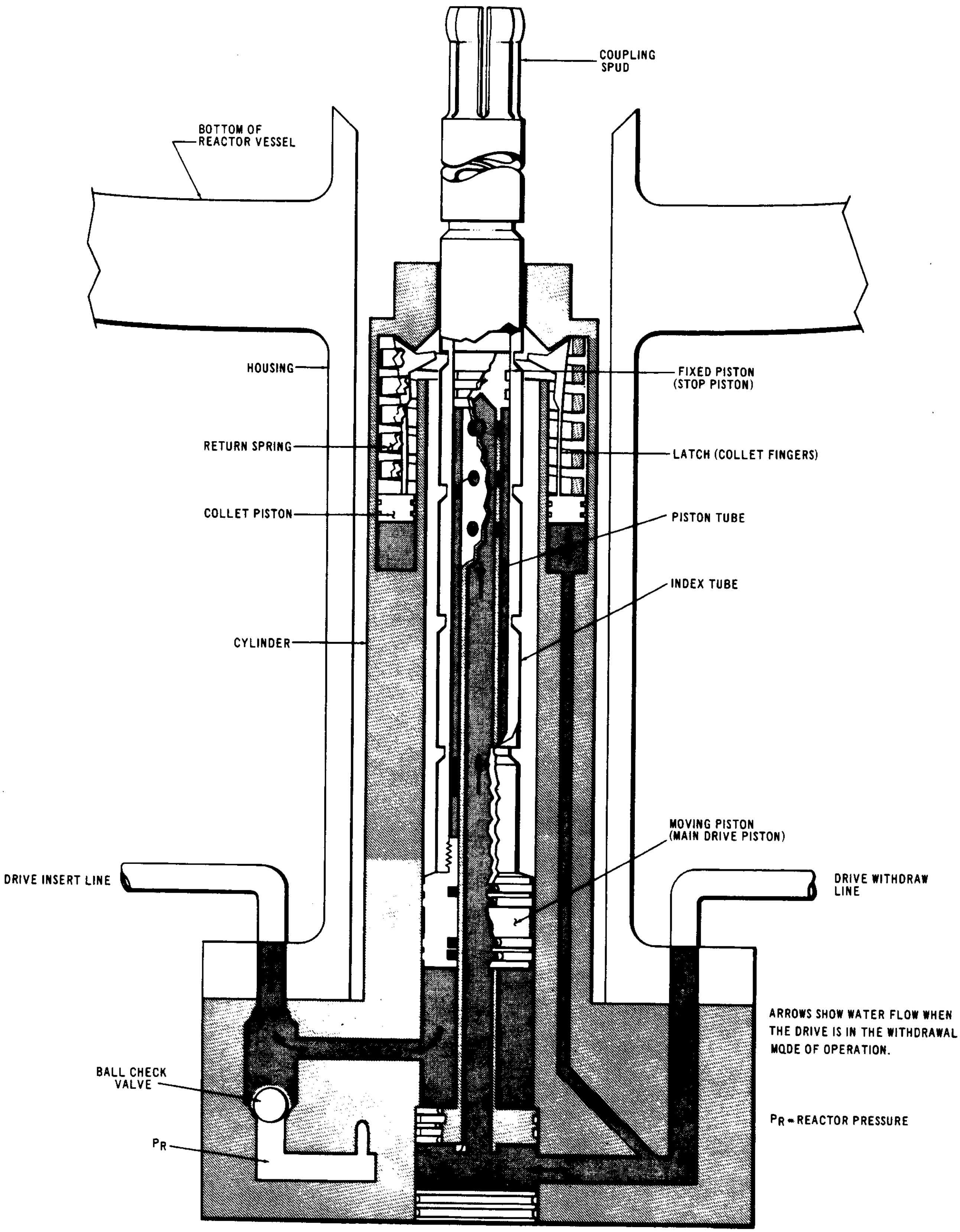


Figure 2-11. Locking Piston Hydraulic Drive

spring action of the fingers holds them against the index tube. To unlatch the rod, pressure must be applied to the collet piston which drives the fingers up against the guide cap and cams them outward free of the index tube. This mode of operation is shown in Figure 2-11.

- The annular passage between the drive and the housing becomes the hydraulic passage which connects reactor pressure to the bottom of the large check valve and, hence, to the bottom of the main piston. This construction ensures that the reactor pressure is always available to the lower end of the piston for a scram and that the drive can be actuated in the withdraw direction only by a pressure higher than reactor pressure.

The control drive-rod coupling, shown in Figure 2-12, contains unique features which permit the drive to be uncoupled from the control rod either from below the reactor (without removing the reactor vessel head) or from above the reactor without draining all the coolant from the reactor vessel.

Hydraulic Control Unit

The operating valves, accumulator, scram valves, and associated piping for each locking piston drive are combined into single hydraulic control unit which is preassembled as a module and tested prior to shipment (Figure 2-13). The modular construction contributes to the demonstrated reliability of the CRD system and has the following advantages:

- The unit is more compact than the equivalent combination of components installed in the field and therefore occupies less building space.
- Installation work is minimized, allowing shorter construction schedules.
- All valve and pressure switch settings are accomplished prior to shipment, thereby minimizing the time required for preoperational testing. Similarly, each component and associated piping is hydrotested prior to shipping.
- To facilitate maintenance of components, isolation valves are used so that all major components can be removed (except the scram valve bodies) without requiring a plant shutdown.
- All components in the design have been selected based on extensive engineering tests or millions of hours of accumulated field experience.

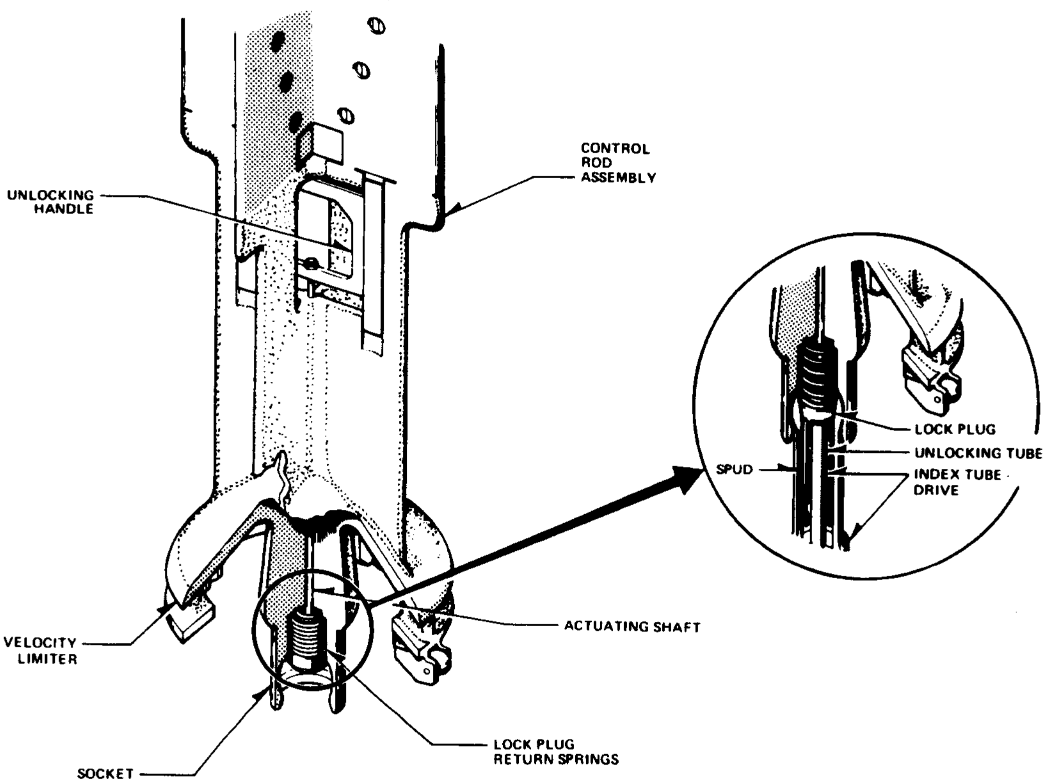


Figure 2-12. Control Rod Coupling

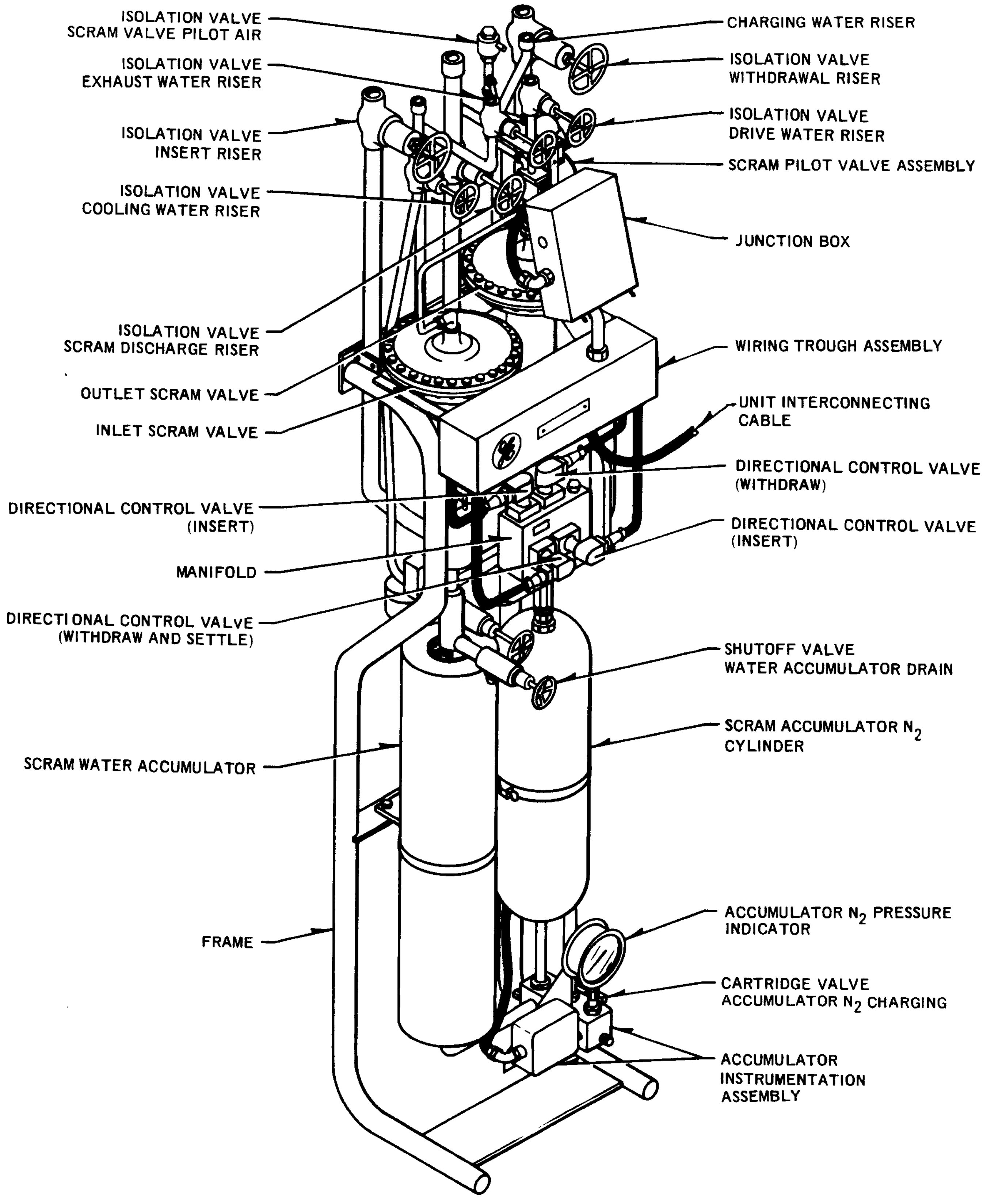


Figure 2-13. Control Rod Drive Hydraulic Control Unit

Section 3

Reactor Core Design

INTRODUCTION AND SUMMARY

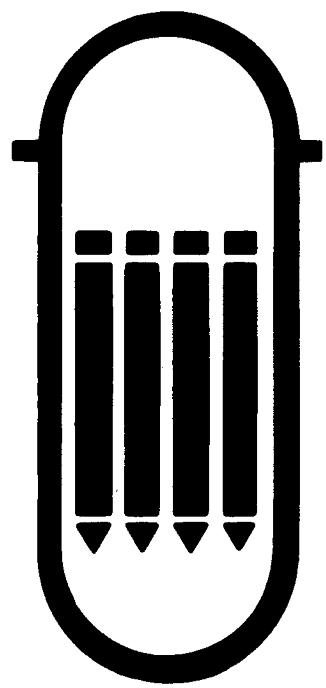
The design of the boiling water reactor core and fuel is based on the proper combination of many design variables and operating experience. These factors contribute to the achievement of high reliability, excellent performance, and improved fuel cycle economy.

Discussed in this Section are such design parameters as moderator-to-fuel volume ratio, core power density, thermal-hydraulic characteristics, fuel exposure level, nuclear characteristics of the core and fuel, heat transfer, flow distribution, void content, cladding stress, heat flux, and the operating pressure.Design analyses and calculations employed in this scope design have been verified by comparison with data from operating plants. General Electric continually evaluates this combination of design variables to be certain that changing conditions, which may significantly affect fuel cycle economics, are properly considered and that the resulting final core design represents an optimum combination of the variables.

The reactor core mechanical structure, lattice configuration, and fuel element design are basically the same as employed in General Electric designed plants at Oyster Creek, Nine Mile Point, Nuclenor, Quad Cities 1 and 2, Fukushima 2, and Brown's Ferry 1, 2, and 3. Previous plants representative of the BWR 1 class such as Dresden 1, Humboldt Bay, KRB (Gundremmingen), and Tarapur differ only slightly. The design has remained constant through the many representatives of the BWR/3, /4, and /5 classes.

A number of important features of the boiling water reactor core design are summarized in the following paragraphs:

- The BWR core mechanical design is based on conservative application of stress limits, operating experience, and experimental test results. The moderate pressure levels characteristic of a direct cycle reactor [approximately 1000 psia (6900 kPa)] reduce cladding temperatures and stress levels.
- The low coolant saturation temperature, high heat transfer coefficients, and neutral water chemistry of the BWR are significant, advantageous factors in minimizing Zircaloy temperature and associated temperature-dependent corrosion and hydride buildup. This results in improved cladding performance at long exposures. The relatively uniform fuel cladding temperatures throughout the BWR core minimize migration of the hydrides to cold cladding zones and reduce thermal stresses.



- The basic thermal and mechanical criteria applied in the BWR design have been proven by irradiation of statistically significant quantities of fuel. The design heat fluxes and linear thermal outputs [approximate maximum of 13.4 kW/ft (44 kW/m)] are similar to values proven in fuel assembly irradiation.
- The design power distribution used in sizing the core represents a worst expected state of operation. Provisions for nonoptimum operation allow operational flexibility and reliability.
- The reactor is designed so that the peak bundle power at rated conditions is significantly less than the critical power limit.
- Because of the large negative moderator density (void) coefficient of reactivity, the BWR has a number of inherent advantages. These are the use of coolant flow as opposed to control rods for load following, the inherent self-flattening of the radial power distribution, the ease of control, the spatial xenon stability, and the ability to override xenon in order to follow load.

  The inherent spatial xenon stability of the BWR is particularly important for large-sized plants. For example, the Dresden 1 reactor has been operated for several years on full power during the day and half power at night load schedule. This produces maximum xenon concentration gradients, yet no xenon instabilities have been observed.

CORE CONFIGURATION

The reactor core of the boiling water reactor is arranged as an upright cylinder containing a large number of fuel assemblies and located within the reactor vessel. The coolant flows upward through the core. A typical core arrangement (plan view) of a large boiling water reactor and the lattice configuration are shown in Figures 3-1 and 3-2. Important components of this arrangement are described in the following pages.

FUEL ASSEMBLY DESCRIPTION

As can be seen from Figures 3-1 and 3-2, the boiling water reactor core comprises essentially only two components: fuel assemblies and control rods. The fuel assembly and control rod mechanical designs are basically the same as used in Dresden 1 and in all subsequent General Electric boiling water reactors.

Fuel Rod

A fuel rod consists of UO_2 pellets and a Zircaloy 2 cladding tube. The UO_2 pellets are manufactured by

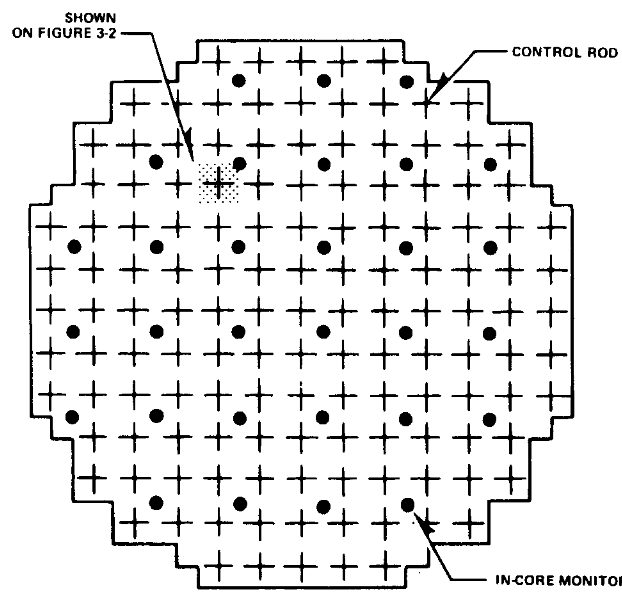


Figure 3-1. Typical Core Arrangement

compacting and sintering UO_2 powder in cylindrical pellets and grinding to size. The immersion density of the pellets is approximately 95% of theoretical UO_2 density. A sample of BWR fuel pellets is shown in Figure 3-3.

A fuel rod is made by stacking pellets into a Zircaloy 2 cladding tube which is evacuated, back-filled with helium to 3 atmospheres pressure, and sealed by welding Zircaloy end plugs in each end of the tube. Typical end plugs and welded tube ends are shown in Figure 3-4. The Zircaloy tube is 0.483 inch (12.3mm) o.d., 160-1/4 inches (4.07m) long, with a 32-mil (0.81mm) wall thickness. The pellets are stacked to an active height of 150 inches (3.8m), with the top 9.5 inches (241mm) of tube available as a fission gas plenum. A plenum spring is located in the plenum space to exert a downward force on the pellets; this plenum spring keeps the pellets in place during the pre-irradiation handling of the fuel bundle. The selected dimensions result in a 9-mil (0.23mm) diametral gap between the pellet and the tube.

Design Basis of Fuel Rods

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

The rod is designed to withstand the applied loads, both external and internal. The fuel pellet is sized to provide sufficient volume within the fuel tube to accommodate differential expansion between fuel and cladding. Overall fuel rod design is conservative in its accommodation of the mechanisms affecting fuel in a BWR environment.

Fuel Bundle

Each fuel bundle contains 64 rods which are spaced and supported in a square (8 by 8) array by lower and upper tie plates. The lower tie plate has a nosepiece which fits into the fuel support piece and distributes coolant flow to the fuel rods. The upper tie plate has a handle for transferring the fuel bundle. Both tie plates, shown in Figures 3-5 and 3-6, are fabricated from stainless steel and are designed to satisfy both flow and mechanical strength considerations. Mechanically, these parts are designed to stay within the yield strength of the material during normal handling operations.

Three types of rods are used in a fuel bundle: tie rods, water rods, and standard fuel rods. The third and sixth fuel rods along each outer edge of a bundle are tie rods. The eight tie rods in each bundle have threaded end

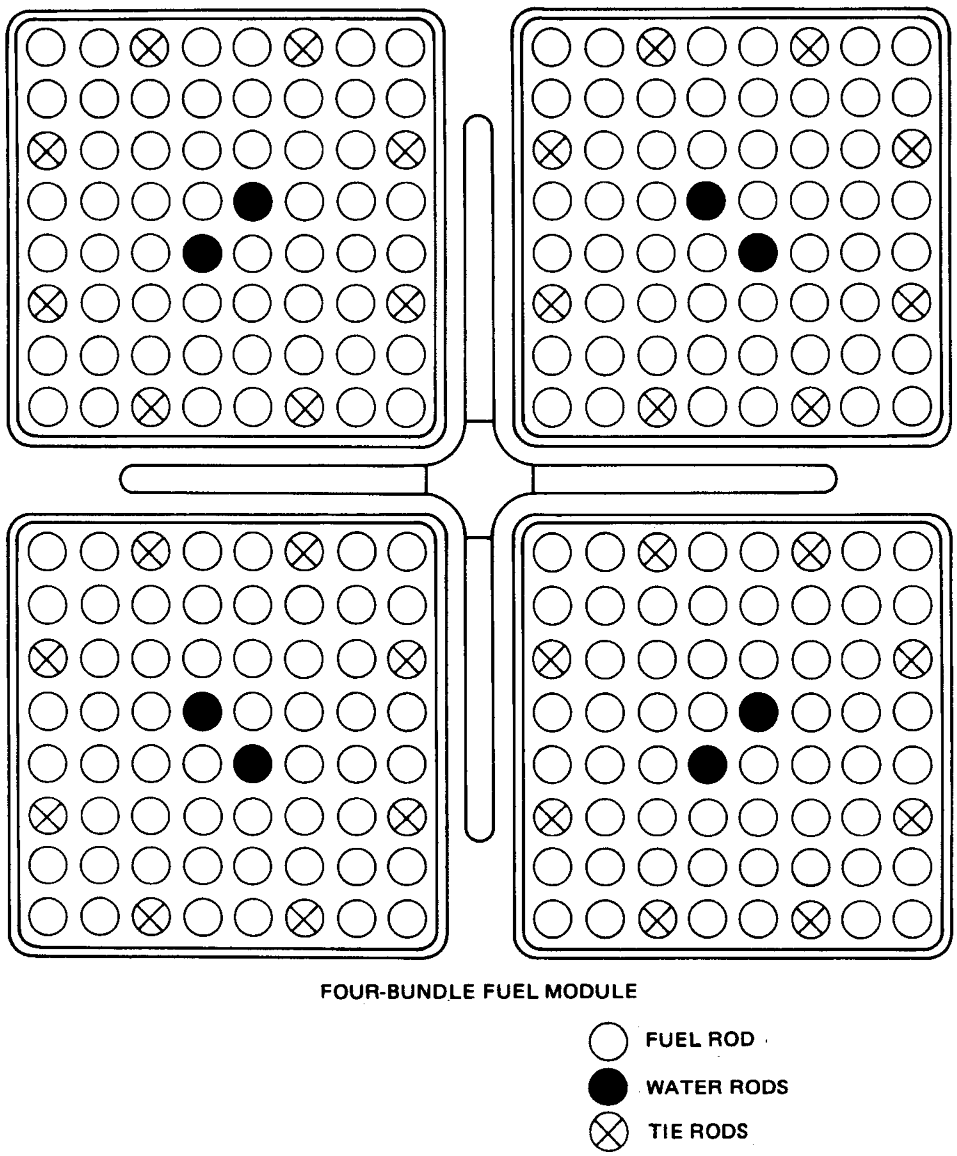


Figure 3-2. Core Lattice

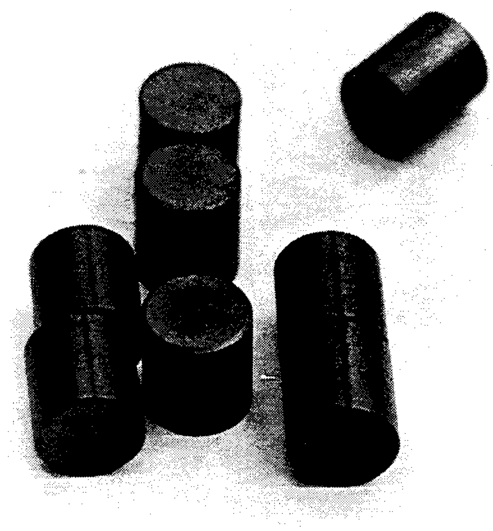


Figure 3-3. Typical BWR Fuel Pellets

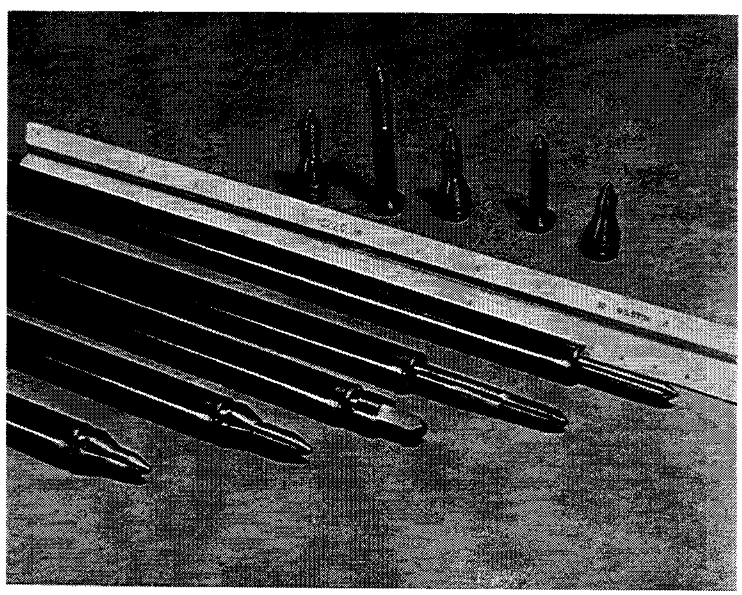


Figure 3-4 Typical End Plugs and Welded Tube Ends

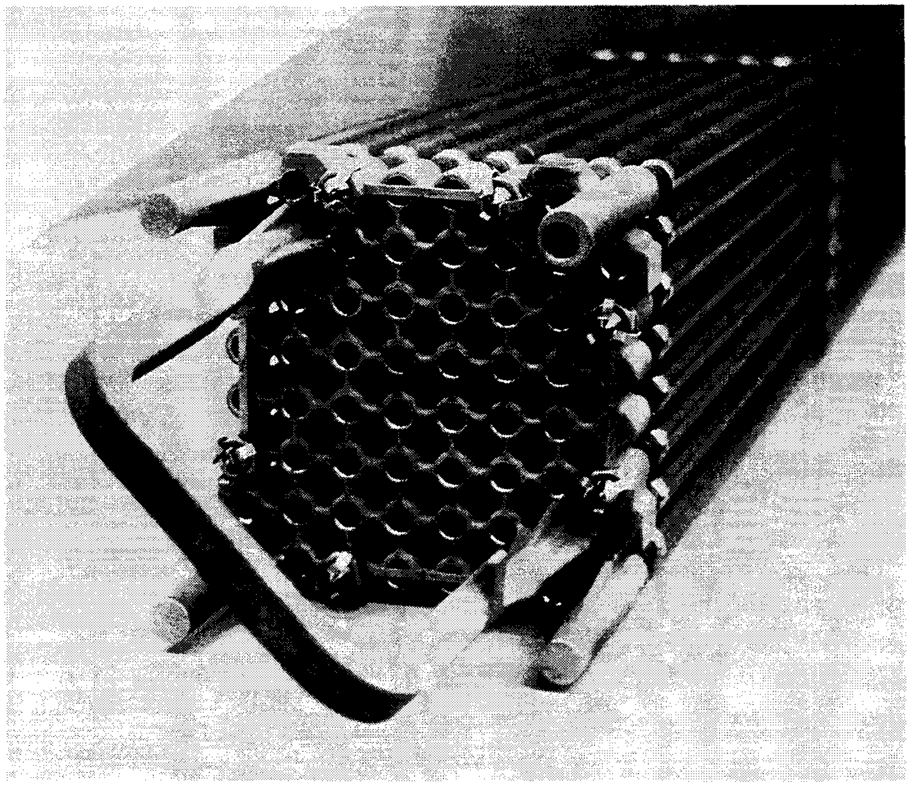


Figure 3-5. Upper Tie Plate

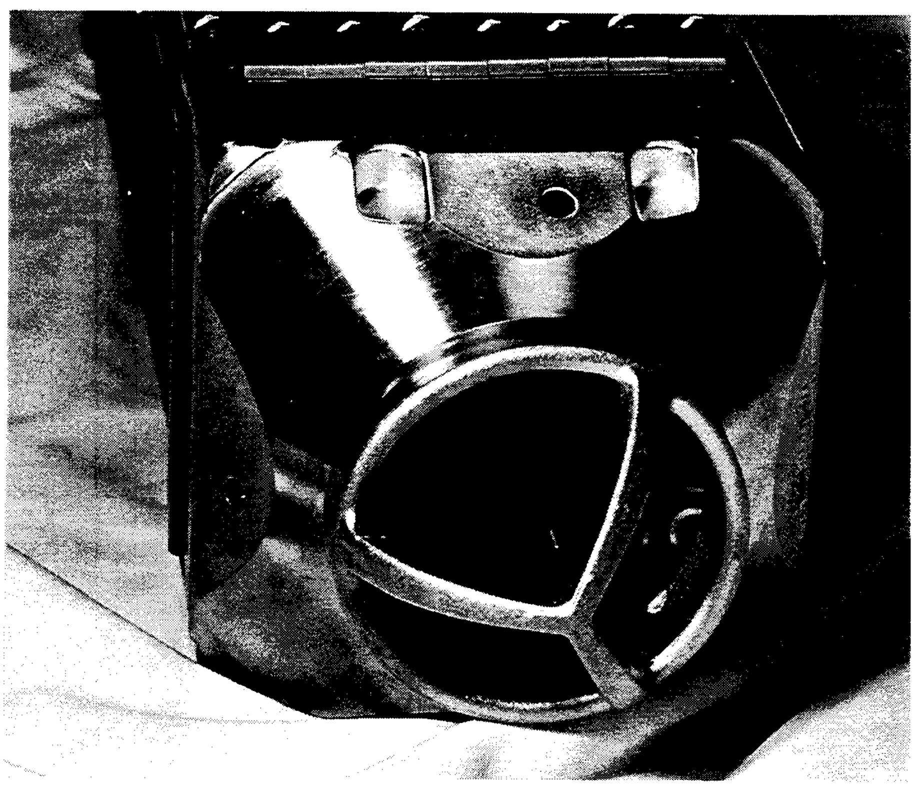


Figure 3-6. Lower Tie Plate (Nosepiece)

plugs which screw into the lower tie plate casting and extend through the upper tie plate casting. A stainless steel hexagonal nut and locking tab, shown in Figure 3-5, is installed on the upper end plug to hold the assembly together. These tie rods support the weight of the assembly only during fuel handling operations when the assembly hangs by the handle; during operation, the fuel rods are supported by the lower tie plate. Two rods in the interior foursome within each bundle (diagonally opposite the control blade) are water rods, i.e., tubes of Zr-2 cladding without UO_2 fuel. Small holes are located at both the lower and upper ends allowing water to be driven through the rod, thus introducing moderating material within the bundle interior. One water rod also serves as the spacer-positioning rod being mechanically locked to each of the seven spacer assemblies (see Figure 3-7), thereby fixing the axial position of each spacer. The fuel rod spacers are equipped with Inconel-X springs to maintain rod-to-rod spacing. The remaining 54 fuel rods in a bundle are standard rods having a single tube of fuel pellets the same length as the tie rods. The end plugs of both the spacer-capture rod and the standard rods have pins which fit into anchor holes in the tie plates. An Inconel-X expansion spring located over the top end plug pin of each fuel rod keeps the fuel rods seated in the lower tie plate while allowing them to expand axially by sliding within the holes in the upper tie plate to accommodate differential axial thermal expansion. These expansion springs are shown on the top of each fuel rod in Figure 3-5.

The initial core has an average enrichment ranging from approximately 1.7 wt % U-235 to approximately 2.0 wt % U-235 depending on initial cycle requirements. Individual fuel bundle enrichments in the initial core are of three or four different average enrichments, ranging from that of natural uranium, 0.711 wt % U-235, to a maximum of approximately 2.2 wt % U-235. The design of the initial core achieves an optimum balance of fuel economy, operating margins and ease of transition to equilibrium cycle refueling. The reload fuel has an average enrichment in the range of 2.6 to 3.05 wt % of U-235.

Different U-235 enrichments are used in fuel bundles to reduce local power peaking. Low enrichment uranium rods are used in corner rods and in the rods nearer the water gaps; higher enrichment uranium is used in the central part of the fuel bundle. Selected rods in each bundle are blended with gadolinium burnable poison.* The fuel rods are designed with characteristic mechanical end fittings, one for each enrichment. End fittings are designed so that it is not mechanically possible to complete assembly of a fuel bundle with any high enrichment rods in positions specifed to receive a low enrichment. A complete fuel bundle is shown in Figure 3-8.

* Poison: a material that absorbs neutrons unproductively and hence removes them from the fission chain reaction in a reactor, thereby decreasing its reactivity.

A recent innovation in BWR core design and operating strategy known as the Control Cell Core (CCC) has been developed by General Electric and is being demonstrated in operating BWR's. With successful completion of the demonstration projects, the CCC design will be available in future initial BWR cores and in reload cycles. A discussion of the CCC design is included as an appendix to this Section.

Bundle Features

The design has two important features:

- The bundle design places minimum external forces on a fuel rod; each fuel rod is free to expand in the axial direction.
- The unique structural design permits the removal and replacement, if required, of individual fuel rods.

Fuel Channel

A fuel channel encloses the fuel bundle; the combination of a fuel bundle and a fuel channel is called a fuel assembly (see Figure 3-9).

The channel is a square shaped tube fabricated from Zircaloy 4; its outer dimensions are 5.455 inches (138.6mm) by 5.455 inches (138.6mm) by 166.9 inches (4.2m) long. The reusable channel makes a sliding seal fit on the lower tie plate surface. It is attached to the upper tie plate by the channel fastener assembly (consisting of a spring and a guard), and a capscrew secured by a lock-washer. Spacer buttons are located on two sides of the channel to properly space four assemblies within a core cell. The fuel channel is shown in Figure 3-10. The fuel channels direct the core coolant flow through each fuel bundle and also serve to guide the control rod assemblies.

The use of the individual fuel channel greatly increases operating flexibility because the fuel bundle can be separately orificed and thus the reload fuel design can be changed to meet the newest requirements and technology. The channels also permit fast in-core sampling of the bundles to locate possible fuel leakers.

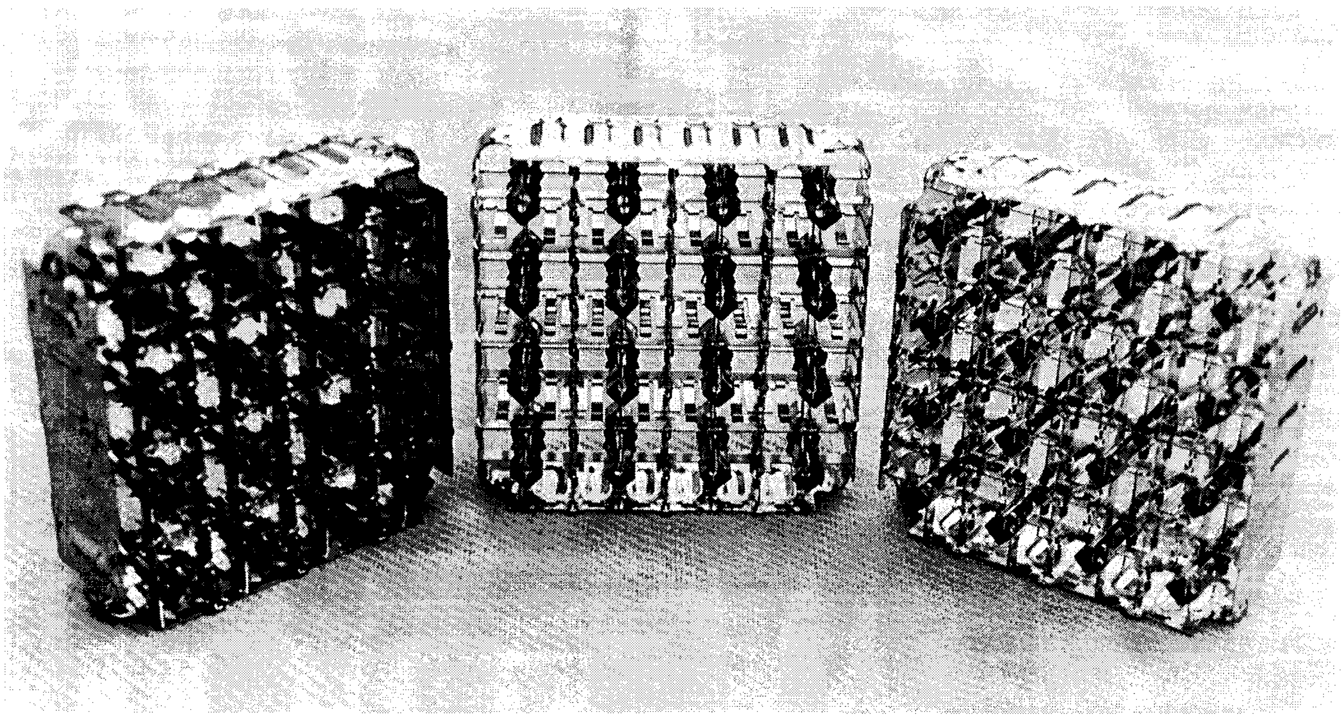


Figure 3-7. Typical Fuel Rod Spacers

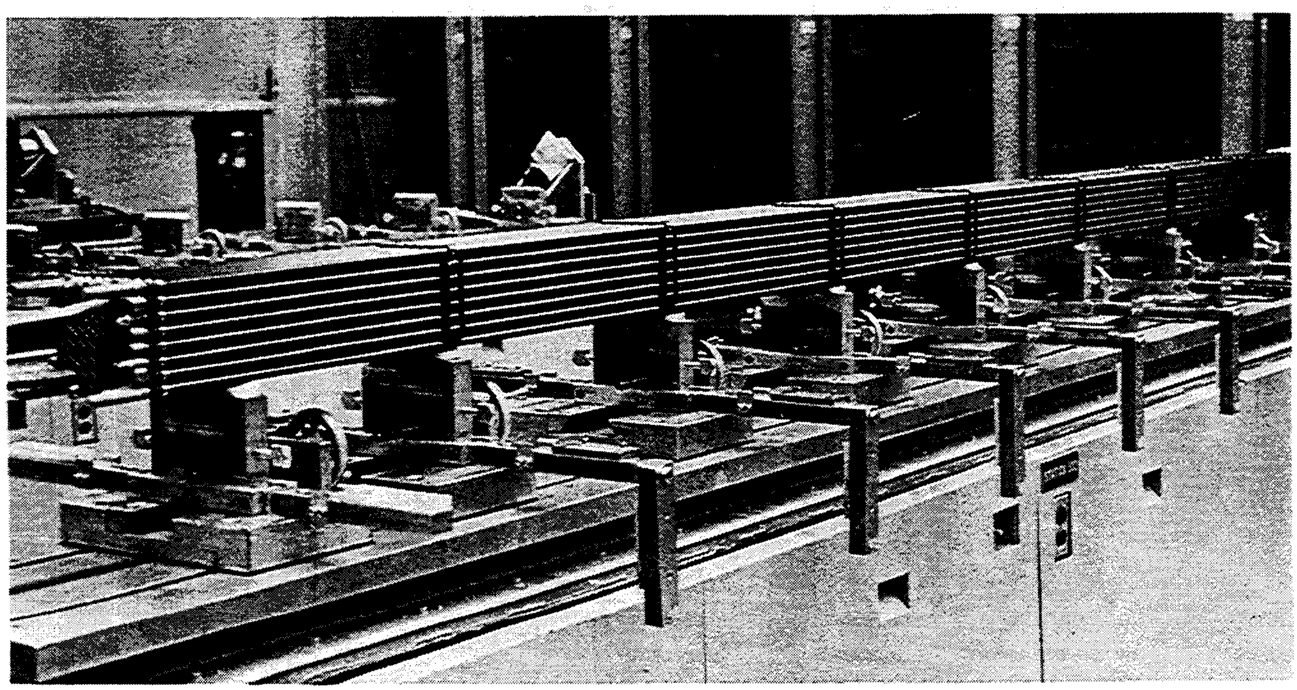


Figure 3-8. Completed Fuel Bundle

Initial Core

The initial core uses gadolinia in some fuel rods for temporary reactivity control.This gadolinia is loaded with slightly higher concentrations in the lower portion of the fuel rods (i.e., below the core midplane) in order to lower the axial power peaking factor. The magnitudes of gadolinia concentrations will be adjusted according to the enrichment requirements of initial core fuel.

Reload Fuel

Reload fuel also contains gadolinia. The distribution of gadolinia within the rods is similar to the gadolinia rods in the initial core.

For all fuel types, the gadolinia concentrations are established to result in nearly complete depletion within a single operating fuel cycle.

Neutron Sources

Several antimony-beryllium startup sources are located within the core. They are positioned vertically in the reactor by "fit-up" in a slot (or pin) in the upper grid and a hole in the lower core support plate (see Figure 3-11). The compression of a spring at the top of the housing exerts a column-type loading on the source. Though anchored firmly in place, the sources can easily be removed, but they need not be disturbed during refueling.

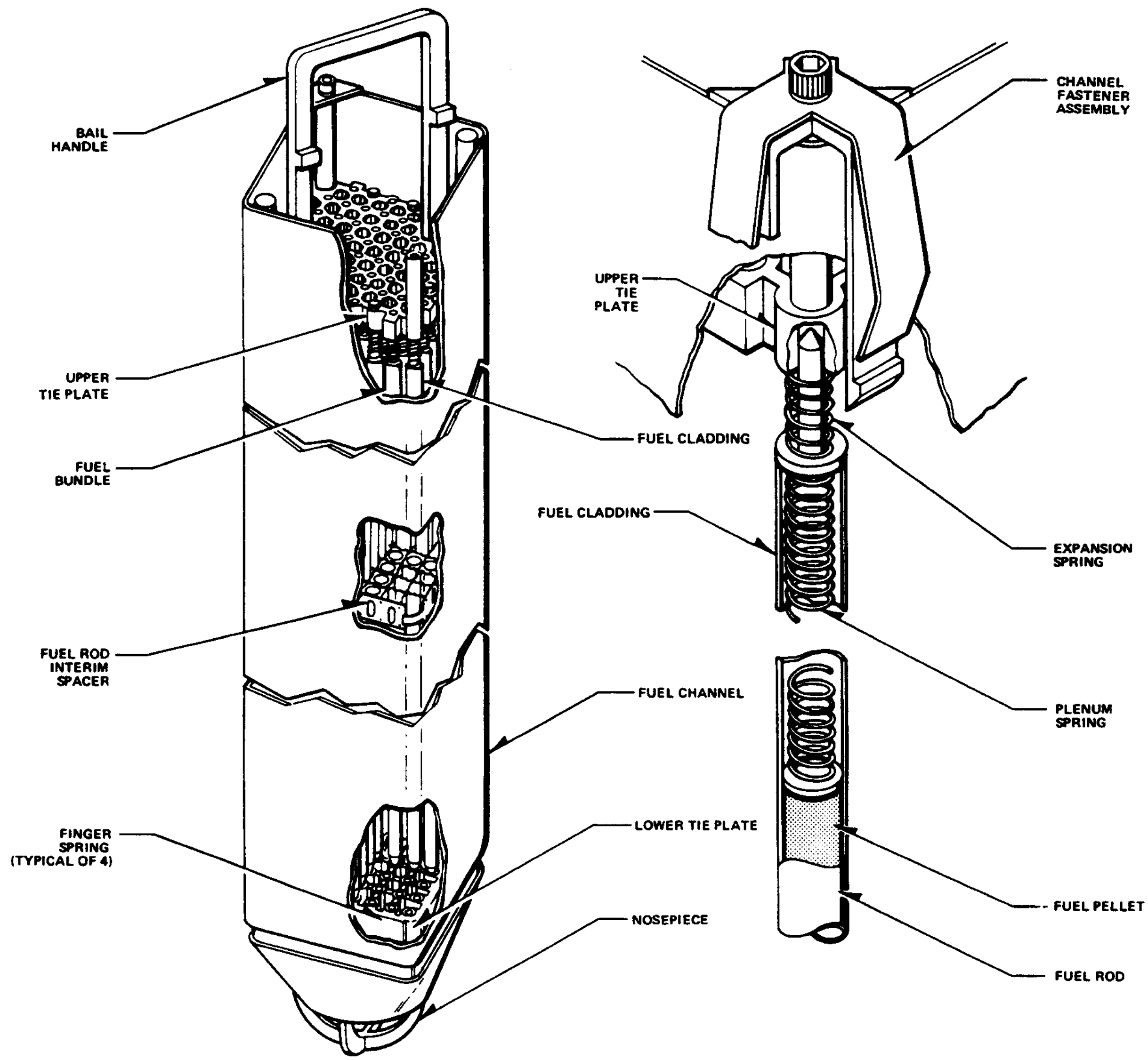


Figure 3-9. Fuel Assembly

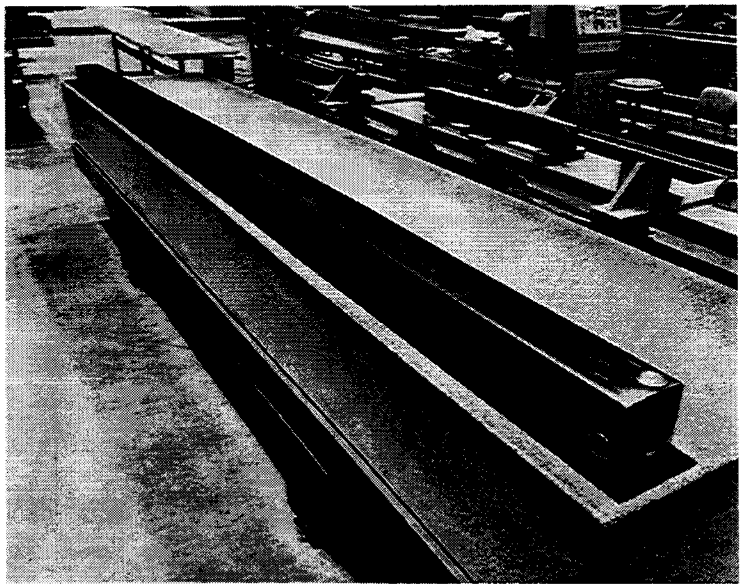


Figure 3-10. Fuel Channel

The active portion of each source consists of a beryllium sleeve enclosing two antimony-gamma sources. The resulting neutron emission strength is sufficient to provide indication on the source range neutron detectors for all reactivity conditions equivalent to the condition of all rods inserted prior to initial operation.

The active source material is entirely enclosed in a stainless steel cladding with an outside diameter of approximately 0.7 inch (17.8mm). The source is cooled by natural circulation of the core leakage flow in the annulus between the beryllium sleeve and the antimony-gamma sources.

The current neutron source mechanical design and the design analysis methods have been verified in the various General Electric operating BWR's.

CORE DESIGN

The reactor core is designed to operate at rated power with sufficient design margin to accommodate changes in reactor operations and reactor transients without damage to the core. In order to accomplish this objective the core is designed, under the most limiting operating conditions and at 100% of rated power, to meet the following bases:

- The maximum linear heat generation rate, in any part of the core, is less than 13.4 kW/ft (44 kW/m).
- Less than 0.1% of the core experiences transition boiling during the worst expected transient.

A later part of this section, "Margin Between Operating Limits and Damage Limits," discusses these two design bases and their associated margins.

Power Distribution

The design power distribution is divided for convenience into several components. The relative assembly power peaking factor is the maximum fuel assembly average power divided by the reactor core average assembly power. The axial power peaking factor is the maximum heat flux of a fuel assembly divided by the average heat flux in that assembly. The local power peaking factor is the maximum fuel rod heat flux at a horizontal plane in an assembly divided by the average fuel rod heat flux at that plane. Peaking factors vary throughout an operating cycle, even at steady-state full power operation, since they are affected by withdrawal of control rods to compensate for fuel burnup.

Representative maximum peaking factors for a typical core are —

• Relative assembly power	1.40
• Axial peaking	1.40
• Local peaking	1.13
• Total peaking	2.22

Axial Distribution — Because of the presence of steam voids in the upper part of the core, there is a natural characteristic for a BWR to have the axial power peak in the lower part of the core. During the early part of an operating cycle, bottom-entry control rods permit a reduction of this axial peaking by locating a larger fraction of the control rods in the lower part of the core. At the end of an operating cycle, the higher accumulated exposure and greater depletion of the fuel in the lower part of the core reduces the axial peaking.The operating procedure is to locate control rods so that the reactor operates with approximately the same axial power shape throughout an operating cycle.

Relative Assembly Power Distribution — The maximum-to-average fuel bundle peaking or radial distribution is reduced in a boiling water reactor core because of greater steam voids in the center bundles of the core. A control rod operating procedure is also used to maintain approximately the same radial power shape throughout an operating cycle.

Local Power Distribution — The local power distribution is reduced by the use of several uranium enrichments in a fuel bundle. Lower uranium enrichments are located near the water gaps, and higher enrichments are located in the center of a fuel bundle.

Core Thermal Hydraulics

Central UO_2 Temperature — The maximum UO_2 temperature will occur in new fuel operating at the maximum linear heat generation rate of 13.4 kW/ft (44 kW/m). Based on published conductivity data, the maximum temperature is approximately 3400F (1871C).

Core Orificing — Control of core flow distribution among the fuel assemblies is accomplished by fixed orifices. These orifices are located in the fuel support pieces and are not affected by fuel assembly removal and replacement.

The core is divided into two orifice zones. The outer zone of fuel assemblies, located near the core periphery, has more restrictive orifices than the inner zone. Thus, flow to the higher power fuel assemblies is increased. The orificing of all fuel assemblies increases the flow stability margin.

Thermal Hydraulic Limits — There are three types of boiling heat transfer to be considered in defining thermal limits: nucleate boiling, transition boiling and film boiling. Nucleate boiling is the extremely efficient mode of heat transfer in which the BWR is designed to

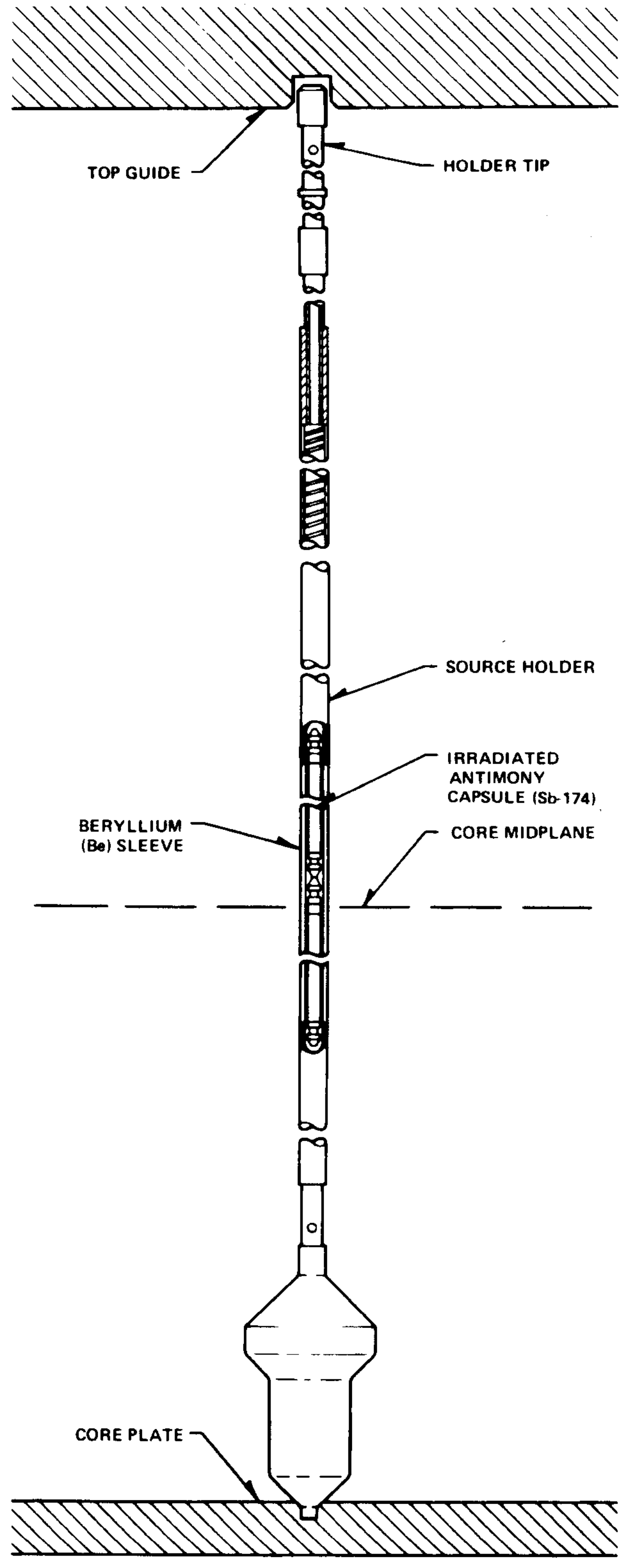


Figure 3-11. Neutron Source Schematic

operate. Transition boiling is manifested by an unstable fuel cladding surface temperature which rises suddenly as steam blanketing of the heat transfer occurs, then drops to the nucleate boiling temperature as the steam blanket is swept away by the coolant flow, then rises again. At still higher bundle powers, film boiling occurs which results in higher fuel cladding temperatures. The cladding temperature in film boiling, and possibly the temperature peaks in transition boiling, may reach values which could cause weakening of the cladding and accelerated corrosion. Overheating is conservatively defined as the onset of the transition from nucleate boiling. The relationship between cladding surface temperature and nucleate, transition and film boiling modes of heat transfer is shown in Figure 3-12. The core and fuel design basis has been defined, accommodating uncertainties, such that margin is maintained between the most limiting operating condition and the transition boiling condition at all times in core life.

The onset of transition boiling is predicted by a correlation which was developed as part of the General Electric Thermal Analysis Basis (GETAB). A major feature of the development work which produced data for this correlation is the ATLAS Experimental Heat Transfer Facility, the largest and most versatile facility of its kind in the world. ATLAS permits full-scale thermal hydraulic testing of BWR fuel assemblies under conditions of power, flow, pressure, enthalpy, and power peaking which duplicate the actual reactor environment. This facility is also capable of simulating transient and emergency core cooling conditions. Data from this prototypical testing form the extensive data base on which the correlation is founded.

The GETAB correlation predicts the steam quality (critical quality) at which boiling transition occurs as a function of the distance above the boiling boundary (boiling length) for any given mass flow rate, power level, pressure, local steam quality, bundle flow geometry, and local peaking pattern. The figure of merit used for reactor design and operation is the critical power ratio (CPR). The use of CPR is both convenient for application and descriptive of the relation between normal operating conditions and conditions which produce a boiling transition. As previously mentioned, the critical quality is predicted by the GETAB correlation. The "critical power" is defined as that bundle power which would produce the critical quality. The critical power ratio is then defined as the ratio of the critical power to the operating bundle power, at the reactor condition of interest. These relationships are illustrated in Figure 3-13. Shown in this figure are the correlation line and steam quality distributions over the length of the bundle for three bundle powers. P_1, P_2, and $P_{critical}$. As seen in the figure, the critical power ($P_{critical}$) is that bundle power which produces a steam quality versus boiling length curve that is tangent to the correlation line. The CPR must be held above a prescribed value for all fuel assemblies in the reactor core. The minimum CPR for the most limiting fuel assembly in the core is defined as the minimum critical power ratio (MCPR). Reactor operating limits are thus stated in terms of MCPR.

The statistical basis for the selection of design and operational limits explicitly recognizes that there is a possibility, however small, that some combination of a transient and various uncertainties and tolerances may cause transition boiling to exist locally for some period of time. In applying this statistical basis, the MCPR limits are derived from the following design basis requirement:

Transients caused by single operator error or single equipment malfunction shall be limited such that, considering uncertainties in monitoring the core operating state, more than 99.9% of the fuel rods are expected to avoid boiling transition.

Determination of actual GETAB operating limits is made by application of the above design requirement to specific reactors using a full core statistical analysis. In addition to the uncertainty of the GETAB correlation itself, explicit uncertainties for the following parameters are input to the statistical analysis: feedwater flow, feedwater temperature, reactor pressure, core flow, core inlet temperature, friction factor multipliers, nuclear instrument (TIP) readings, and local power distributions. These uncertainties are combined in a Monte Carlo analysis which calculates the number of boiling transition events which would occur in the core under the input conditions.

The result of this analysis is an MCPR limit for the worst transient (design basis) condition. The steady-state operating limit is derived from this by applying an additional MCPR margin for the worst expected transient.

Thermal-Hydraulic Analysis

A computer program is used to analyze the thermal and hydraulic characteristics of the reactor core as a whole. The geometric, hydraulic, and thermal characteristics of the core design are represented, including number of fuel assemblies in each orifice zone of the

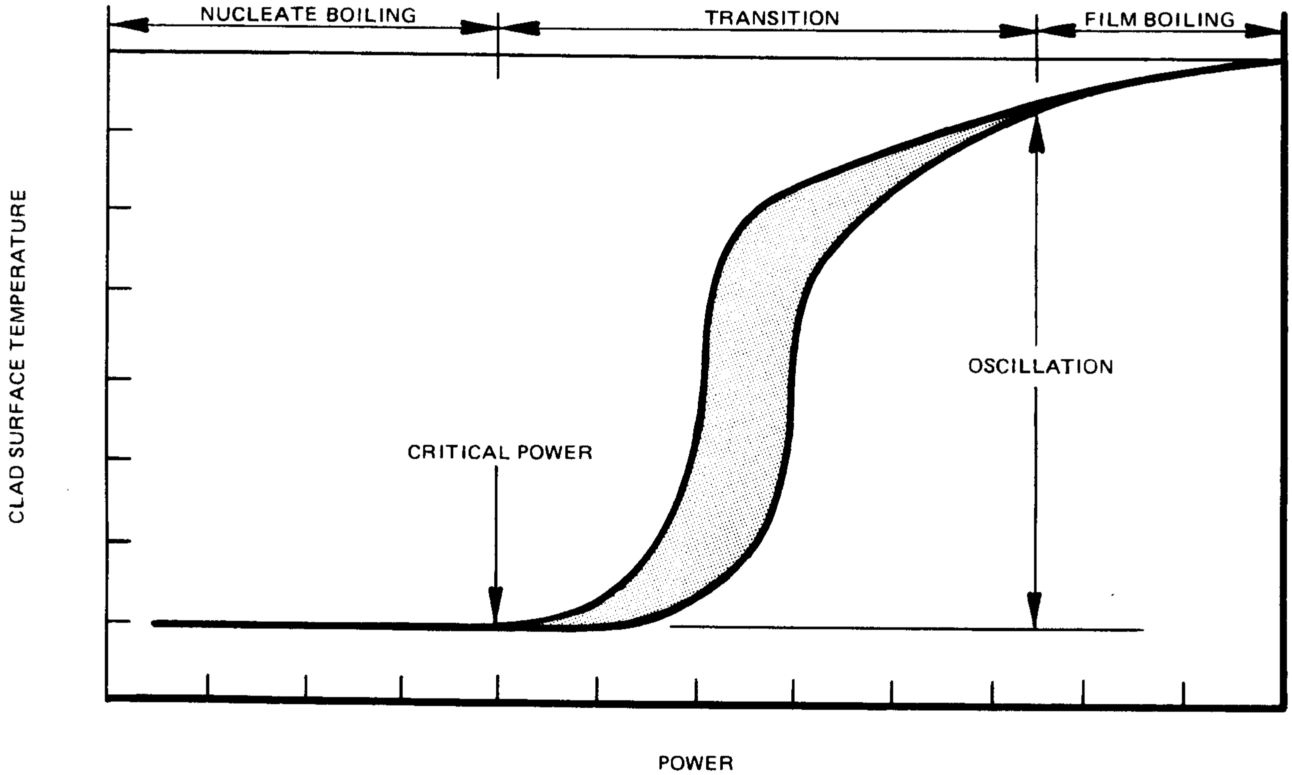


Figure 3-12. Thermal Hydraulic Boiling Modes

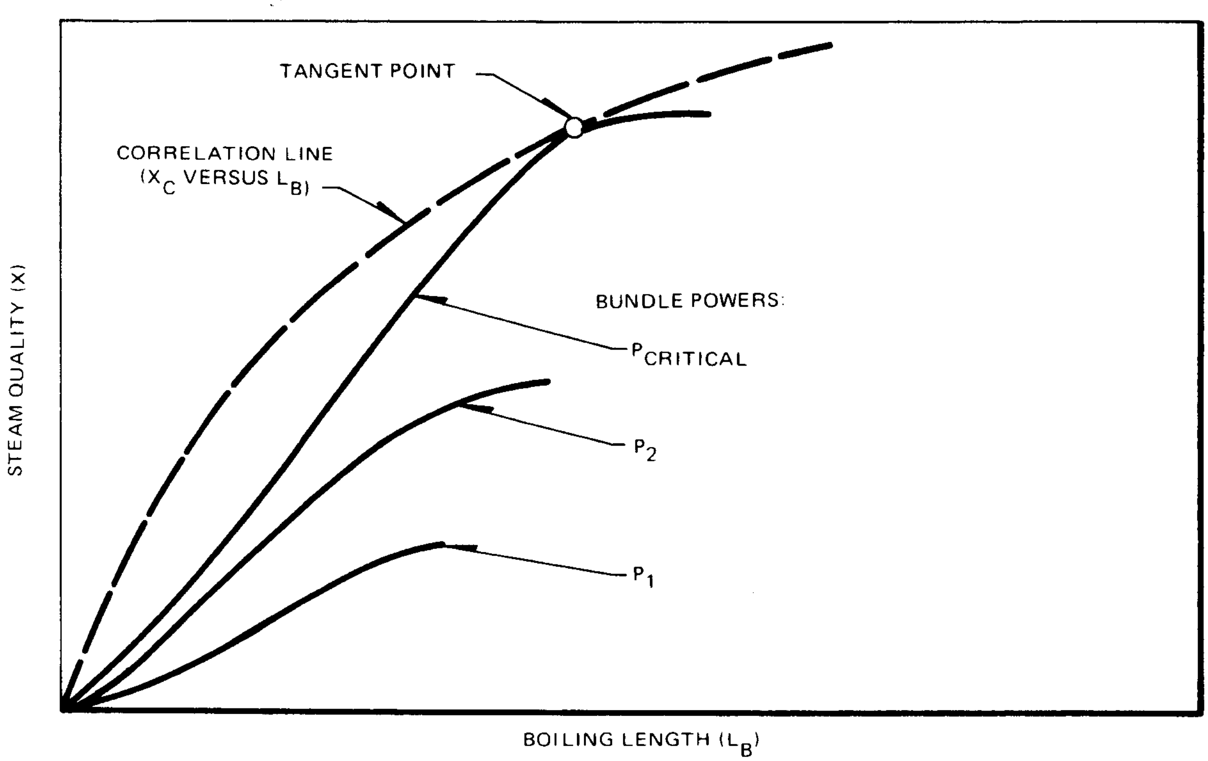


Figure 3-13. Critical Power

reactor core, fuel assembly dimensions, friction factors and flow restrictions, and the flow characteristics of the fuel orifices, inlet plenum region of the reactor, along with bypass and leakage flow paths around the fuel assembly channels. Individual cases are analyzed by providing reactor power, flow, inlet enthalpy and appropriate power distribution factors as input to the above computer program. The output of the program includes the calculated flow distribution among the several channel types and a detailed analysis of the heat fluxes, steam quality, void fraction, and MCPR at as many as 24 axial nodes for the average and peak power fuel assemblies in each orifice zone.

Comparisons of the analytical models used with fuel assembly design details such as fuel-rod-to-fuel-rod and fuel-rod-to-fuel-assembly-channel clearances and spacer configurations have been made to ensure that the computer programs adequately represent the actual core and fuel design, and that design correlations are applicable.

In addition, fuel thermal design calculations, including calculation of fuel rod temperature, UO_2 pellet thermal expansion characteristics, and rate of UO_2 swelling due to irradiation, have been performed. Thermal effects of irradiation, including reduction in local power peaking factor due to U-235 depletion, buildup of plutonium near the surface of the pellet, and effect of gap width and gas composition on gap conductance, have been considered in confirming that the thermal-hydraulic performance objectives will be met.

Core Nuclear Characteristics

Basis of Nuclear Characteristics — The nuclear calculations are based on nuclear data selected from the best current sources of information throughout the nuclear industry and on calculational computer codes developed by General Electric for the BWR. Of even greater importance is the fact that these data and computer codes are continually checked against experimental data obtained from critical facilities, startup tests of BWR's going into commercial operation, and against the long-term operating data of BWR's such as Dresden 1, Garigliano, KRB, and other more recently operating BWR plants. Predicted isotopic compositions are checked by comparisons with measured isotopic data of discharged BWR fuel.

Reactivity Coefficients — In a boiling water reactor, two reactivity coefficients are of primary importance: the fuel Doppler coefficient and the moderator density reactivity coefficient. The moderator density reactivity coefficient may be broken into two components: that due to temperature and that due to steam voids.

Fuel Doppler Reactivity Coefficient — As in all light water moderated and low enrichment reactors, the fuel Doppler reactivity coefficient is negative and prompt in its effect, opposing reactor power transients. When reactor power increases, the UO_2 temperature increases with minimum time delay and results in higher neutron absorption by resonance capture in the U-238.

Moderator Density Reactivity Coefficient — During normal plant operations, the steam void component of the moderator density reactivity coefficients is of prime importance. The steam void component is large and negative at all power levels. At full rated power, the steam voids are equivalent to approximately 3% reactivity.

The fuel assembly design is such that the moderator density reactivity coefficient of the water within the fuel channel is negative for all conditions of operation. The in-channel moderator coefficient is smallest at the cold, zero power condition.

The large and negative moderator density coefficient at operating power levels is due to the steam void effect. This steam void effect results in the following operating advantages:

Xenon Override Capability — As the steam void reactivity effect is large compared with xenon reactivity, the BWR core has the excellent capability of overriding the xenon effect, thereby increasing power after a power decrease.

Xenon Stability — The steam void reactivity is the primary factor in providing the high xenon stability characteristic. See "Reactor Stability" later in this section.

Load Changing by Flow Control — Since the fuel Doppler reactivity opposes a change in load, the void effect must be and is larger than the fuel Doppler effect in order to provide load changing capability by flow (or moderator density) control.

Thermal-Hydraulic Stability — The negative void effect is an important contributor to reactor thermal-hydraulic stability.

Reactivity Control

The movable boron-carbide control rods are sufficient to provide reactivity control from the cold shutdown condition to the full load condition. Supplementary reactivity control in the form of solid burnable poison is used only to provide reactivity compensation for fuel burnup or depletion effects.

The movable control rod system is capable of maintaining the reactor in a subcritical condition when the reactor is at ambient temperature (cold), zero power, zero xenon, and with the strongest control rod fully withdrawn from the core. In order to provide greater assurance that this condition can be met in the operating reactor, the core design is based on calculating a reactivity less than 0.99, or a 1% margin on the "stuck rod" condition.

Supplementary solid burnable poisons are used to assist in providing reactivity compensation for fuel burnup. For all operating cycles, the supplementary control is provided by gadolinia mixed into a portion of the UO_2 reload fuel rods.

Margin Between Operating Limits and Damage Limits

Two mechanisms which could result in fuel damage (i.e., perforation of the cladding) are:

- Severe overheating of the fuel cladding.
- Fracture of the fuel cladding due to excessive strain resulting from UO_2 thermal expansion.

Although significant weakening of the fuel cladding due to overheating is not expected to occur until well into the film boiling region, fuel damage is conservatively defined as the onset of transition boiling. This, by definition, corresponds to MCPR = 1.00. In addition to this limit a statistical margin of approximately 6% is made to allow for the various uncertainties in predicting and detecting the actual boiling state. Thus, during the worst expected transient, the MCPR is not permitted to go below a value of approximately 1.06. An additional margin for the effects of the worst transient produces the normal operating limit. A typical value for this operating limit is MCPR = 1.23. During full power operation, the fuel will typically operate with an MCPR greater than 1.30. The difference between the actual operating value of MCPR and the operating limit is termed the operating margin.

A value of 1% plastic strain of Zircaloy cladding is conservatively defined as the limit below which fuel damage is not expected to occur. Available data indicate that the threshold for damage in irradiated Zircaloy cladding is in excess of this value. The linear heat generation rate required to cause this amount of cladding strain is approximately 25 kW/ft (82 kW/m). The linear heat generation rate for the worst expected transient is approximately 16 kW/ft (52.5 kW/m). During normal full power operation the maximum linear heat generation rate will not exceed 13.4 kW/ft (44 kW/m). The fuel damage limit of 1% plastic strain is based on tensile tests of irradiated cladding from both the Vallecitos Boiling Water Reactor (VBWR) and Dresden 1 reactors and based on fuel tests at linear heat rate conditions of more than 25 kW/ft (82 kW/m).

Strain Localization

It was determined for 7×7 BWR fuel that strain localization due to pellet-to-cladding interaction at pellet interfaces (ridging) and pellet cracks can cause a small but statistically significant number of fuel rod perforations during normal reactor operation. The following fuel design improvements have been made for 8×8 BWR/6 fuel to reduce pellet-to-cladding localized strain:

1. The fuel pellet length-to-diameter ratio is decreased from 2:1 to 1:1, which reduces ridging.
2. The fuel pellet is chamfered, which reduces ridging.
3. The maximum linear heat generation is decreased from 18.5 kW/ft (60.7 kW/m) to 13.4 kW/ft (44 kW/m), which reduces thermal distortion and ridging.
4. The cladding heat treatment procedure is improved to reduce the variability of the cladding ductility.

Reactor Stability

The large fuel time constant and inherent negative moderator feedback are major contributors to the stability of the boiling water reactor. The Doppler reactivity feedback appears simultaneously with a change in fuel temperature and opposes the power change that caused it, while heat conduction to water and the subsequent formation of steam voids must await transfer of heat through the fuel material. Since the Doppler reactivity opposes load changes, it is desirable to maintain a large ratio of moderator-to-Doppler coefficient for optimum load following capability. The BWR takes advantage of its inherently large moderator-to-Doppler coefficient ratio by permitting a variation of coolant flow for load following.

Xenon instability is an oscillatory phenomenon of xenon concentration throughout the reactor that is theoretically possible in any type of reactor. If such a condition should occur, it can restrict load following performance, cause increased local power peaking in the core and possibly reduce the fuel economics of the core. The BWR as designed by General Electric has characteristics which provide a large margin of damping to such oscillations. This is primarily brought about by the high negative power coefficient characteristic of the core. In addition, the use of in-core ion chambers for local monitoring of core conditions and for local reactivity adjustment brought about by the control rods and local steam void control provide complete knowledge of core conditions and adequate control capability. Since xenon oscillations are local phenomena within the core, they are not evident when looking at core averaged values and unless in-core instrumentation is provided, the presence of such oscillations may not be known until they have caused power peaking with possible core damage.

If the magnitude of the power coefficient of reactivity becomes too small, spatial xenon oscillations will occur and restrict reactor load following and performance. Even in the stable region, it is important to have well-damped power distributions and to select reactor load following variables which do not tend to encourage spatial xenon oscillations. Studies have shown that underdamped, unacceptable behavior would occur for large boiling water reactors with a power coefficient less negative than $-0.014\ \Delta k$ per unit fractional change in power. Current boiling water reactor designs result in power coefficients well beyond the range of instability of xenon. This advantage of the boiling water reactor is of major importance for large, loosely coupled nuclear cores. Flow control further aids spatial xenon stability by providing a power shape which remains relatively constant at varying reactor power levels.

The water-to-fuel volume ratio is determined from consideration of the reactivity coefficient for safe and stable operation. This ratio is selected to provide cold lattice coefficients which preclude detrimental startup transients. In parallel to the allowance of considerable margin in design for good load following and spatial xenon stability, the water-to-fuel volume ratio selected is close to the optimum for minimum fuel cycle costs.

With proper design, the dynamics of a boiling water reactor allow satisfactory performance with respect to stability. The fact that such design stability exists has been demonstrated through extensive tests on Dresden 1, Garigliano, KRB, and other General Electric designed reactors, and on the experimental boiling water reactor. For example, the Dresden 1 reactor has been operated for several years on full power during the day and half power at night load schedule, which produces maximum xenon concentration gradients; no xenon instabilities have been observed.

REACTOR REACTIVITY CONTROL

Control Rods

The control rod design, using boron carbide (B_4C) compacted in stainless steel tubes, was introduced into service in the Dresden 1 boiling water power reactor in April 1961. Since that date, this design has been the standard reference control element in all General Electric boiling water reactors and has replaced the 2% boron-steel rods previously used. Over the years, B_4C control rods have been produced routinely in quantity, by tested manufacturing procedures. During the years since the first rods were placed in service, they have demonstrated excellent mechanical and nuclear performance.

The control rods perform dual functions of power distribution shaping and reactivity control. Power distribution in the core is controlled during operation of the reactor by manipulation of selected patterns of rods. The rods, which enter from the bottom of the near-cylindrical reactor, are positioned in such a manner to counterbalance steam voids in the top of the core and effect significant power flattening. These groups of control elements, used for power flattening, experience a somewhat higher duty cycle and neutron exposure than the other rods.

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

Description of Rods

The cruciform control rods contain 72 stainless steel tubes (18 tubes in each wing of the cruciform) filled with boron carbide powder compacted to approxi-

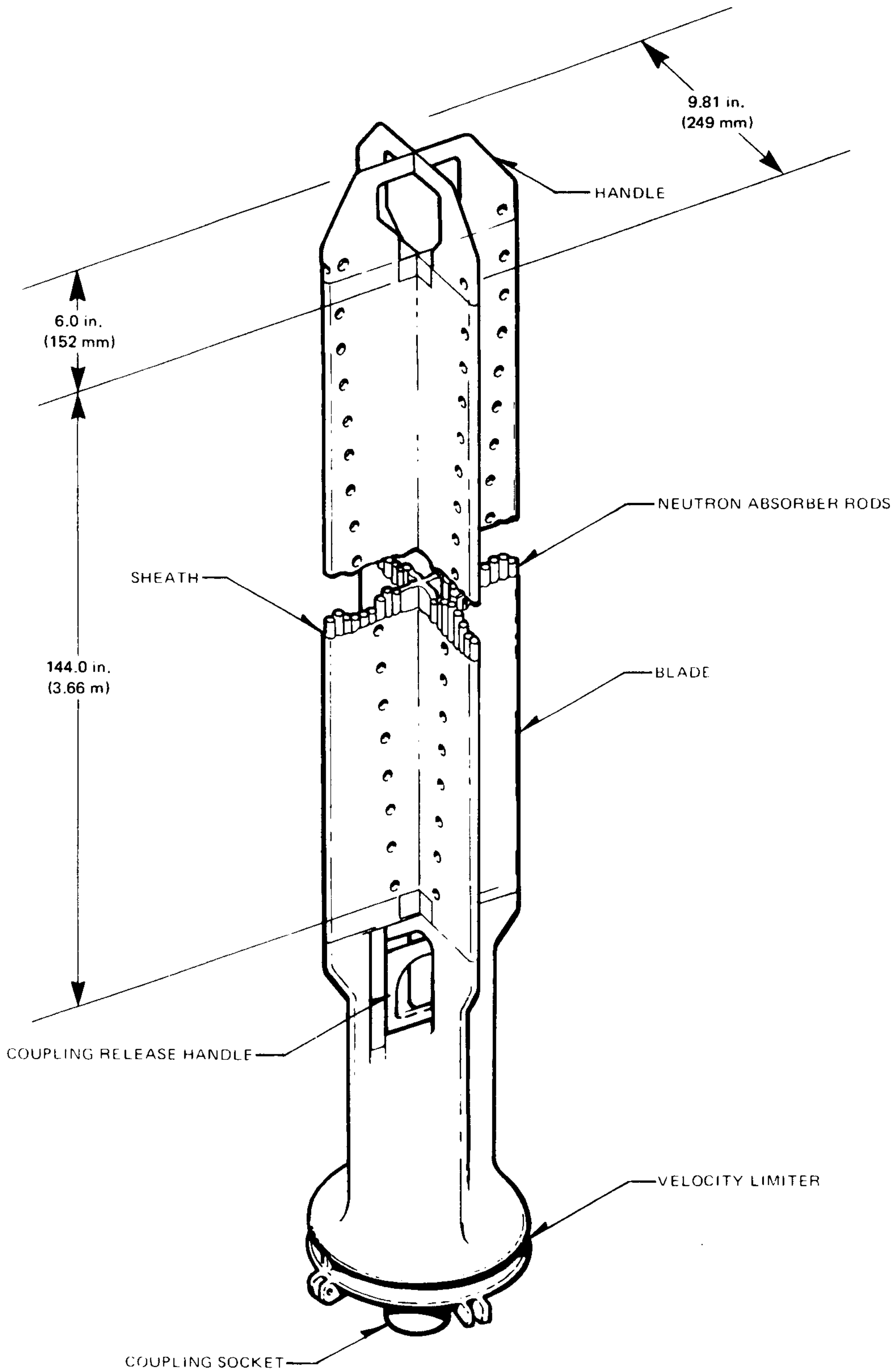


Figure 3-14. Control Rod

mately 75% of theoretical density. The tubes are seal welded with end plugs on either end. Stainless steel balls are used to separate the tubes into individual 18-inch (0.46m) longitudinal compartments. The stainless steel balls are held in position by a slight crimp in the tube. The individual tubes, 3/16 inch (4.8mm) in diameter, act as pressure vessels to contain the helium gas released by the boron-neutron capture reaction.

The tubes are held in cruciform array by a stainless steel sheath extending the full length of the tubes. A top casting and handle, shown in Figure 3-14, aligns the tubes and provides structural rigidity at the top of the control rod. Rollers, housed by the top casting, provide guidance for control rod insertion and withdrawal. A bottom casting is also used to provide structural rigidity and contains positioning rollers and a parachute shaped velocity limiter. The castings are welded into a single structure by means of a small cruciform post located in the center of the control rod. The control rods have an active length of 144 inches (3.7m) of boron carbide, a span of 9.75 inches (0.25m), and an overall length of 173.75 inches (4.4m). The control rods can be positioned at 6-inch (152mm) steps and have an operations withdrawal and insertion speed of 3 inches (76.2mm) per second.

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

- Four holes in fuel assembly nosepiece (lower tie plate)
- The area between fuel channel and fuel assembly nosepiece
- The area between fuel assembly nosepiece tie and fuel support piece
- The area between fuel support piece and core plate
- The area between core plate and shroud
- Holes in the core plate for bypass flow control

Control Rod Nuclear Characteristics

The control rod system is designed so that adequate shutdown capability is available at all times. To permit a margin for credible reactivity changes, the control system has the capability to shut down and maintain the core continuously subcritical with the maximum worth control rod fully withdrawn. This capacity is experimentally demonstrated when reactivity alterations are made to the reactor core. The use of mechanical control permits periodic tests on the core reactivity during refueling. Control rods are withdrawn adjacent to an inserted fresh fuel assembly to verify subcriticality and predicted excess reactivity of the fuel.

The control rod insertion rates on scram are sufficient to protect the reactor against damage in all transients which are expected to occur during the life of the plant.

The control rods are used primarily for power distribution shaping and for shim control of long-term reactivity changes which occur as a result of fuel irradiation. The flow control function, which is used to follow rapid load changes, reduces requirements on speed of control rod response and thus improves plant safety. Every 2 to 3 months, the control rod patterns are altered to provide more uniform fuel and control rod burnup. In normal daily operation, little control rod movement is required for depletion of reactivity. The resulting low frequency of control rod changes reduces the possibility of operator error.

With the normal control rod patterns required to maintain an acceptable power distribution in the operating core, an average control rod will be worth about 0.005 Δk effective. The maximum worth of a rod in a typical power operation pattern will be about 0.01 Δk effective. The notch increment dimensions and spacing of the rods are set to limit the reactivity insertion to about 0.0003 $\Delta k/k$ for any notch increment of control withdrawn. Preplanned withdrawal patterns and procedural patterns and procedural controls are used to prevent abnormal configurations yielding excessively high rod worths.

The rod worth monitoring function is contained in the on-line digital computer. Its purpose is to determine that low worth scatter control rod patterns are maintained. The function is preventive in concept. It allows maximum freedom of control rod scheduling and prevents those rod patterns which result in excessive rod worth.

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

Supplementary Reactivity Control

The control requirements of the initial core are designed to be considerably in excess of the equilibrium core requirements because all of the fuel is fresh in the initial core. The initial core control requirements are met by use of the combined effects of the movable control rods and a supplemental burnable poison. The supplementary burnable poison is digadolinia trioxide (Gd_2O_3), called gadolinia, mixed with UO_2 in several fuel rods in each fuel bundle.

Only a few materials have nuclear cross sections that are suitable for burnable poisons. An ideal burnable poison must deplete completely in one operating cycle so that no poison residue exists to penalize initial U-235 enrichment requirements. It is also desirable that the positive reactivity from poison burnup match the almost linear decrease in fuel reactivity from fission product buildup and U-235 depletion. A self-shielded burnable poison consisting of Gd_2O_3 dispersed in a few selected fuel rods in each fuel assembly provides the desired characteristics. The Gd_2O_3 depletes as a cylinder with decreasing radius to provide a linear increase in reactivity. The concentration is selected so that the poison essentially depletes in the operating cycle. It is possible to improve power distributions by spatial distribution of the burnable poison.

FUEL MANAGEMENT

General Electric can supply recommendations on optimum reloading schemes during core life. Long-range loading schedules are furnished with the initial core and can be periodically revised as operating data become available. The flexibility of the boiling water reactor core design permits variation of the intervals between refueling through variation of the refueling batch size.

The first shutdown for refueling will occur approximately 1 to 2 years after commencement of power operation. Thereafter, the plant operating cycle time (time between core reloading) can be varied up to potentially 18 months. During transition from the initial core to the equilibrium core, the refueling batch size can be varied to maintain this equilibrium refueling interval.

The design refueling schedule is based on the specified annual load factor. Variations in the annual load factor can be accommodated by the following:

- Maintaining a constant reload fuel enrichment and varying the refueling batch size to maintain the equilibrium refueling interval. A large refueling batch size is required as the load factor is increased.
- Varying the refueling batch size and the reload fuel enrichment to maintain the equilibrium refueling interval and constant discharge exposure.
- Maintaining constant reload fuel enrichment and refueling batch size and allowing the refueling interval to vary up to 18 months.

For annual refueling at high capacity factors, approximately one-fourth of the fuel is discharged each cycle. For 18 month refueling cycles, approximately one-third of the fuel is discharged at the end of each cycle.

Section 3, Appendix

Control Cell Core Improved Design

by

S. R. Specker, L. E. Fennern, R. E. Brown, R. L. Crowther, and K. L. Stark

A new BWR core design and operating strategy known as the Control Cell Core (CCC) concept has been developed by GE and is being demonstrated in an operating BWR. This concept results from a coupled optimization of fuel bundle design, refueling patterns and control rod patterns. The primary objectives of the CCC concept are to simplify BWR operation, improve fuel reliability, increase operating thermal margins and improve BWR capacity factors. The need for periodic control rod pattern exchanges during an operating cycle is either eliminated or substantially reduced by the concept, with subsequent major impact on utility operations. The design concept has potential for backfit to all existing BWRs.

Description

The CCC concept is based on an operating strategy in which control rod movement to offset reactivity changes during power operations is limited to a fixed group of control rods. Each of these rods and its four surrounding fuel assemblies comprise a control cell. All other control rods are normally completely withdrawn from the core while operating at power. Low reactivity fuel assemblies are placed in the control cells so that control rod motion occurs adjacent only to low power fuel.

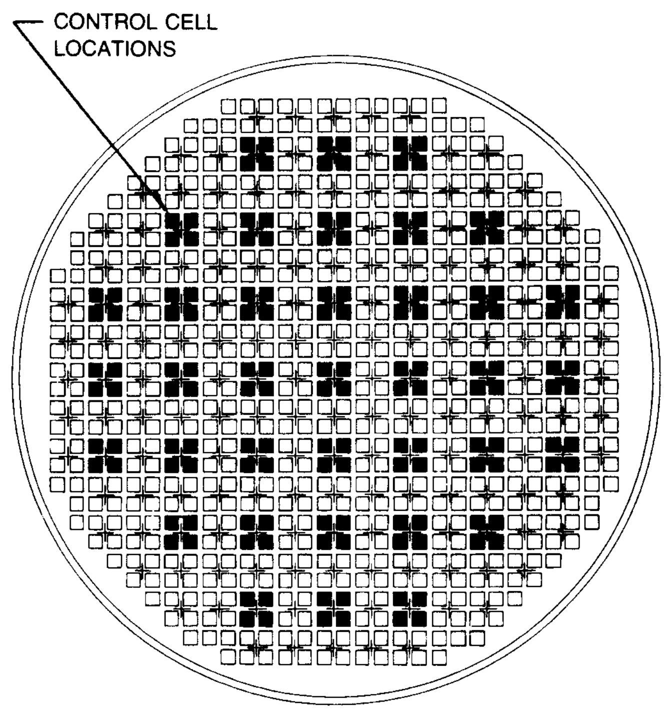


Figure 1. Typical Control Cell Locations for a Large BWR Core

The control cell locations for a typical large BWR are shown in Figure 1. For an initial cycle, low enrichment fuel assemblies are placed in the control cells. In the reload cycles, the fuel assemblies from the lowest reactivity batch residing in the core after the discharge batch has been selected are placed in the control cells. For a typical equilibrium cycle, these correspond to the highest burnup assemblies.

The cumulative effects of an inserted control rod on the local power distribution of a control cell fuel assembly are minimized by designing the fuel with a relatively low enrichment rod in the corner of the assembly nearest the control rod and by allowing a fuel assembly to reside in a control cell for only one cycle.

There are two principal differences between the design and operation of the CCC and a conventional BWR core. They are:

1) In a conventionally designed BWR core, fuel assemblies of differing reactivity are scatter loaded in the core so that surrounding any particular control rod location there is normally at least one assembly of relatively high reactivity. As a result, control rod motion occurs adjacent to relatively high power fuel in a conventional core.

   With the CCC design, low reactivity fuel assemblies are placed in the control cells so that control rod motion occurs adjacent only to relatively low power fuel.

2) In conventional BWR operation, four basic control rod patterns designated as A-1, A-2, B-1, and B-2 are used to develop operating control rod pattern sequences through each cycle. Alternating sets of control rods are designated for use as deep or as shallow rods for each of these basic patterns. During a typical annual operating cycle, about five control rod pattern exchanges are performed.

   With the CCC design, a fixed group of control rods located in the low power control cells is utilized for reactivity and power distribution control throughout the entire cycle. Conventional rod pattern exchanges are not required.

Performance Characteristics

Analytical studies and feedback from a current operating plant demonstration indicate that, relative to conventional cores, the CCC design has the potential to simplify operations and increase BWR capacity factors. To understand the underlying physical bases for these improvements, it is useful to examine in detail some of the important differences in the performance characteristics of conventional cores and CCC's.

Conventional Core Characteristics

To begin the discussion, it is useful to look at the behavior of a typical high reactivity, high power fuel assembly in a conventional core. Figure 2 illustrates the change in the nodal power distribution of such a high power fuel assembly as an adjacent inserted control rod is withdrawn. Note the large increase in power as the control rod tip is withdrawn past a specific axial node and the magnitude of the resulting peak nodal power. Note also that the maximum nodal power occurs when the control rod is partially inserted rather than when completely withdrawn. This behavior results from the presence of relatively low in-channel voids in a region of high neutron importance immediately off the tip of the control rod. Such a partially controlled fuel assembly is frequently the location of the peak nodal kW/ft in a conventional core.

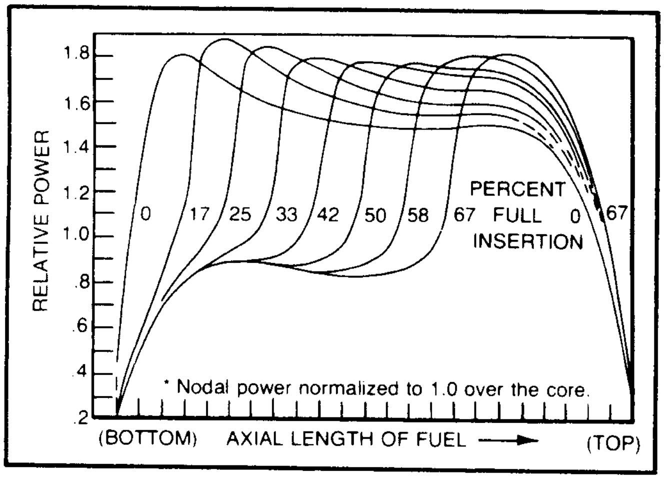


Figure 2. Power in Fresh Fuel Assembly of Conventional Design as Adjacent Control Rod is Withdrawn Toward Bottom*

Another characteristic of a conventional core which can be seen in Figure 2 is the underburning of the lower portion of the high power fuel assembly which occurs when a shallow control rod is inserted immediately adjacent to it. When the control rod is withdrawn, this underburning will cause an increase in the peak nodal power in the assembly relative to the uncontrolled peak shown in Figure 2. Thermal margins can be seriously eroded, particularly near the end of a cycle, by underburning the bottom of a conventional core. Careful control rod pattern selection and periodic rod pattern exchanges are required to obtain a good "bottom burn" in the high power assemblies of a conventional core.

Besides its effect on the nodal power and burnup, an inserted control rod also causes a localized distortion in the neutron flux. When inserted, the control rod depresses the neutron flux and hardens the spectrum in nearby fuel rods resulting in a reduced burnup rate and an increased Pu/U conversion ratio relative to the fuel rods located further away from the control rod.

When the control rod is withdrawn, these nearby rods, because of their relatively lower burnup and increased plutonium content, operate at a higher power than if they had previously been uncontrolled. The magnitude of this power increase is greatest in the corner rod and depends primarily on the burnup that an axial segment of a fuel assembly accumulated while previously controlled, and the burnup accumulated since becoming uncontrolled. This phenomenon is commonly referred to as a control blade history (CBH) effect on local power distribution. If the adjacent fuel assembly contains unburned gadolinia, the effect of CBH on local power distribution is further complicated by the reduced gadolinia burnup rate in the nearby gadolinia bearing rods. This can cause additional distortions in local and global power distributions.

Periodic exchanges between the basic A and B rod patterns mitigates the effects of control blade history on core power distributions and makes it possible to more uniformly burn the lower portions of high power fuel assemblies. However, the requirement to periodically exchange rod patterns has some adverse impacts. Operation is complicated and capacity factors are reduced by the need to reduce core power to perform an exchange. Also, the exchange of patterns excites spatial xenon oscillations which, although well damped, can complicate and delay the return to rated power after the exchange.

CCC Characteristics

In the CCC, control rods are inserted adjacent only to the low power control cell fuel assemblies while operating at power. A high power fuel assembly has at least one low power assembly between it and an inserted control rod. Figure 3 illustrates the change in the nodal power distribution of a low power control cell assembly as the adjacent control rod is withdrawn. Figure 4 shows the corresponding nodal power distribution in the highest power assembly adjacent to the control cell assemblies.

There are several important characteristics to note in these figures:

1) The highest peak nodal power in the control cell assembly is about 16% lower than that of the adjacent high power assembly.

2) The power distribution in the high power assembly is much less sensitive to control rod motion than in a conventional design.

3) The peak nodal power in the high power assembly is not significantly increased when the control rod is partially inserted.

These characteristics provide the following potential advantages relative to conventional designs:

1) **The need for conventional rod pattern exchanges is eliminated**

Since the control cell fuel operates at a substantially lower peak nodal power than the highest power fuel in the core, the effects of control blade history on local peaking, and the effects of shallow control rods on axial burnup and power distributions, can be accommodated without the need for conventional rod pattern exchanges. For CCC backfit applications in plants with

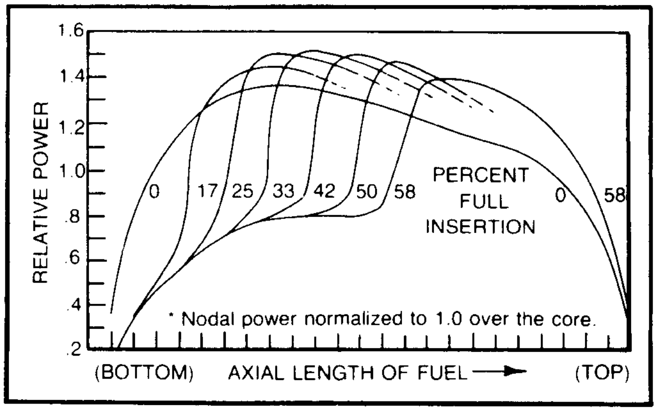


Figure 3. Power in Control Cell Fuel as Control Rod of Cell is Withdrawn Toward Bottom*

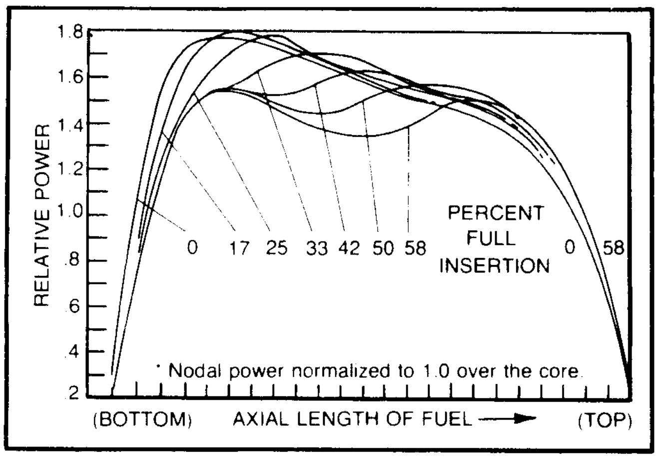


Figure 4. Power in Fresh Fuel Next to Control Cell as Control Cell Control Rod is Withdrawn Toward Bottom*

D-lattice geometry (nonsymmetric water gaps) a deep/shallow exchange within the cell rods may be needed once or twice during an annual cycle to mitigate control blade history effects. No exchanges will be required in plants with symmetric water gaps (C-lattice).

2) **Shallow control rods are easier to use**

Since shallow control cell rods do not cause large distortions in the axial power and burnup distributions of high power fuel assemblies, they are easier and more effective to use for axial power shaping than in conventional cores. As seen in Figure 4, there is little or no underburning of the high power fuel due to shallow rod insertion. Comparing Figures 2 and 4 it can also be seen that a shallow control cell rod is actually more effective in reducing the axial peak in a high power assembly than is a shallow rod in a conventional core.

3) **There is potential for lower peak kW/ft**

In the CCC, the partial insertion of a control cell rod does not significantly increase the peak nodal power of nearby high power assemblies whereas in conventional cores there can be a 3% or more increase. Also, with the CCC, the peak nodal power of the high power assemblies would not be significantly increased if shallow rods caused an underburning of the lower portions of the control cell fuel assemblies.

4) **Margin to the onset of transition boiling is increased in partially controlled assemblies**

In the CCC, the integrated power in a control cell assembly adjacent to a partially inserted control rod is typically at least 20% lower than that of the high power, uncontrolled fuel assemblies in the core. Therefore, any decrease in the margin to the onset of transition boiling due to the distortion of the local and axial power distribution by an inserted control rod can be accommodated without affecting the minimum thermal margin of the core.

5) **Fuel duty is reduced**

The maximum change in kW/ft (Δ kW/ft) experienced by a node in a high power fuel assembly due to a one notch (6 inch) control rod withdrawal is reduced by about a factor of five with the CCC relative to a conventional core. Also, the peak nodal power of the fuel which is experiencing adjacent control rod movement is lower in the CCC design.

In addition to the effects apparent from Figures 2 to 4, there are several other potential advantages which result from the fuel loading and operating strategy used in the CCC. They are:

6) **Significant burnup is not accumulated with control rods inserted adjacent to fuel assemblies containing unburned gadolinia**

The control cells are composed of either low enrichment assemblies without gadolinia or high exposure assemblies in which the initial gadolinia has been depleted. In conventional cores, the interaction between inserted control rods and adjacent assemblies with unburned gadolinia is quite complicated and tends to make core power distributions more sensitive to deviations from pre-planned control rod patterns and to uncertainties in predictive methods.

7) **Control rod patterns are always one-eighth core symmetric**

The control cells are located in octant symmetric positions in the core. In conjunction with an octant symmetric loading pattern, this provides maximum in-core instrumentation coverage so that most assemblies have two applicable LPRM and TIP readings. In conventional cores, the B1 and B2 rod patterns are not octant symmetric so that there is not redundant instrumentation coverage. Octant symmetry contributes to simpler operation, improved thermal margins and increases instrumentation and process computer reliability.

8) **Less fuel experiences adjacent control rod motion**

Control rod motion at power is experienced only by control cell fuel selected from the highest exposure batch in the core. A typical 748-bundle CCC equilibrium cycle core has 37 control cells containing 148 assemblies. Since a normal reload batch size for such a core is at least 190 assemblies, about 22% of the fuel assemblies loaded in the core will never experience adjacent control rod motion while operating at power during their entire residence in the core. In conventional cores, all of the fuel assemblies will normally experience adjacent control rod motion at various times during their cycles of residence in the core.

Operating Plant Demonstration

A demonstration of the CCC concept is currently in progress in cycle 6 of the Millstone Nuclear Power Station Unit 1. The Millstone core contains 580 fuel assemblies and operates at a power density of 40.8 kW/ℓ. In cycle 6, the core is comprised of 188 assemblies of the 7x7 fuel design and 392 assemblies of the 8x8 design.

The CCC was implemented in Millstone by changing only the refueling and control rod patterns. No change was made in the reload bundle design. The core loading pattern for cycle 6 is illustrated in Figure 5. The reload pattern is one-eight core symmetric as are the planned rod patterns. The control cells are comprised of the highest burnup 8x8 fuel assemblies from the third reload batch. Although of lower reactivity than the 8x8 fuel, the 7x7 reload batch 1 and 2 fuel was not loaded in control cells, thereby avoiding any control rod motion adjacent to the 7x7 fuel while operating at power.

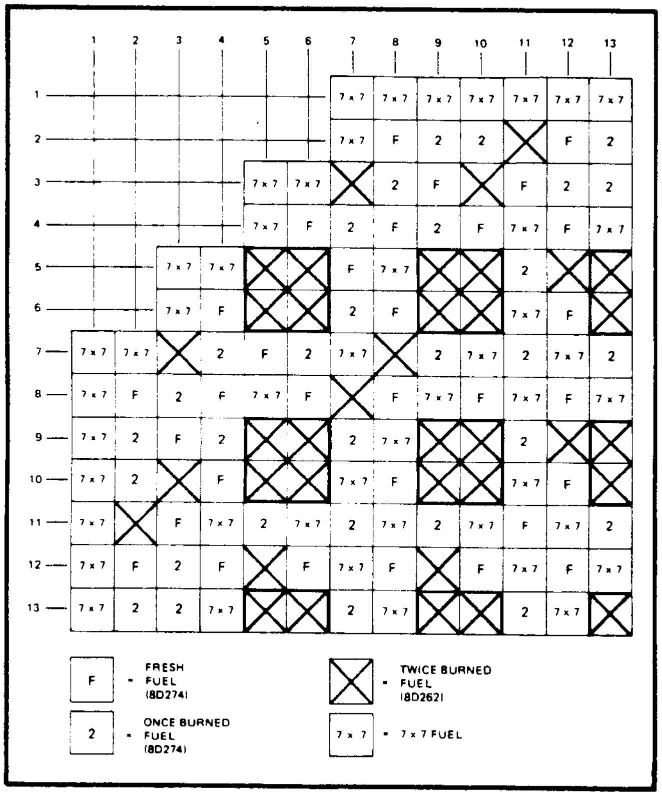


Figure 5. Millstone Control Cell Core Design Cycle 6

The Millstone reload pattern illustrates the flexibility of the basic CCC concept to adapt to a specific operational state. The capability to utilize existing bundles in the control cells, while not ideal in the long term, enables the CCC concept to be immediately backfit to an operating plant, thereby obtaining most of the CCC performance improvements at an early date. The Millstone CCC also differs from the ideal or optimum CCC design in that the control cells contain fuel with an average beginning-of-cycle burnup of 10 to 13 GWd/st rather than high exposure, last cycle fuel. As a result, the control cell fuel at Millstone will operate at a somewhat higher power than would be the case in a more optimum CCC application.

The Millstone CCC demonstration started up in April, 1978. The experience to date has been excellent. The pre-planned operating strategy is being closely followed and is meeting the primary demonstration objectives of simplifying operations and improving capacity factor.

The Millstone CCC demonstration cycle will continue until the Spring of 1979. It is expected that the CCC operating experience will continue to be favorable throughout the duration of cycle 6 and that the CCC design will be continued in subsequent cycles of Millstone operation.

Conclusion

The Control Cell Core improved design has the potential to simplify the operation and increase the thermal margins and capacity factors of present and future BWRs. Feedback to date indicates that these improvements are being realized during the operation of Millstone Unit 1. The feasibility of implementing a CCC demonstration on a BWR/4 plant is currently being studied and, if favorable, would result in program initiation in late 1979. Until additional design and operating experience is gained for a variety of reactor core conditions, the feasibility of, and incentives for, implementing the CCC concept must be carefully evaluated on a case-by-case basis.

Section 4

Reactor Auxiliary Systems

INTRODUCTION

Because the reactor is basically a water boiler, process systems are required which clean and control the chemistry of the water in the reactor vessel as well as protect the reactor core. Called the reactor auxiliary systems, these systems may be divided into two general categories: systems necessary for normal nuclear boiler operations, including startup and shutdown; and systems which accommodate or provide backup in case of an abnormal condition.

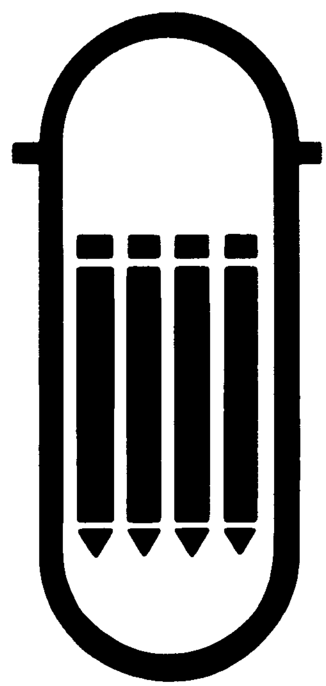
Auxiliary systems used during normal plant operation include the reactor water cleanup system, the fuel building and containment pools cooling and filtering system, the closed cooling water system for reactor services, the shutdown cooling function of the residual heat removal system, radioactive waste treatment system, and offgas treatment system.

Backup auxiliary systems used during abnormal plant operation include the reactor core isolation cooling system, the standby liquid control system, the steam condensing function of the RHR system (hot standby), and the suppression pool cooling function of the RHR system. Other process systems, commonly referred to as emergency core cooling systems (ECCS) are designed as safety systems to mitigate the consequences of postulated emergency situations that could otherwise lead to core damage and release of fission products to the environs. Emergency core cooling systems consist of the low pressure coolant injection function of the residual heat removal system, the high and low pressure core spray systems, and automatic depressurization (blowdown). The essential service water system and the area cooling systems which service the areas where ECCS equipment is located are also required during abnormal plant operation.

REACTOR WATER CLEANUP SYSTEM

The purpose of the reactor water cleanup system (Figure 4-1) is to maintain high reactor water quality by removing fission products, corrosion products, and other soluble and insoluble impurities. In addition, the system provides a means for water removal from the primary system during periods of increasing water volume.

The cleanup system is sized to process the water volume of the reactor system in approximately 3-1/2 hours. The system can be operated during startup, shutdown, and refueling operation, as well as during normal plant operations.



Water is removed from the reactor through the reactor recirculation pump suction line and returned through the feedwater line. Under normal operation, the water is removed at reactor temperature and pressure and pumped through regenerative and nonregenerative heat exchangers where it is cooled, and then through the filter-demineralizer units. The flow continues through the shell side of the regenerative heat exchanger where it is heated before returning to the reactor.

During startup and other times of increasing water volume, excess water is normally removed from the reactor by blowdown through the cleanup system to the main condenser, or alternately to the waste collector tank, or waste surge tank. During this operation, the return flow to the regenerative heat exchanger is reduced, thereby reducing the cooling capability of this exchanger and correspondingly increasing the duty of the nonregenerative heat exchanger. The nonregenerative heat exchanger is designed to cool reactor water flow to the filter-demineralizer units to approximately 120°F (49°C) during both normal operation and reactor vessel blowdown. Cooling water is supplied to the nonregenerative heat exchanger by the closed cooling water system for reactor service.

The operation of the reactor water cleanup system is controlled from the main control room. Filter resin backwashing and precoating operations are controlled from a local panel. The cleanup system is isolated from the reactor automatically by closure of motor-operated isolation valves on any of the following signals:

- High temperature after the nonregenerative heat exchanger
- Low reactor water level
- Standby liquid control solution injection
- High ambient temperature in the cleanup system equipment area
- High flow rate differential between system inlet and outlet
- High differential temperature across the system's ventilation system.

FUEL BUILDING AND CONTAINMENT POOLS COOLING AND CLEANUP SYSTEM

The fuel building and containment pools cooling and cleanup system (Figure 4-2) accommodates the spent fuel cooling heat load as well as drywell heat transferred to the upper containment pool. The equipment for the cooling and cleanup systems consists of circulating pumps, heat exchangers, filter-demineralizers,

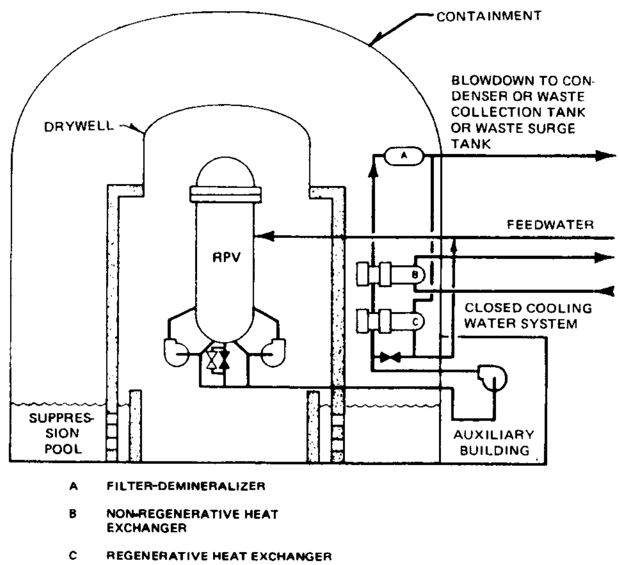


Figure 4-1. Reactor Water Cleanup System

and the required piping, valves, and instrumentation. Pumping loops circulate pool water through the heat exchangers and fuel pool filters and return the flow by discharging it through diffusers mounted in the fuel storage pool and in the containment pool. The suction for the circulating pumps is taken from the skimmer surge tank. The skimmer surge tank is fed from skimmers located at the top of these pools.

The upper containment pool has a shield wall with a removable gate between the reactor well, the fuel holding pool and the fuel transfer pool. With the gate inserted in the slot, the upper containment pool can be drained for work at the pressure vessel flange level. With the pools full of water, the gates are removed during refueling operations to permit the transfer of fuel and equipment between pools.

The residual heat removal system heat exchangers are also available to supplement the fuel pool cooling heat exchangers. The need for the residual heat removal system heat exchangers is not normally required but may be needed when more than the normal number of spent fuel assemblies are stored in the pool.

The system pumps and heat exchangers are located in the fuel building below the normal fuel pool water level. The heat exchangers are cooled by essential plant service water.

Control of water clarity, radioactivity levels, and purity of the pools is accomplished by filtration-demineralization. The filter-demineralizer units, which

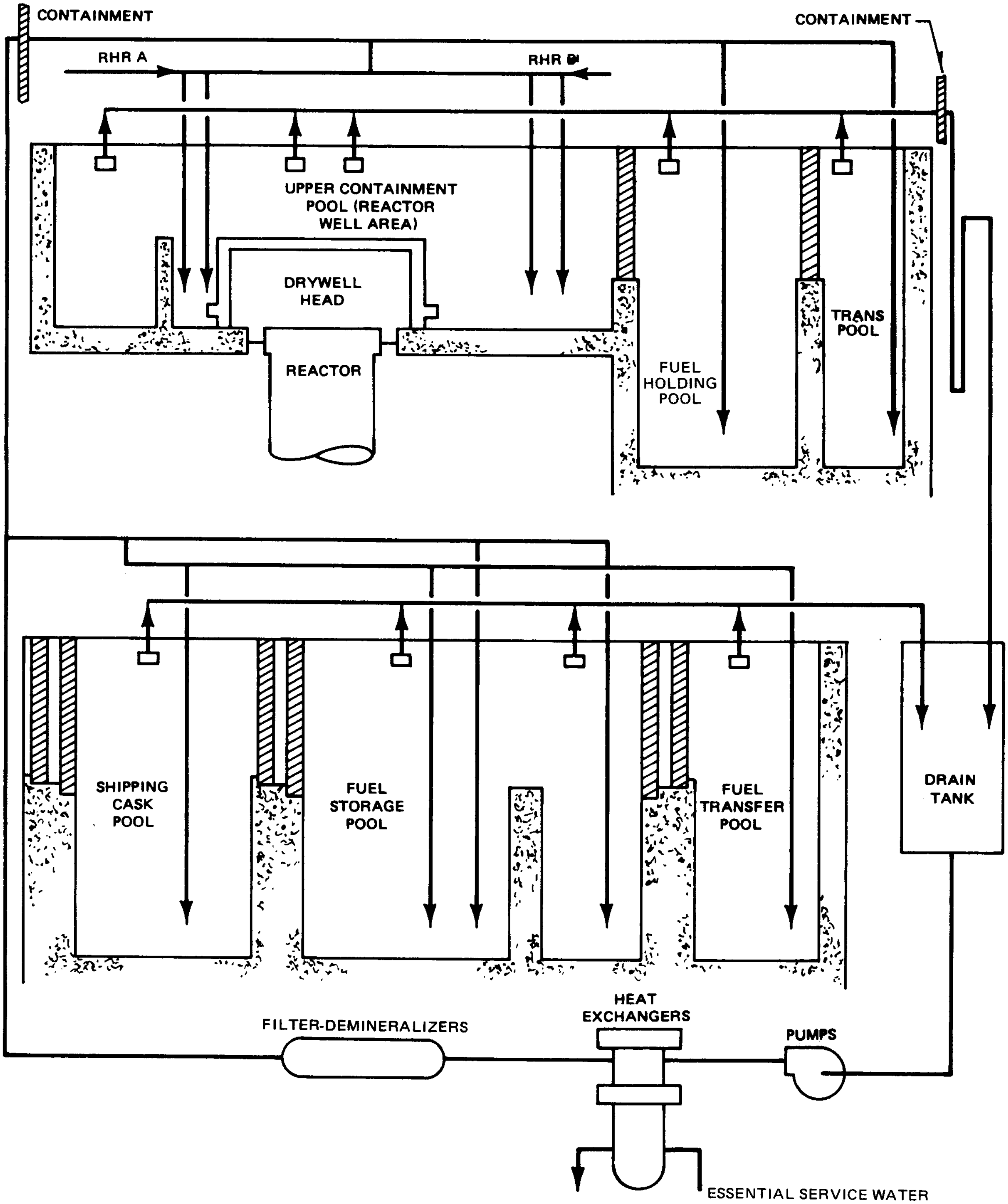


Figure 4-2. Fuel Building and Containment Pools Cooling and Cleanup System

may become radioactive as they collect corrosion products, are normally located in the fuel building. Pool water is usually filtered continually.

Because there are no drain connections at the bottom of the fuel storage pool, the spent fuel assemblies can never be exposed by an accidental valve opening or pipe break. Fuel is not stored in the upper containment pool during normal operation. A portable underwater vacuum system, similar to that used in swimming pools, can be used to clean pool walls, floors, and internals removed from the reactor vessel. Deposition at the water line of the pool walls is minimized by a number of surface skimmers. These devices remove a portion of the surface water and recycle it to the pool.

CLOSED COOLING WATER SYSTEM FOR REACTOR SERVICE

The closed cooling water system (Figure 4-3) consists of a separate, forced circulation loop. This system uses water piped from the site service water source to provide a heat sink for selected nuclear system equipment. Its purpose is to provide a second barrier between the primary systems containing radioactive products and the service water system that is the final heat sink and, therefore, eliminates the possibility of radioactive discharge into plant effluents that could result from heat exchanger leaks. The plant service water pumps provide coolant to the closed cooling water system for reactor service which in turn generally service the following equipment:

- Reactor recirculation pump seal coolers
- Reactor recirculation pump motor coolers
- Nonregenerative cleanup heat exchanger
- Clean sump coolers
- Sample coolers
- Drywell coolers
- Cleanup recirculation pump coolers
- Offgas system glycol coolers
- Control rod drive supply pumps
- Radwaste concentrator condensers
- Radwaste concentrated waste tank
- Control rod drive supply pumps

Any possible radioactive leakage from the foregoing reactor equipment would be to, and would be confined in, the closed loop cooling water system which is monitored continuously for radioactivity. A surge tank is used to accommodate system volume swell and shrinkage and to provide a means for adding makeup water and inhibitors.

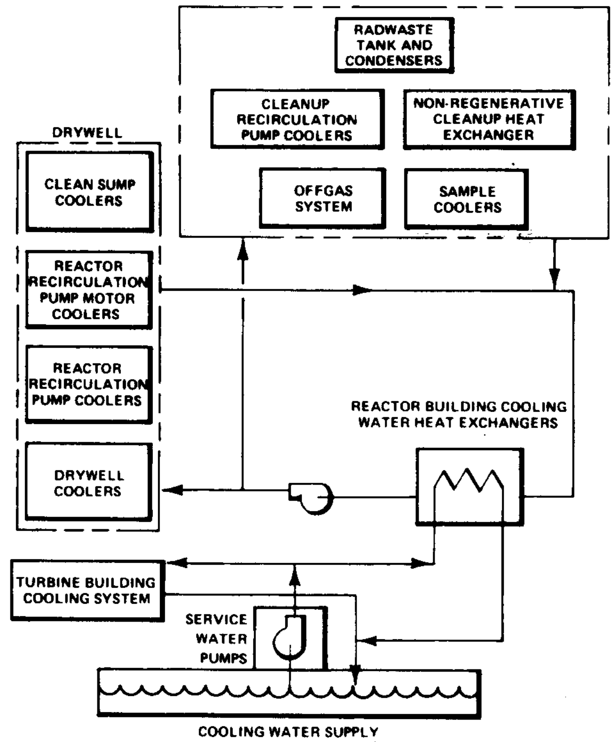


Figure 4-3. Closed Cooling Water System for Reactor Service

The closed cooling water system design temperature depends on the maximum temperature of service water intake. The closed cooling water system satisfies the plant's full power load requirements. Extra cooling capability, with all spares operating, is adequate to handle plant startup duty.

EMERGENCY EQUIPMENT COOLING SYSTEM

An emergency equipment cooling system services certain equipment required for normal and emergency shutdown of the plant. The system provides cooling water for the residual heat removal system pump motor and pump seal cooler and the high pressure and low pressure core spray systems pump motors and pump seal coolers. Upon loss of normal ventilation such as may occur upon loss of external a-c power, the emergency equipment cooling system provides ventilation cooling for the high pressure and low pressure core spray systems, the residual heat removal system, and the reactor core isolation cooling system equipment as required to prevent overheating. On failure of any single component, the emergency equipment cooling system will service at least two residual heat removal system pumps or one residual heat removal system pump and the low pressure core spray system pump, the high pressure core spray system pump, and any standby core cooling system equipment being cooled by ventilation equipment being serviced by the emergency equipment cooling system.

STANDBY LIQUID CONTROL SYSTEM

The standby liquid control system (Figure 4-4) is a redundant control system capable of shutting the reactor down from rated power operation to the cold condition in the postulated situation that the control rods cannot be inserted. The operation of this system is manually initiated from the reactor control room. No operating BWR has ever required the use of the standby liquid control system.

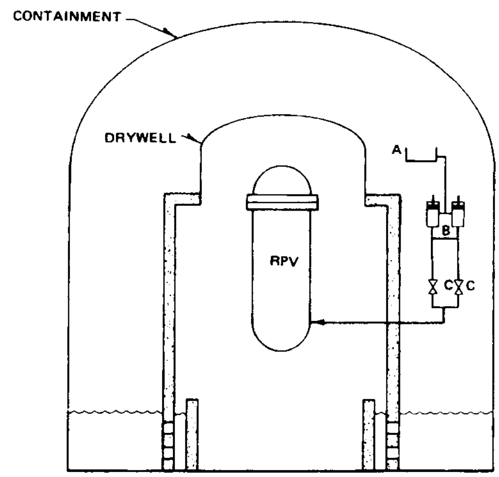
The equipment for the standby liquid control system is located in the reactor building and consists of a stainless steel storage tank, a pair of full capacity positive displacement pumps and injection valves, a test tank, and the necessary piping, valves, and instrumentation.

The standby liquid control system is adequate to bring the reactor from the hot operating condition to cold shutdown and to hold the reactor shut down with an adequate margin when considering temperature, voids, Doppler effect, equilibrium, xenon, and shutdown margin. It is assumed that the core is operating at normal xenon level when injection of liquid control chemical is needed.

The liquid control chemical used is boron in the form of sodium pentaborate solution. It is injected into the bottom of the core where it mixes with the reactor coolant. The sodium pentaborate is stored in solution in the standby liquid control tank. Electric heaters automatically keep the solution above the saturation temperature. The system temperature and liquid level in the storage tank are monitored and abnormal conditions are annunciated in the control room.

Analyses, backed by results from plant startup and operation, have shown that automatic initiation of the standby liquid control system is not required,* and that ample time and signals are available to allow for safe shutdown by manual operator action. When initiation of liquid control chemical is required, the operator inserts a key into the key switch to open all system valves and to start one of the pumps. Use of the key switch minimizes the probability of an accidental injection of the liquid chemical solution. The admission valves for this system are of the explosive type to provide a high assurance of opening when actuated and to prevent boron from leaking into the reactor when the injection pumps are being tested.

* Automatic initiation may be required by the Nuclear Regulatory Commission as defined in NUREG 0460 (Anticipated Transients without Scram for Light Water Reactors).



A STORAGE TANK
B PUMPS
C EXPLOSIVE VALVES

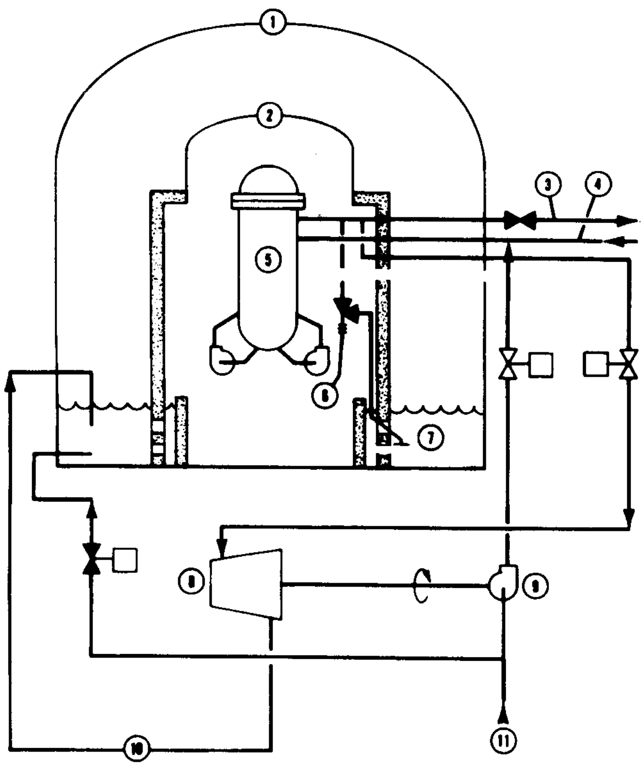
Figure 4-4. Standby Liquid Control System

To verify the operability of components, testing is done in two parts. With the injection valves closed, the pumps are tested by pumping demineralized water from the test tank through a test line and back to the test tank. During plant shutdown, the injection system can be tested by firing the explosive valves and pumping demineralized water from the test tank into the reactor vessel. The operability of the injection system is done during plant shutdown to permit the replacement of the explosive charges. A low no-fire current can be passed through the primer in the trigger assembly of each of the explosive valves to monitor circuit continuity to assure that the valve is in state of firing readiness. The containment isolation valves (two check valves located in series near the drywell penetration) are tested by means of a bleed-off between the two.

Should the liquid control system ever be used to shut down the reactor, the sodium pentaborate would be removed from the primary system by flushing for gross dilution and by operation of the reactor cleanup system for final polishing.

REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

The reactor core isolation cooling system (Figure 4-5) maintains sufficient water in the reactor pressure vessel to cool the core and then maintain the nuclear



1. CONTAINMENT
2. DRYWELL
3. MAIN STEAM
4. FEEDWATER
5. RPV
6. SAFETY/RELIEF
7. SUPPRESSION POOL
8. TURBINE
9. TURBINE DRIVEN MAKEUP PUMP
10. TURBINE EXHAUST
11. CONDENSATE FROM CONDENSATE STORAGE TANK OR RHR HEAT EXCHANGER

Figure 4-5. Reactor Core Isolation Cooling System

boiler in the standby condition in the event the vessel becomes isolated from the turbine steam condenser and from feedwater makeup flow. The system also allows for complete plant shutdown under conditions of loss of the normal feedwater system by maintaining the necessary reactor water inventory until the reactor vessel is depressurized, allowing the operation of the shutdown cooling function of the residual heat removal system. The system delivers rated flow within 30 seconds after initiation. A "water leg" pump keeps the piping between the pump and the discharge shutoff valve full of water to ensure quick response and to eliminate potential hydraulic damage on system initiation.

Following a reactor scram during normal plant operation, steam generation continues at a reduced rate due to the core fission product decay heat. The turbine bypass system directs the steam to the main condenser, and the feedwater system provides makeup water required to maintain the reactor vessel inventory.

In the event the reactor vessel becomes isolated from the main condenser, the relief valves automatically (or by operator action from the control room) maintain vessel pressure within desirable limits. In the event feedwater becomes unavailable, the water level in the reactor vessel drops due to continued steam generation by decay heat. Upon reaching a predetermined low level, utilizing one-out-of-two-twice logic, the RCIC system is initiated automatically to maintain safe standby conditions of the isolated primary system. The turbine-drive pump supplies makeup water from one of the following sources capable of being isolated from other systems: the condensate storage tank (first source), the steam condensed in the RHR heat exchangers (second source), or the suppression pool (an emergency source). The turbine is driven with a portion of the decay heat steam from the reactor vessel and exhausts to the suppression pool.

The makeup water is pumped into the reactor vessel through a connection to the reactor feedwater line.

A design flow functional test of the RCIC system may be performed during normal plant operation by drawing suction from the condensate storage tank and discharging through a full flow test return line to the condensate storage tank. The discharge valve to the reactor feedwater line remains closed during the test, and reactor operation remains undisturbed. If the system requires initiation while in the test mode, the control system automatically returns to the operating mode.

Cooling water for pump and turbine operations and for the lube oil cooler and the gland seal condenser is supplied from the discharge of the pump.

The RCIC system operates independently of auxiliary a-c power, plant service air, or external cooling water systems. System valves and auxiliary pumps are designed to operate by d-c power from the station batteries.

Two turbine control systems include a speed governor limiting the speed to its maximum operating level and a control governor with automatic set-point adjustment which is positioned by demand signal from a flow controller. Manual operation of the control governor is possible when in the test mode, but automatically repositioned by the demand signal from the controller if system initiation is required. The operator has the capability to select manual control of the governor, and adjust power and flow to match decay heat steam generation.

The turbine and pump automatically shut down upon:

- turbine overspeed,
- high water level in the reactor vessel,
- low pump suction pressure,
- high turbine exhaust pressure, and
- automatic isolation signal.

The steam supply system to the turbine is automatically isolated upon:

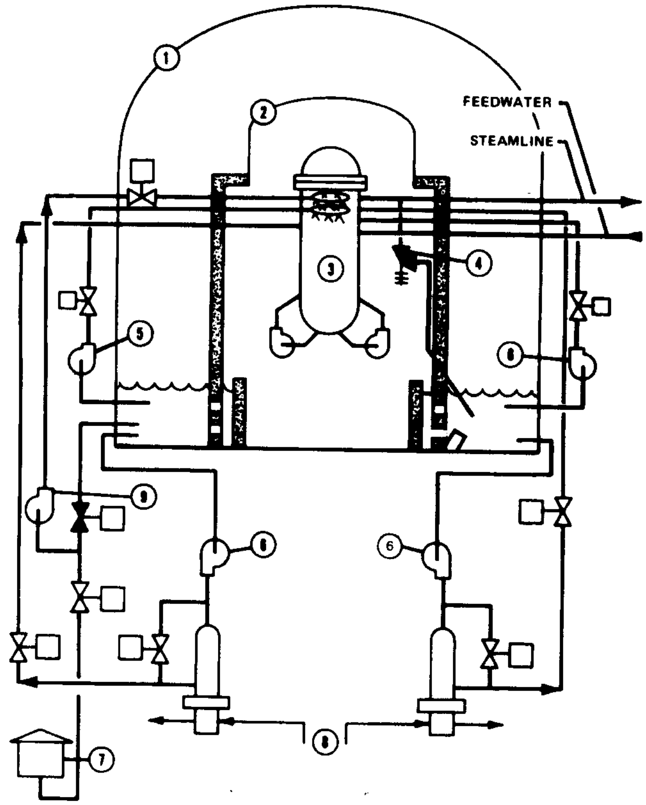
- high pressure drop across two pipe elbows in the steam supply line,
- high area temperature,
- low reactor pressure (two-out-of-two logic), and
- high pressure between the turbine exhaust rupture diaphragms (two-out-of-two logic).

EMERGENCY CORE COOLING SYSTEM

The emergency core cooling system (ECCS) comprises the low pressure coolant injection function of the residual heat removal system, the high and low pressure core spray systems, and automatic depressurization of the primary system. The ECCS (Figure 4-6) is designed to perform the following:

- Prevent fuel cladding fragmentation for any mechanical failure of the nuclear boiler system up to, and including, a break equivalent to the largest nuclear boiler system pipe.
- Provide this protection by at least two independent, automatically actuated cooling systems.
- Function with or without external (off-site) power sources.
- Permit testing of all emergency core cooling systems by acceptable methods, including, wherever practical, testing during power plant operations.
- Provide this protection for long time periods and from secure sources of cooling water with the capability of dissipating the rejected heat for a minimum of 30 days.

The aggregate of the ECCS is designed to protect the reactor core against fuel cladding damage (fragmentation) across the entire spectrum of line break accidents. The operational capability of the various ECCS's to meet functional requirements and performance objectives is outlined in the following paragraphs.



1. CONTAINMENT
2. DRYWELL
3. RPV
4. SAFETY/RELIEF VALVE DEPRESSURIZATION
5. LOW PRESSURE SPRAY
6. LOW PRESSURE COOLING INJECTION FUNCTION OF RESIDUAL HEAT REMOVAL SYSTEM
7. CONDENSATE STORAGE TANK
8. SERVICE WATER
9. HIGH PRESSURE CORE SPRAY

Figure 4-6. Emergency Core Cooling System

The operation of the ECC network is automatically activated by the reactor protection system upon redundant signals that are indicating low reactor vessel water level or high drywell pressure or a combination of indicators showing low reactor vessel water level and high drywell pressure.

During the first 10 minutes following initiation of operation of the ECCS, any one of the following three combinations satisfies the functional requirements of the system objectives:

- The operation of the automatic depressurization function, the high pressure core spray system, and two low pressure coolant injection loops of the residual heat removal system. (Failure of Division 1)
- The operation of the automatic depressurization function, the high pressure core spray system, the

low pressure core spray system, and one low pressure coolant injection loop of the residual heat removal system. (Failure of Division 2)

- The operation of the automatic depressurization function, three low pressure coolant injection loops of the residual heat removal system, and the low pressure core spray system. (Failure of Division 3)

In the event of a break in a pipe that is part of the ECCS, any one of the following four combinations satisfies the functional requirement:

- The operation of the automatic depressurization function and two low pressure coolant injection loops of the residual heat removal system.
- The operation of the automatic depressurization function, one low pressure coolant injection loop of the residual heat removal system, and the low pressure core spray system.
- The operation of the automatic depressurization function, the high pressure core spray system, and one low pressure coolant injection loop of the residual heat removal system.
- The operation of the automatic depressurization function, the high pressure core spray system, and the low pressure core spray system.

A combination of either the high pressure core spray system or the low pressure core spray system plus any two other ECCS pumps provides two phenomenological cooling methods (flooding and spraying).

After the first 10 minutes following the initiation of operation of the ECCS and in the event of an active or passsive failure in the ECCS or its essential support system, one of the following two combinations satisfies the performance objectives and the requirement for removal of decay heat from the containment.

- Two low pressure coolant injection loops of the residual heat removal system with at least one heat exchanger and 100% service water flow.
- Either the high pressure core spray system or the low pressure core spray system, one low pressure coolant injection loop of the residual heat removal ststem with one heat exchanger, and 100% service water flow.

The separation of redundant equipment of the various systems that make up the ECCS is maintained to assure maximum operational availability. Electrical equipment and wiring for the engineered safeguard features of the ECCS are broken into segregated divisions, further assuring a high degree of redundancy. Refer to Figure 4-7.

The power for operation of the ECCS system is from regular a-c power sources. Upon loss of the regular power, operation is from on-site standby a-c power sources. The standby diesel-generator set is capable of accommodating full capacity of the low pressure coolant injection and spray function. The high pressure core spray system is completely independent of external power sources, having its own diesel generator as shown.

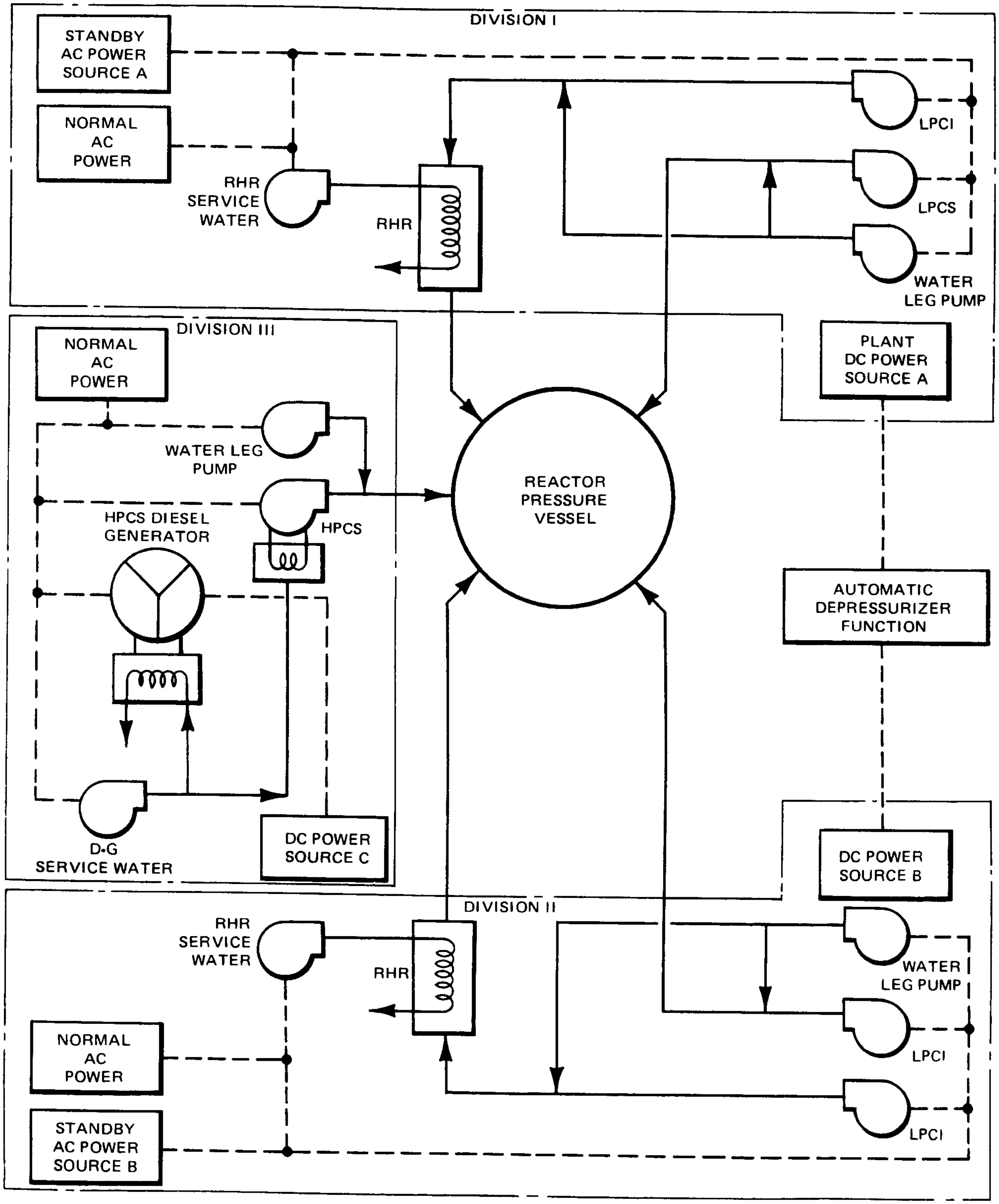
The operation of ECCS pumps is also possible from a local keylock hand switch and from the control room. Automatic signals pre-empt all others. Should normal power fail while the system is operating or in the process of going into operation, the system will restart from the standby sources. All system alarms annunciate in the control room.

Although the feedwater system is not considered a part of the ECCS, under some circumstances, it could either refill the vessel or at least maintain a water level, depending upon the location of the postulated break, for a given spectrum of break sizes. In the case of turbine-driven feedwater pumps, this additional coolant source would still be available from the electrically driven condensate pumps.

High Pressure Core Spray System

The purpose of the high pressure core spray system (Figure 4-8) is to depressurize the nuclear boiler system and to provide makeup water in the event of a loss of reactor coolant inventory. In addition, the high pressure core spray system prevents fuel cladding damage (fragmentation) in the event the core becomes uncovered due to loss of coolant inventory by directing this makeup water down into the area of the fuel assemblies. The makeup water is jetted as a spray over the area of the fuel assemblies from nozzles mounted in a sparger ring located inside the reactor vessel above the fuel assemblies. The high pressure core spray system is an integral part of the total design for ECC which provides for adequate core cooling and depressurization for all rates of coolant loss from the nuclear boiler.

The high pressure core spray system includes a sparger ring with spray nozzles located inside the reactor pressure vessel, a motor-driven pump, diesel-



RHR – RESIDUAL HEAT REMOVAL
HPCS – HIGH PRESSURE CORE SPRAY
LPCS – LOW PRESSURE CORE SPRAY
LPCI – LOW PRESSURE COOLANT INJECTION MODE OF RHR
— — — ELECTRICAL
——— PIPING

Figure 4-7. Emergency Cooling System Network generator, valves, piping and instrumentation necessary to provide an operating system with the capability of being tested during plant operation.

Cooling water for the operation and testing of the high pressure core spray system is from the condensate storage tank. Upon depletion of this supply, the system automatically transfers to the water in the containment suppression pool. Water inventory lost from the nuclear boiler system drains to the drywell to weir wall level and then into the suppression pool thereby providing an inexhaustible supply of cooling water allowing continued operation of the high pressure core spray system until it is manually stopped by the operator from the control room. System piping and equipment are maintained full of condensate water at all times to avoid time delays in filling the lines and to avoid hydraulic hammer.

The high pressure core spray system can operate independently of normal auxiliary a-c power, plant service air, or the emergency cooling water system. Operation of the system is automatically initiated from independent redundant signals indicating low reactor vessel water level or high pressure in the primary containment. The system also provides for remote-manual startup, operation, and shutdown. A testable check valve in the discharge line prevents back flow from the reactor pressure vessel when the reactor vessel pressure exceeds the high pressure core spray system pressure such as may occur during initial activation of the system. A low flow bypass system is placed into operation until pump head exceeds the nuclear system pressure and permits flow into the reactor vessel.

The high pressure core spray system can be tested during normal plant operation or when the plant is shut down. During normal plant operation, pump suction is from the condensate storage tank with a full flow return line to the condensate storage tank. During plant shutdown, pump suction is from the primary containment pressure suppression pool with a full flow return line to the suppression pool. The control system provides for the automatic transfer to the service mode upon the presence of ECC demand signal.

The integrity of the piping internal to the reactor vessel is determined by comparing the difference in pressure between spray sparger and the bottom of the core area with the pressure drop across the core. An increase in this comparison initiates an alarm in the control room.

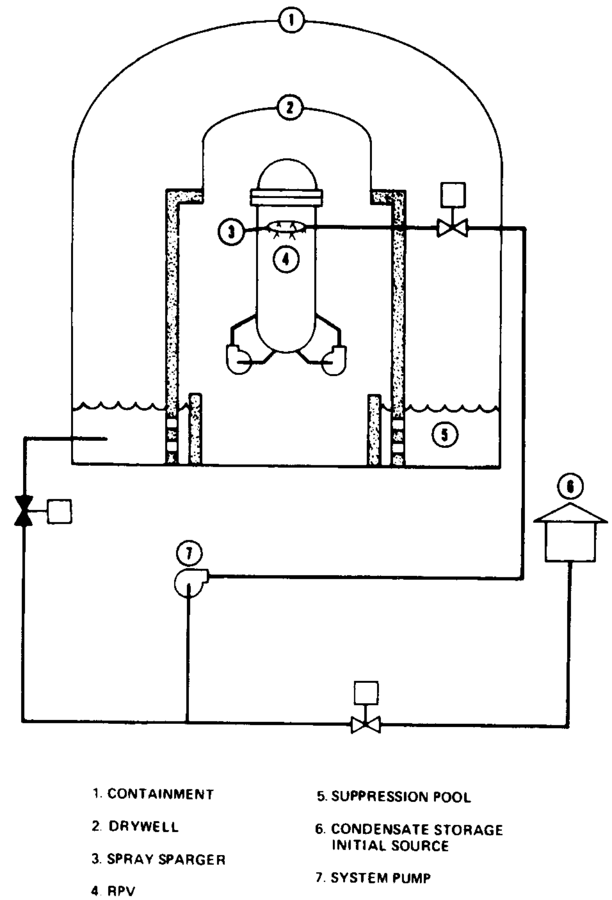


Figure 4-8. High Pressure Core Spray System

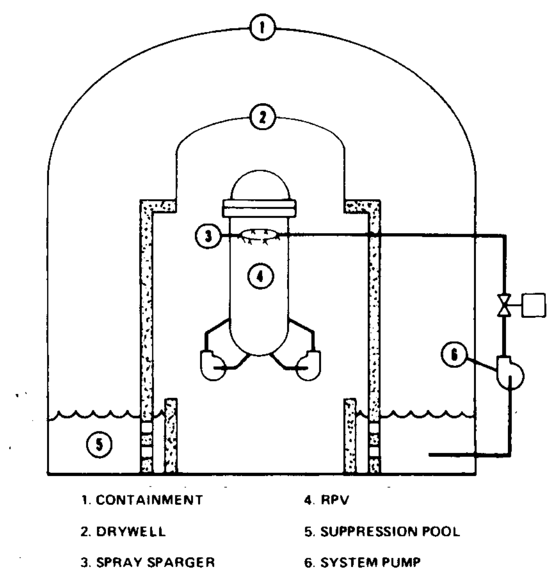


Figure 4-9. Low Pressure Core Spray System

Low Pressure Core Spray System

The function of the low pressure core spray system (Figure 4-9) is to prevent fuel cladding damage (fragmentation) in the event the core is uncovered by the loss of coolant. The cooling effect is accomplished by directing jets of water down into the fuel assemblies from spray nozzles mounted in a sparger ring located above the reactor core. The system is an integral part of the total design for ECC which provides for adequate core cooling for all rates of coolant loss from the nuclear boiler. The system goes into operation once the reactor vessel pressure has been reduced and the operation of the other systems of the ECCS prove inadequate to maintain the necessary water level in the reactor vessel at the reduced vessel pressure.

The low pressure core spray system includes a sparger ring with spray nozzles located in the reactor vessel above the core, a motor-driven pump, motor-operated valves, piping, valves, and instrumentation necessary to provide a system for required operation with the capability of being tested. The system is connected to the containment suppression pool for its supply of water for cooling and connectable to the residual heat removal system for testing and flushing. The elevation of the pump, with respect to the minimum water level of the suppression chamber, ensures adequate net positive suction head. The system pump is protected from overheating during operation against high reactor vessel pressure or closed injection or test valves by a low-flow bypass line to the suppression pool. A "water leg" pump keeps the piping between the pump and the injection valve full of water to ensure quick response and to eliminate potential hydraulic damage on system initiation.

In the event of complete loss of normal electrical power, the spray system may be operated (automatically or manually) from the standby diesel-generator.

The operation of the low pressure core spray system pump is initiated from independent, redundant signals indicating low-low-reactor-water level and/or high pressure in the drywell, both using a one-out-of-two-twice logic. (The same signals initiate starting of the standby diesel generators.) The motor-operated valve in the discharge line opens automatically upon activation of the pump and a permissive pressure differential across the valve. As the reactor vessel pressure decreases, the flow rate of water to the reactor vessel increases. A testable check valve in the discharge line located inside the containment precludes back flow from the reactor vessel when the vessel pressure is greater than the pump discharge pressure. The operation of the system can be initiated from the main control room.

Water lost from the reactor vessel collects in the drywell to the level of the weir wall and then flows into the suppression chamber. This establishes a closed loop allowing the spray system to continue to operate until it is manually stopped by the operator.

A bypass line to the suppression pool, capable of rated core spray flow, permits testing while the power plant is in service. A motor-operated valve controls bypass flow and is operated by a keylocked switch in the control room. The position of the valve (as is true for all air- or motor-operated valves) is indicated in the control room. The valve receives a signal to close, which pre-empts all others, in the event that operation of the low pressure core spray system is required.

To allow for system testing during plant shutdown, reactor water, via a temporary connection (removable spool piece) to the residual heat removal system, is discharged into the reactor vessel through the core spray sparger. The spool piece is removed prior to plant startup and the open pipe capped.

Automatic Depressurization Function

Blowdown, through selected safety/relief valves, in conjunction with the operation of the low pressure coolant injection function of the residual heat removal system and/or low pressure core spray system, functions as an alternate to the operation of the high pressure core spray system for protection against fuel cladding damage (fragmentation) upon loss of coolant over a given range of steam or liquid line breaks. The blowdown depressurizes the reactor vessel, permitting the operation of the low pressure coolant injection function and/or the low pressure core spray system. Blowdown is activated automatically upon coincident signals of low water level in the reactor vessel and high drywell pressure. A time delay of approximately 2 minutes after receipt of the coincident signals allows the operator time to bypass the automatic blowdown if the signals are erroneous or the condition has corrected itself. The operator can initiate blowdown from the control room at any time.

RESIDUAL HEAT REMOVAL (RHR) SYSTEM

The residual heat removal (RHR) system (Figure 4-10) removes residual heat generated by the core under normal (including hot standby) and abnormal shut-

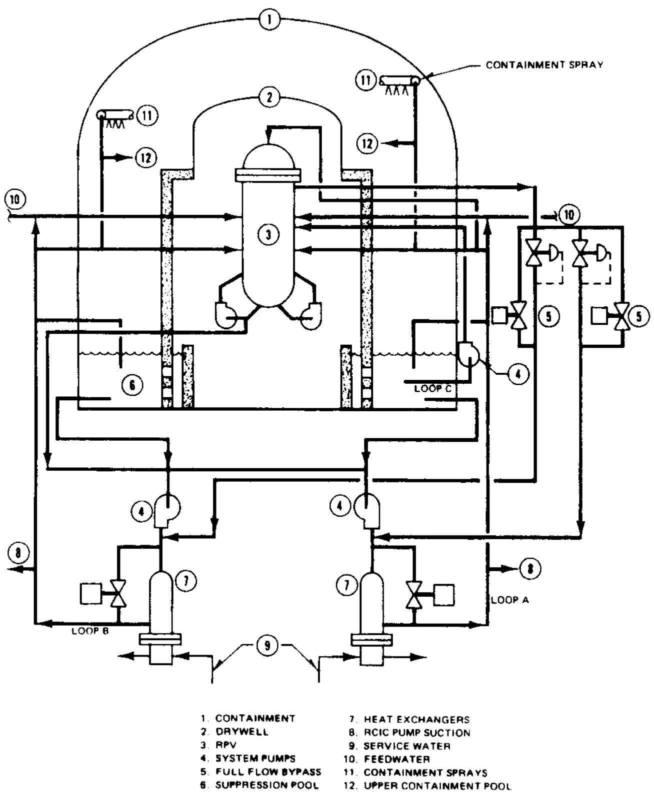


Figure 4-10. Residual Heat Removal System

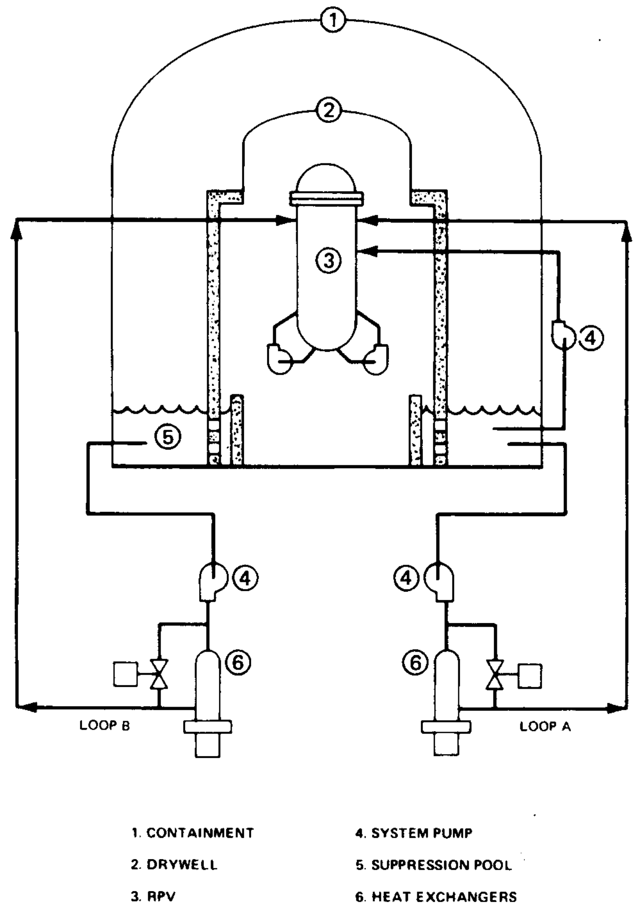


Figure 4-11. Residual Heat Removal System, Low Pressure Coolant Injection Function

down conditions. The low pressure coolant injection function of the residual heat removal system is an integral part of the ECCS. The design objectives of the system follow:

- To restore and maintain, if necessary, the water level in the reactor vessel after a loss-of-coolant accident so that the core is sufficiently cooled to prevent fuel cladding damage (fragmentation)
- To limit suppression pool water temperature
- To remove decay heat and sensible heat from the nuclear boiler system while the reactor is shut down for refueling and servicing
- To condense reactor steam so that decay and residual heat may be removed if the main condenser is unavailable (hot standby)
- To supplement the fuel and containment pools cooling and cleanup system capacity when necessary to provide additional cooling capability.

The RHR system is made up of various subsystems with the following operational functions to satisfy these objectives.

Low Pressure Coolant Injection (LPCI)*

The low pressure coolant injection function (Figure 4-11) in conjunction with the low pressure core spray system, the high pressure core spray system, and/or automatic depressurization of the nuclear boiler system (depending upon operability of the high pressure core spray system or level of depletion of reactor vessel water) will restore and maintain the desired water level in the reactor vessel required for cooling after a loss-of-coolant accident.

In conjunction with the low pressure core spray system, redundancy of capability for core cooling is achieved by sizing the RHR pumps so that the required flow is maintained with one pump not operating. Using a split bus arrangement for pump power supply (essential power system), two RHR pumps are connected to one bus and the third RHR pump and a low pressure

* Part of the emergency core cooling network.

core spray pump are connected to the second bus to obtain the desired cooling capability. The pumps deliver full flow inside the core shroud when the differential pressure beteen the reactor vessel and the containment approaches 20 psi (138 kPa).

The availability of the LPCI function is not required during normal nuclear system startup or cooldown when the reactor vessel gage pressure is less than 135 psi (931 kPa).

The operability of the pumps can be tested at any time during normal plant operation by bypassing the reactor vessel and pumping the flow back to the pressure suppression pool.

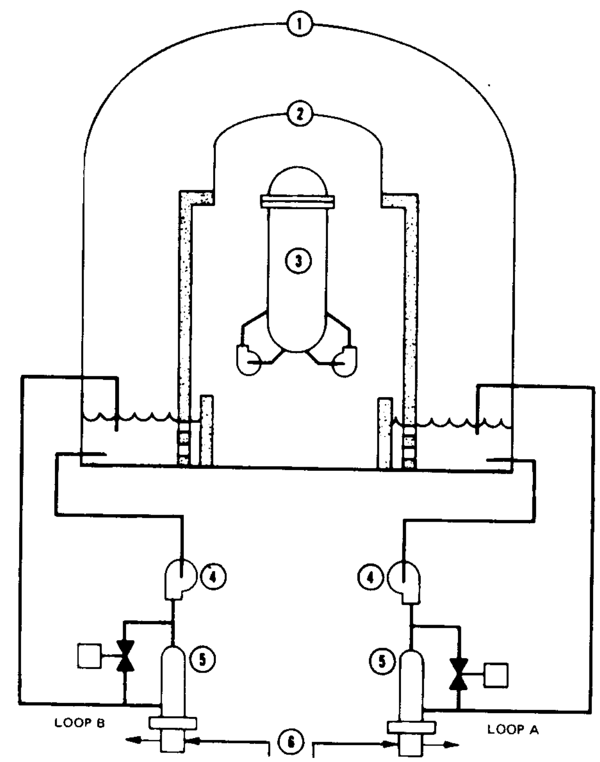
Suppression Pool Cooling

The suppression pool cooling function of the residual heat removal system (Figure 4-12) ensures that the temperature in the suppression pool immediately after blowdown, and when the reactor vessel gage pressure is greater than 135 psi (931 kPa) does not exceed a predetermined limit [generally 170°F (77°C)*]. Suppression pool water is pumped from the pool through either or both of two completely independent loops, including pump and heat exchanger, and returned to the pool. The heat removed by the heat exchanger is transferred to the residual heat removal system service water. Suppression pool cooling is manually initiated.

Reactor Steam Condensing (Hot Standby)

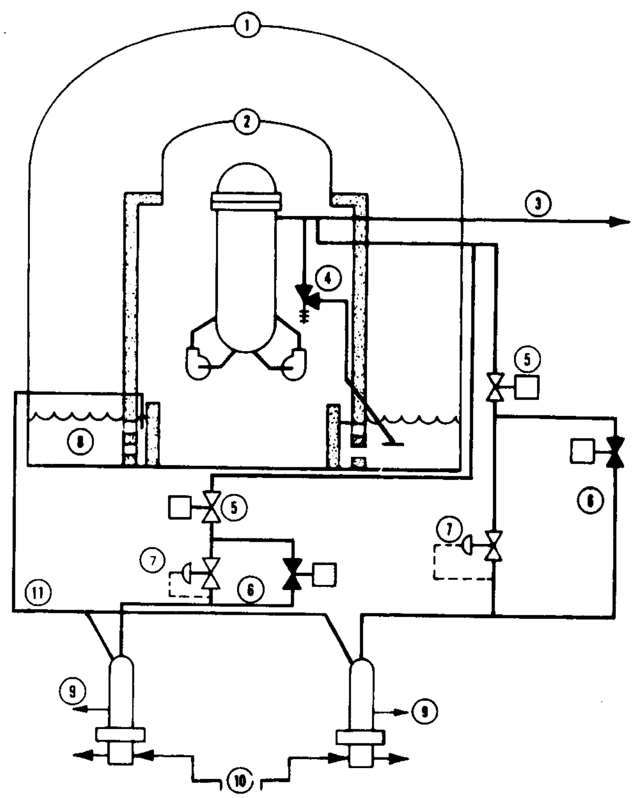
During nuclear boiler system isolation and in conjunction with the operation of the reactor core isolation cooling system and steam blowdown to the suppression pool, steam at reduced pressure and temperature is directed from the main steam lines to the residual

* This may be exceeded following blowdown in the event of a design basis accident.



1. CONTAINMENT
2. DRYWELL
3. RPV
4. SYSTEM PUMP
5. HEAT EXCHANGERS
6. SERVICE WATER

Figure 4-12. Residual Heat Removal System, Suppression Pool Cooling Function



1. CONTAINMENT
2. DRYWELL
3. MAIN STEAM LINE TO TURBINE
4. SAFETY/RELIEF
5. MANUAL REMOTE CONTROL VALVE
6. FULL-FLOW BYPASS
7. PRESSURE REDUCING SYSTEM
8. SUPPRESSION POOL
9. TO RCIC PUMP SUCTION
10. SERVICE WATER
11. VENT

Figure 4-13. Residual Heat Removal System, Steam Condensing Function heat removal system heat exchanger (see Figure 4-13). Condensate at a temperature not exceeding 140°F (60°C) is directed to the reactor core isolation system for return to the nuclear boiler system. Noncondensable gases from the heat exchangers are vented to the suppression pool. Steam condensing is manually initiated.

Nuclear Boiler Shutdown Cooling

The shutdown cooling function of the residual heat removal system (Figure 4-14) removes residual heat (decay heat and sensible heat) from the nuclear boiler system after reactor shutdown in preparation for refueling or nuclear system servicing. When the reactor vessel gage pressure is reduced to 135 psi (931 kPa) after shutdown, the shutdown cooling function is manually initiated by first draining the loops of water and flushing with condensate. [At reactor gage pressure above 135 psi (931 kPa), the low pressure coolant injection function of the residual heat removal system, which shares equipment and piping with the shutdown cooling function, remains functional.] The shutdown cooling function has the capability of reducing the reactor vessel to a temperature of 125°F (52°C), including draining and flushing, within approximately 20 hours after the control rods are inserted for shutdown and then to continue to reduce the temperature. Reactor water is taken from one of the reactor water recirculation loops, pumped through the heat exchanger, and return to the reactor vessel by way of the feedwater lines. Shutdown cooling is manually initiated.

During the operation of the shutdown cooling functions a portion of the flow is diverted to the reactor vessel head spray nozzle to condense the steam concentrated there.

Containment Spray Cooling

The containment spray cooling function (see Figure 4-15) of the residual heat removal system condenses and removes heat of any steam that bypasses the drywell, and prevents overpressurization of the containment. The suppression pool water is pumped from the pool through either or both of the same two completely independent loops, including pumps and heat exchanger. The containment spray function is manually initiated and terminated. Full spray flow is achieved within approximately 3 minutes of initiation.

Other Features

A flanged connection between the RHR system and the fuel building and containment pools cooling system can be made through use of a removable spool piece. This connection permits use of the RHR system to augment the cooling capability of the fuel building and containment pools cooling system, if required, during periods of larger than normal spent fuel storage.

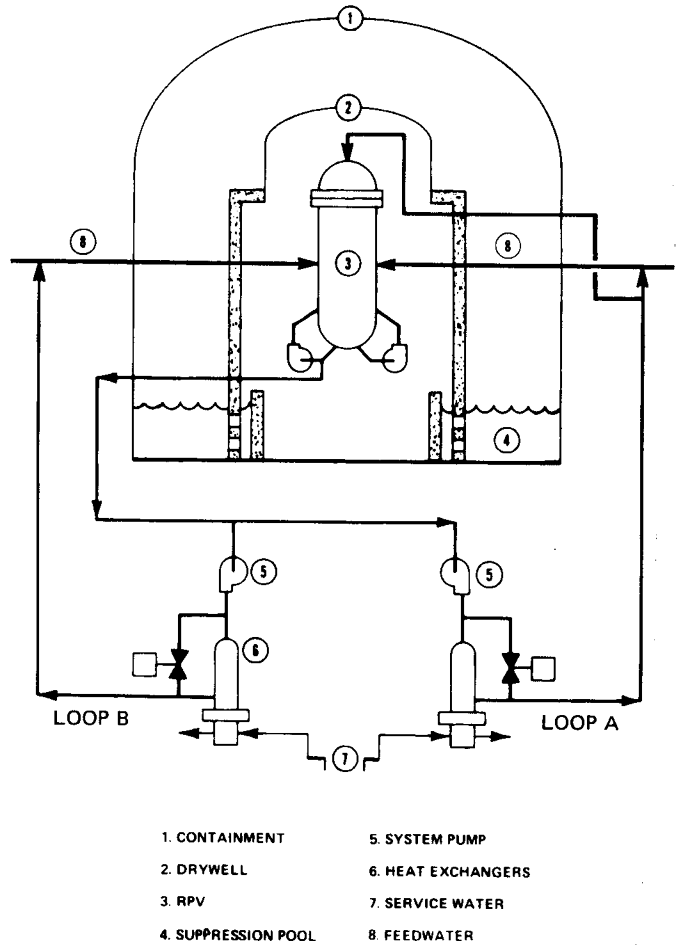


Figure 4-14. Residual Heat Removal System, Shutdown Cooling Function

A similar flange connection between the RHR system and the low pressure core spray system provides testing capability for the low pressure core spray system.

Double valve drain lines, between the containment isolation valves, provide a manual capability of checking for leaks from the nuclear boiler system.

The arrangement and location of equipment, piping, and instrumentation is designed to assure maximum operational availability by means of division into segregated groups of redundant equipment. Flow loops are thus separated and piping runs routed along structural

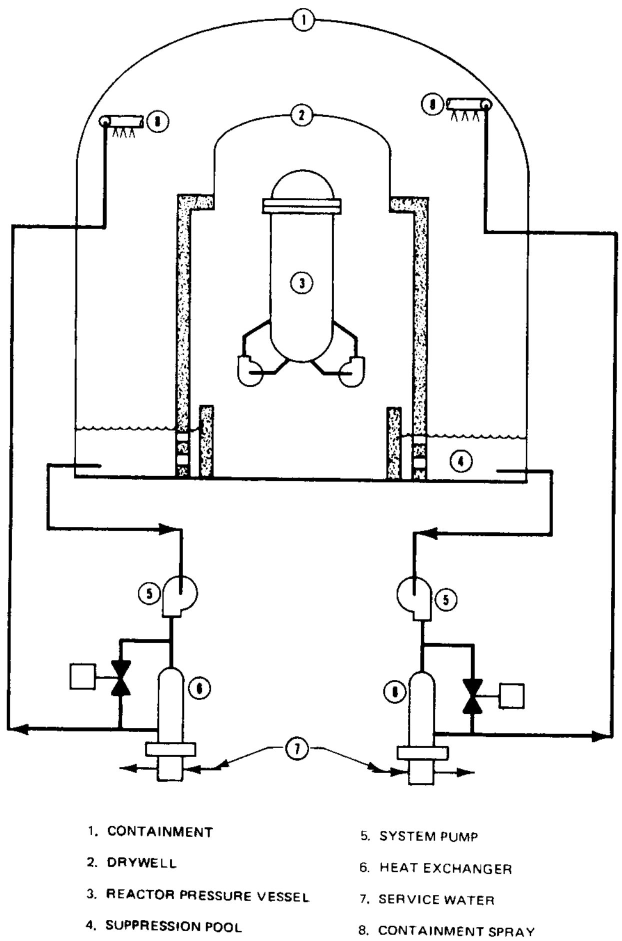


Figure 4-15. Residual Heat Removal System, Containment Spray Function

walls to provide protection against damage from missile-like fragments. The RHR service water pumps are located where adequate suction is assured at all operating conditions.

During fuel bundle movement above the core, after reactor pressure vessel head removal, reactor water is returned to the upper containment pool through distribution headers as an aid to visibility in the reactor well area.

SUMMARY

For ease of reference, a summary of the design and operating characteristics of the reactor auxiliaries is presented in the following tables. Table 4-1 is a summary of the system and equipment capacities given as a percentage of the design requirement. Table 4-2 is a summary of the mode of control for the various auxiliary systems.

Table 4-1
SUMMARY OF SYSTEM CAPACITIES

System	Process Capacity	Pumps	Heat Exchangers	Remarks
Reactor Water Cleanup System	100%	Two 50%	One 100% regenerative One 100% nonregenerative	Two 50% filter-demineralizer units
Fuel Building and Containment Pools Cooling and Cleanup System	100% cooling 100% filtering	Two 50% pumps	Two 50% cooling	Two 50% filter-demineralizer units. Additional cooling capacity by connection to RHR system
Closed Cooling Water System for Reactor Service	100%	100% plus one spare	100% plus one spare	
Standby Liquid Control System	100%	Two 100%	—	Positive displacement pumps; two 100% explosive valves and one 100% storage tank
Reactor Core Isolation Cooling System	100%	100%	—	Uses RHR system for cooling pressure suppression pool water
High Pressure Core Spray System	100%	100%	—	Uses RHR system for cooling pressure suppression pool water
Low Pressure Core Spray System	100%	100%	—	Uses RHR system for cooling pressure suppression pool water
Residual Heat Removal System				
Low Pressure Coolant Injection Function	>100%	Three 33%	—	The LPCS pump is a backup to this mode
Hot Standby Function	100%	—	Two at 50% (steam condensing)	Uses the RCIC turbine driven pump to return condensate to reactor vessel
Suppression Pool Cooling Function	200%	Two 100%	Two 100%	
Shutdown Cooling Function	100%	Two	Two 50%	
Containment Spray Function	200%	Two 100%	Two 100%	

Table 4-2
SUMMARY OF CONTROL MODES FOR AUXILIARY SYSTEMS

System	Type of Control	Remarks
Reactor Water Cleanup System	Manual*	Backwashing and resin changing from automatic cycle, manually initiated
Fuel Building and Containment Pool Cooling and Cleanup System	Manual*	Backwashing and precoating from automatic cycle, manually initiated (remote manual control from radwaste panel)
Reactor Building Closed Cooling Water System	Manual*	Local valve adjustment
Standby Liquid Control System	Manual*	Local motor test
Reactor Core Isolation Cooling System	Automatic Manual*	Low vessel water level
Automatic Depressurization	Automatic* Manual*	Two coincident signals (water level and drywell pressure) with time delay for manual override
High Pressure Core Spray System	Automatic Manual*	Low reactor water level, high containment pressure
Low Pressure Core Spray System	Automatic Manual*	Low reactor water level, high containment pressure
Residual Heat Removal System		
Low Pressure Coolant Injection Function	Automatic Manual*	Low reactor water level, plus high containment pressure
Hot Standby Function	Manual*	
Suppression Pool Cooling Function	Manual*	Emergency control at local panel
Shutdown Cooling Function	Manual*	
Containment Spray Function	Manual*	

* Manual is remote operator initiation from control room.

Section 5

Radioactive Waste Treatment

INTRODUCTION

The radioactive waste treatment system provides for the collection, processing and reclaiming of liquid waste which can be reused for plant services. The system also provides for the processing and packaging of radioactive solid waste preparatory to disposal. A decontaminating offgas treatment system provides for the low temperature holdup, filtering, and decay of radioactive gases which are continuously released from the main condenser by way of the steam jet injectors during plant operation. The guideline for processing of radioactive waste is to limit the release to the environment to a level "as low as reasonably achievable" in accordance with 10CFR50, Appendix I. This is several orders of magnitude below the NRC established limits in 10CFR20.

GASEOUS WASTE TREATMENT

There are two different types of radioactive gases produced in a light water reactor plant. First, there are activation gases. These are elements which become radioactive from exposure to radiation from the reactor core. The most significant of these gases are N-13, N-16, N-17, O-18, O-19, and F-18. Second, there are the fission gases which are released by fuel rod perforations, such as the radioisotopes of the noble gases krypton and xenon. The more fuel rod perforations the greater percentage of the offgas is fission gases.

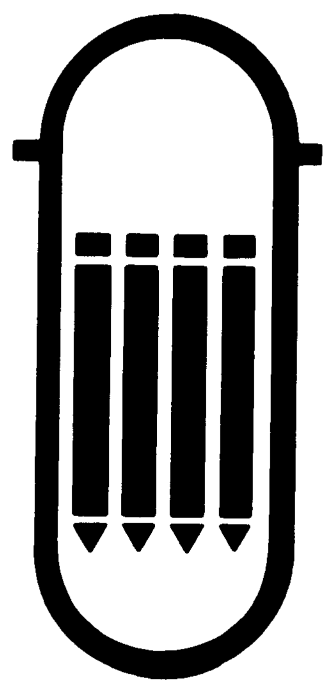
Gaseous Waste Release Paths

Several potential sources of radioactives gases in a BWR plant are:

- Steam jet air ejector offgases
- Turbine steam seal system offgases
- Mechanical vacuum pump exhaust
- Leaks from radioactive systems or components where leakage enters ventilation pathways.

Air Ejector Offgas Treatment — A characteristic of a boiling water type reactor is the release of dissolved gases in the primary coolant to the steam generated in the boiling process. These noncondensable gases are removed from the steam on a continuing basis by the condenser air ejector system. The BWR primary coolant and steam, in contrast to PWR's, are not subject to buildup of gas activity in their streams.

The gases in this system comprise air inleakage into the turbine-condenser, hydrogen and oxygen produced by radiolytic decomposition of the reactor water,



activation gases, and fission gases. The relative volume of the activation and fission gases is insignificant compared with the air and hydrogen-oxygen volumes.

The treatment of air ejector offgas reduces the activity level of the noncondensable fission gases removed from the main condenser prior to their release to the environs. The treatment utilizes the dynamic adsorption of radioactive noble gases on activated charcoal. The essential features of the low temperature off-gas system are shown schematically in Figure 5-1. Redundant equipment is used for gas processing except for the gas cooler charcoal adsorbers and post filter. These have inherent capabilities for continuing operation in the event of mechanical difficulties.

The low temperature offgas treatment system used skid-mounted equipment and a recombiner wherein several components are arranged in a common vessel. The gas reheater, recombiner catalyst, and gas condenser are in a common "recombiner" vessel to eliminate the interconnection of high temperature piping and valves. This also eliminates high piping stresses. Skid mounting substantially reduces design and construction effort required to place the equipment in the power plant. Less interconnecting piping and valves mean less construction effort.

Noncondensable gas removed from the main condenser, including air inleakage, is diluted with steam to less than 4% (by volume) hydrogen. The steam diluted offgas is passed through the recombiner vessel, superheated, catalytically recombined, and cooled to condense the moisture. The gas stream volume is thus reduced to that which is due to condenser air inleakage and traces of radioactive gases. This reduces the volume of charcoal required and combustibility of the gas. From the recombiner, the gases are cooled in a cooler-condenser to minimize the moisture content. The gases then pass through a desiccant dryer, a gas cooler, charcoal adsorbent beds, a HEPA filter and finally to the plant vent pipe. Two parallel desiccant dryers are used alternately to dry the gases. These dryers contain several hundred pounds of charcoal along with desiccant material which serves to remove short-lived krypton isotopes. The delay time is about 10 minutes. Daughter products are retained on the charcoal and desiccant material. The desiccant dryers are regenerated periodically by one of two installed regeneration systems. The gas cooler and charcoal adsorber vessels are located in a refrigerated vault that is maintained at a temperature of about 0°F (−17.8°C) by one of two refrigeration machines. Air from the refrigeration machine passes through one side of the gas cooler, thus cooling the incoming dried process gas. The cold

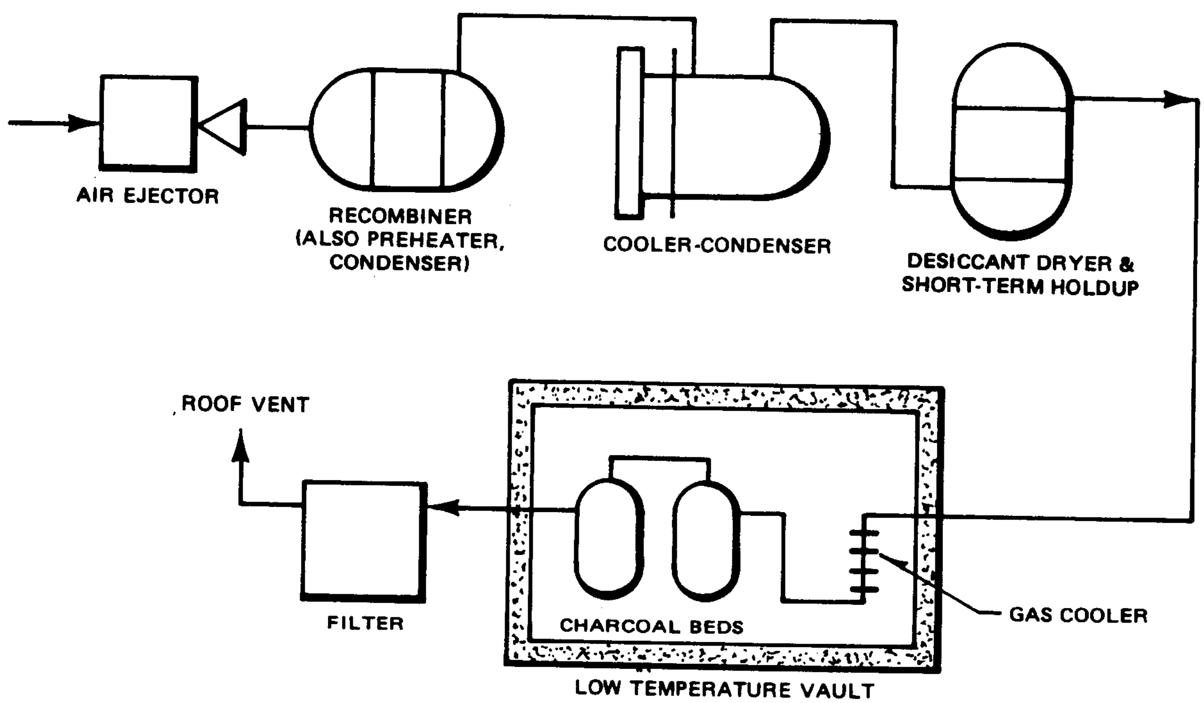


Figure 5-1. Low Temperature Off-Gas System

gas then enters the adsorber vessels which are arranged in series. These two vessels contain a total of 24 tons of charcoal and are dual pass types. The gases pass downward through the center cylindrical section and upward through an outer annular section. Offgas effluent from the adsorbers passes through a high efficiency filter before discharge.

Design retention time for noble gases in the charcoal beds, based on an air flow rate of 30 scfm ($51 m^3/hr$) is 46 hours for krypton and 42 days for xenon. This facilitates decay and removal of all the radioactive gaseous isotopes except Kr-85 and some Xe-133 and Xe-135. Biologically significant daughters such as SS-89, SR-90, Ba-140 and Cs-137 are retained in the charcoal and do not escape. The massive amount of charcoal also retains any iodine present in the ejector offgases so that iodine release by this path is essentially zero. The net output from this system is less than 50 μCi/sec of noble gases. This is based on a fuel leak rate which would result in a noble gas release rate of 100,000 μCi/sec measured at 30 minutes from the reactor. For a typical site, the radiation dose at the "worst" fence post would be less than 0.01 mRem/year which is well below Appendix I design objectives.

Steam Seal System Offgases — Turbine shaft steam seals may be sealed with primary steam or a steam source generated by vaporizing condensate. In the former case, radioactivity contained in the steam seal offgases is that carried by the primary steam. Since about 0.1% of the primary steam flow is used for shaft sealing, a corresponding amount of radioactivity would be present in the steam seal offgases, less the amount removed in the steam seal packing condenser. If a tall stack is used for the plant, discharge of these gases by way of a 2-minute offgas pipe may be used.

For plants using a roof vent, discharge of steam seal system offgases using only a holdup pipe approximately 2 minutes long would produce but a fraction of the design objective dose of Appendix I of 10CFR50. Therefore, the usual approach is to provide essentially non-radioactive steam for sealing the steam seals. The significant radioactivity escaping by way of the steam seal offgases is then only tritium and its contribution to environmental radiation dosage is minor.

In this system, condensate is evaporated to produce steam by use of a steam generator heated by primary steam. Condensate has a low radioactivity content and the decontamination factor achieved in steam production further reduces the radioactivity content by a factor of 10,000. The tritium present is not removed and so is present in the sealing steam. Most of the tritium stays in the plant, however, since the sealing steam is condensed in the steam seal condenser and returned to the main condenser hotwell.

Condensate is used as a sealing steam source rather than demineralized water so that it can be recycled. If demineralized water was used, this would represent an input of "new" water and an equivalent amount of water would have to be discharged by way of radwaste. This would create an unnecessary load on the radwaste system and cause an unnecessary discharge to the environment.

Mechanical Vacuum Pump Operation — During the startups of a plant and before operation of the steam jet air ejector is initiated, a mechanical vacuum pump is used for gas evacuation of the main condenser.

The initial rate of removal of radioactive noble gases by the vacuum pump may be significant (depending upon prior fuel history) but decreases rapidly as a vacuum is attained. The residual gases — primarily xenon from iodine decay during shutdown — pumped from the condenser during a startup produce a relatively small dose rate to the nearest neighbors, about 0.2 mRem per year or less.

LIQUID WASTE TREATMENT

Sources of radioactivity in the water and steam are produced from reactor coolant activation, corrosion product activation, and fission products arising from fuel leaks. Radioactive materials may be present in both soluble and insoluble forms. Radioactive liquids accumulated in the waste collection facilities originate principally from various controlled drains in the station, backwashing of filter-demineralizers, regeneration of deepbed demineralizers (if present), and the infrequent chemical decontamination of various pieces of primary system equipment. Other various sources of radioactive liquid wastes are leaks from plant equipment, laboratory and laundry drains, and floor drains.

The Process

The basic functions are the collection of wastes, treatment, storage, and measurement to determine quality for disposition or further treatment. The processes and instrumentation used provide complete control at all times. The radioactive waste treatment system process flow diagram is shown in Figure 5-2. A single treatment system is used to collect, treat, and dispose of radioactive and potentially radioactive effluents or solid wastes in an economical manner without limiting station operation and availability, while staying well within the criteria for environmental disposal.

The treatment of liquid waste has the objective of the elimination of liquid process effluents, treatment of liquids to allow reuse in plant systems and the removal of radioactivity in controlled liquid discharges to meet Appendix I.

To accommodate the range of environmental conditions and dilution water availability that arise in the siting of nuclear power stations, the liquid waste treatment system has the capability of "zero liquid release" under common station operating conditions. The use of the detergent treatment and excess water subsystems provides this capability. The various liquid process discharges are collected as batches and treated as necessary to achieve the desired condition for processing. However, processed wastes are under control of redundant in-line instrumentation and are normally discharged to their in-plant destination without further batch collection and sampling. Any processed waste which would be discharged off site is collected, sampled, and analyzed as a batch prior to discharge. Wastes discharged off site would meet the dose objectives of 10CFR50, Appendix I. To demonstrate compliance, environmental program sampling and analysis of mud and marine life near the plant provide data about environmental concentrations.

The radwaste treatment system can be used in plants having condensate treatment systems using either regenerated deep bed units or filter-demineralizer units which do not regenerate their resin.

For a plant cooled by salt or brackish water, a condensate system using regenerated, high flow rate, deep-bed demineralizers should be used to prevent entry of salt water into the reactor during periods of condenser leaks. An ultrasonic resin cleaner is used with resins which minimizes wastes resulting from chemical regeneration of the resins.

Radioactive liquid wastes are collected as three classes: high purity, low purity and chemical, and detergent or laundry wastes. These are collected by suitable connection of drains, piping, and tanks or sumps in the various plant buildings to the collection tanks in the radwaste building. Oil is removed from sump and equipment drains by means of oil separators on these streams ahead of the collection tanks. Normally, the high-purity waste is filtered, demineralized, monitored, and returned to condensate storage for further use. The low-purity and chemical wastes are batch treated by a chemical neutralization process and are then concentrated by evaporation for packaging as solid waste for off-site shipment. Water boiled off in the concentrator is processed through a demineralizer, monitored, and returned to condensate storage. Water returned to condensate storage is of makeup water quality. Laundry and detergent wastes are either filtered, sampled and discharged to the environment as batches by way of the discharge canal or evaporated in the detergent evaporator. If evaporated, the vapor is not condensed but is discharged to the atmosphere by way of a vent pipe.

An excess water system is used to accommodate water input in excess of the station inventory. A large water input may usually be caused by leakage from the main condenser or service water heat exchanger tubes. This system is comprised of water storage tanks to receive water in excess of condensate storage capacity, as space permits; water can be sent from this tank to condensate storage for reuse. Alternately, this water can be vaporized using the detergent waste evaporator. Radioactive vapor release from this source is a small fraction of the release of radioactive materials via ventilation air.

The combination of this system, the detergent waste treatment system, and the treatment and reuse of high purity and floor drain-chemical wastes accomplishes "zero liquid release" of liquid radioactive wastes.

If treated effluent is to be discharged from the site, it is discharged only after batch sampling and analysis. Such batches are injected into the cooling water discharge canal at a rate such that the maximum permissible concentrations for an unidentified mixture are not exceeded. Annual discharges meet the guidelines of Appendix I of 10CFR50. Actual experience has shown that the use of the limit based on an unidentified mixture is conservative when compared with the permissible limits based on determination of the actual isotopes that compose the discharge activity.

Control and Instrumentation

Overall control of the radwaste treatment is exercised from a local control room situated in the radwaste building. A central panel in this room contains instruments, controls and alarms for the operation of the radwaste treatment process. This panel also carries indicating lights to show whether or not the various sump pumps which transfer effluents from the other station buildings to the radwaste plant are running.

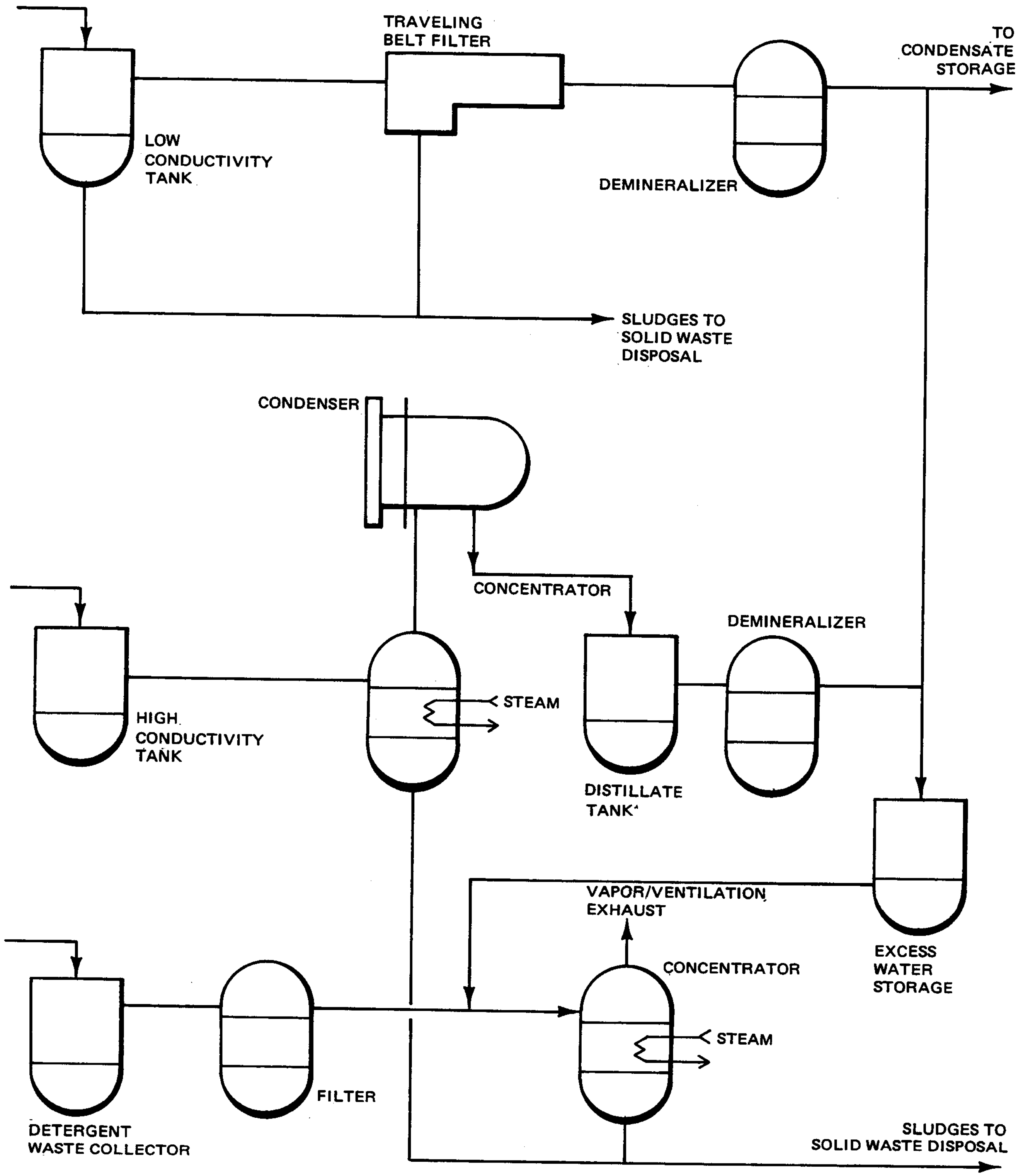


Figure 5-2. Simplified Radwaste Flow Diagram

Radwaste system alarm signals are also received in the main control room as a common trouble alarm.

Sampling

Sampling lines are used to sample the various collector tanks, and to monitor and control the process. Sample lines, vented to a central sampling station, provide sample flows for continuous monitoring instrumentation (conductivity, turbidity, radioactivity) and also provide for grab samples. These are taken to the station laboratory for analysis. Samples of circulating water are also taken at the intake to the station and in the circulating water discharge to audit the background and discharge levels of radioactivity as necessary.

Ventilation Sources — During all phases of plant operation, various portions of the containment, turbine and auxiliary buildings are provided with fresh air. This ventilation air is exhausted to the atmosphere via building vents and/or a stack. It is required that less than 50 μCi/sec of radiogases and 0.001 to 0.01 μCi/sec of I-131 are released via this pathway. These small releases are estimated to contribute a whole body dose of less than 0.1 mRem/yr to the nearest neighbor and a thyroid dose in the range of a few mRem/yr if the milk-exposure pathway exists in the immediate off-site environment. These ventilation sources do not require treatment to meet the proposed guidelines of 10CFR50, Appendix I.

Offgas Release Monitoring — Air ejector offgas release rates are continually monitored and recorded by duplicate, monitor-recorders after about 2 minutes of delay.

Duplicate, continuous monitors near the release point of the offgases record the gross activity release rate for radiogases. Particulate and iodine samplers also permit monitoring of any significant release of these species, and gas samples are taken for isotopic analysis.

To obtain background data on radiation levels and meteorological information, an environs and plant site monitoring program is instituted in advance of plant startup. Environmental monitoring is normally continued during plant operation to verify the information provided by in-plant monitors and to enable actual release to be related to actual environs radiation dose. Environs monitoring is usually performed with dosimeters supplemented by periodic surveys and sampling of appropriate natural materials from the region. Further description of the offgas monitoring system is provided in Section 6.

SOLID WASTE DISPOSAL

Source and Types of Solid Wastes

The following are typical of potentially radioactive solid wastes:

- Filter and resin sludges
- Concentrated wastes from liquid waste treatment concentrators
- Air filters from offgas and ventilation systems
- Solid laboratory wastes
- Contaminated clothing, tools and small pieces of equipment which cannot be economically decontaminated
- Miscellaneous paper, rags, etc., from contaminated areas
- Used reactor equipment such as spent control rod blades, fuel channels and in-core ion chambers

Waste Resulting from Process

The sludges result from filters and filter-demineralizers in the reactor water cleanup, condensate treatment (if a filter-demineralizer type), fuel building and containment pool cooling and cleanup, and in the radwaste systems. Reactor water cleanup sludges are generally kept separate because of the usually higher radiation levels of the resultant solid wastes. This minimizes the shielding required for off-site shipment of the other solid wastes.

Reactor water cleanup filter-demineralizer backwashes are collected in backwash receivers located beneath the cleanup filter-demineralizers. The backwash slurry is pumped to one of two cleanup phase separators located in the radwaste facility. After settling, water is decanted from the phase separator and sent to low conductivity tanks. The remaining sludge is accumulated for radioactive decay prior to further processing as solid waste. After accumulation in a phase separator for a predetermined time, input is switched to the other phase separator. Decay occurs in the first phase separator for about 60 days until accumulation in the second phase separator is to be stopped. At this time, the decayed slurry is filtered through the traveling belt filter (TBF) and the water proceeds to the waste demineralizer and condensate storage. The dewatered sludge goes to the solid waste system for packaging as a solid waste.

In plants having condensate treatment filter-demineralizers, they are backwashed to a backwash receiver tank located beneath these units. Each backwash batch is then routed to one of a pair of condensate phase separators located in the radwaste facility. Decantate is periodically routed to the low conductivity tanks. Sludges are accumulated in the phase separators for accumulation prior to further processing as solid wastes as with the cleanup system sludges.

Fuel building and containment pool filter-demineralizers are backwashed to a backwash receiver located beneath the filter-demineralizers and, periodically, this slurry is pumped to the low conductivity tanks for processing.

The radwaste filters are traveling belt (flat bed) type precoat filters which discharge a solid cake to the solid waste system.

The sludges resulting from filter-demineralizer backwashing are normally routed periodically, on a programmed basis, from the phase separators to a TBF as described earlier. The TBF dewaters the slurry to produce a moist solid which has no free water present. The solids are discharged to the solid waste system.

Spent demineralizer resins from the waste demineralizers (and condensate demineralizers if condensate treatment is a deep-bed type) are sent as a resin-water slurry to a spent resin tank as batches. The resins slurry is then routed to the TBF for dewatering. The resins go to the solid waste system and the filtrate is demineralized and returned to condensate storage.

Concentrated wastes from the waste concentrator are collected in a concentrated waste tank. Periodically, these are pumped in measured amounts to the solid waste system.

The solid waste system is shown in Figure 5-3. This system uses large containers which are 6 feet (1.8m) in diameter by 6 feet (1.8m) high, carbon steel fitted with a disposable mixer. The system is remotely operated using closed circuit television to view the various operations.

The solid waste portion of the radwaste building contains a track entrance and loading area, a shielded storage area, and a container filling/capping area, all serviced by an overhead bridge crane. Empty containers are stored outside the radwaste building and are brought in as necessary.

The empty container is picked up by the bridge crane and moved to a transfer car. The transfer car moves the container and locates it under the mixer/filling station. This station contains the fill pipes for solid waste, cement, and concentrated waste (or water). In addition, there is a mixer motor, the positioning mechanism, and a vent pipe leading to a dust collector.

After the mixer/filling station is engaged with the container and the mixer, solid wastes are conveyed from one or the other of the traveling belt filters to the mixer/filling station by a screw conveyor. Normally, it will receive dewatered, moist filter sludges. Over a period of several days, when the weighing scale on the transfer car indicates receipt of a predetermined amount of sludge, filter operations are stopped. (High container radiation can also be used to indicate sufficient solid wastes have been added.) Next, a measured amount of concentrated waste (or water) is added and the mixer is started. A measured amount of Portland cement is added. After the contents are well mixed, the mixer is stopped. The mixer/filling station assembly is then remotely disengaged from the container and the container is moved by the transfer car to a capping station. After capping, the container is moved for remote pickup by the bridge crane which delivers the filled container to shielded storage.

When shipping is desired, a flat-bed truck is driven inside the building. Depending upon the radiation dose measured at the mixing/filling station, a shipping cask for shielding during shipment may be required. Either a steel or lead type may be used, and after selection, it is placed on the truck bed. The waste container is placed inside the cask by remote control and the cask lid fastened. The waste container thus enclosed and shielded is ready for shipment to a licensed burial ground.

If spent resins or filter sludges from phase separators are to be put into containers, a traveling belt filter is operated to dewater these wastes. After appropriate quantities are added to the container, concentrated waste, or water, and cement are measured in as previously described.

While the system described includes solidification by the addition of Portland cement, the solidification step may be omitted as dewatered resins and sludges may be shipped alone. However, concentrated wastes will always need the addition of cement so no free water is present.

The system is so arranged and engineered that normal maintenance will occur when no waste container is present. Means are available for safely removing containers to shielded storage in the event of equipment malfunction during any of the process steps.

Dust from the container filling operation is removed and contained by a dust collector. Adequate ventilation minimizes any solid waste contamination in the radwaste system.

Other Solid Wastes

The general plan for handling all low activity level solid wastes is to store temporarily on site in fiber cartons or steel drums (shielded as necessary). Ultimate disposal is by shipment to off-site storage.

Equipment too large to be accommodated in this manner is handled as a special case. Since the frequency and need for the handling of large equipment is quite infrequent, suitable procedures for decontamination, shielding, shipment, and storge of such items are developed as necessary.

Used reactor equipment to be disposed of is first stored for sufficient time in the fuel storage pool to obtain decay of the short-lived isotopes before removal for off-site shipment. Normally, such equipment requires shielding when removed from the fuel pool.

The activity of most other categories of dry solid wastes is low enough to permit handling of packages by contact. These wastes are collected in containers located in appropriate zones around the plant, as dictated by the volumes of wastes generated during operation and maintenance. The containers (fiber, drums, cartons, or boxes) are monitored periodically during filling so that the contents do not exceed a practical maximum before disposal (50 to 100 mRem/hr at the surface would be a usual maximum). The containers are then sealed and moved to a controlled-access, enclosed storage area for temporary storage. Compressible wastes are compacted into drums by a hydraulic press-baling machine to reduce their volume and then stored temporarily. Ventilation and filtration are used to maintain control of contaminated particles when operating packaging equipment. Compacted and noncompressible waste packages are eventually shipped to an approved off-site facility for storage.

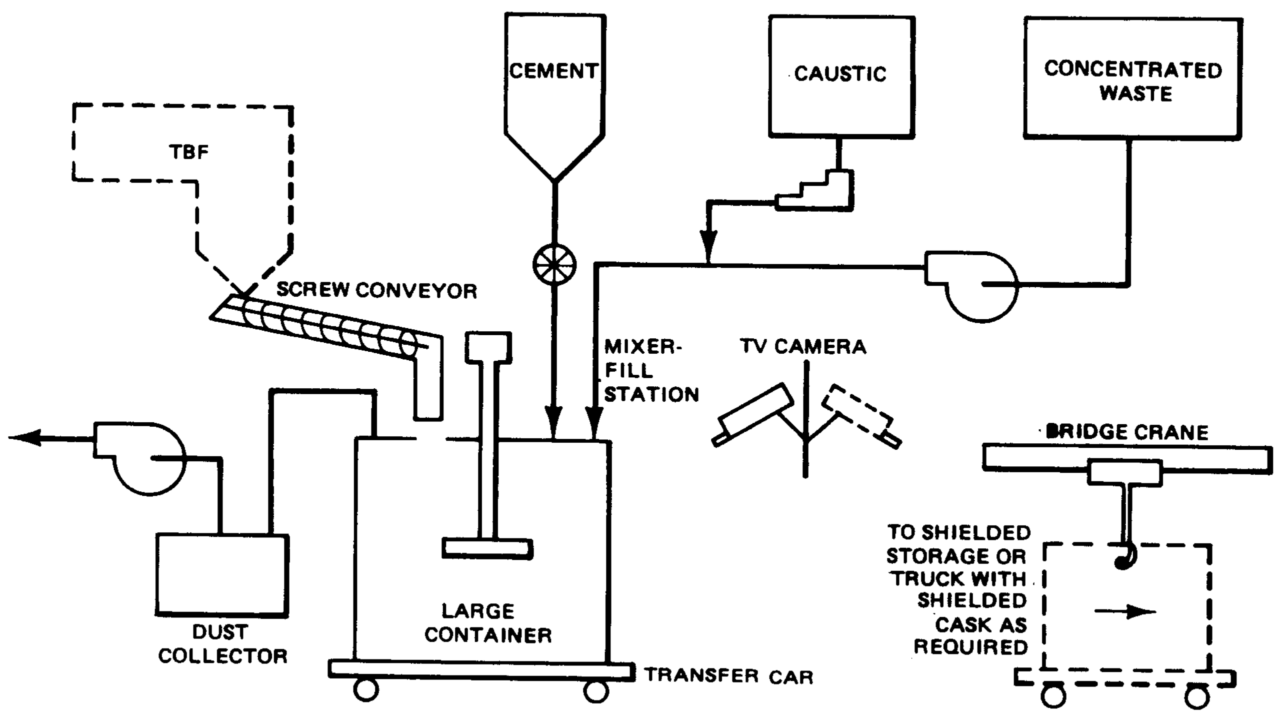


Figure 5-3. Simplified Solid Radwaste Flow Diagram

Section 6 Instrumentation and Controls

INTRODUCTION

The instrumentation of the boiling water reactor is generally associated with the control of the reactor, the prevention of the operation of the plant under unsafe or potentially unsafe conditions, the monitoring of process fluids and gases, and for monitoring of the performance of the plant. The control of the plant is from the control room. Instrumentation for monitoring the performance of the plant is located in the control room and locally. A typical control room console is shown in Figure 6-1.

Power output from the boiling water reactor is controlled by either changes in reactor water recirculation flow rate or by the moving of control rods. As the reactor power output changes, the turbine initial pressure regulator adjusts the turbine admission valve to maintain nearly constant reactor pressure, admitting the new steam flow to produce the desired change in the turbine-generator power output. The boiling water reactor is operated at constant reactor pressure because pressure changes caused by turbine throttle operation in response to load changes tend to bring about reactor power changes opposite to the desired change. However, small controlled pressure changes are used to improve load response.

PLANT STARTUP

Startup of the plant from a cold standby condition to a power producing condition requires:

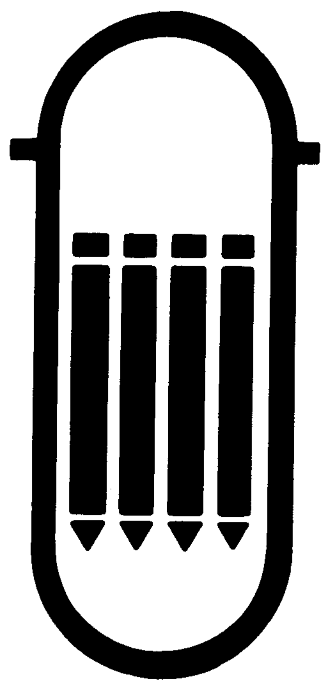
- the startup of the reactor water recirculation pumps,
- the pumps brought to rated speed,
- the manipulation of the reactor water recirculation, flow control valves to provide the required flow,
- the movement of control rods to attain the desired power level, and
- the monitoring of the reactor to record and monitor reactor behavior.

The startup of the plant from a cold standby condition is manually controlled by the operator.

Reactor Startup and Operation

The operational sequence for the startup of the plant from a cold standby condition is as follows:

- The flow control valves are set at the minimum position which corresponds to approximately 25% of rated flow.



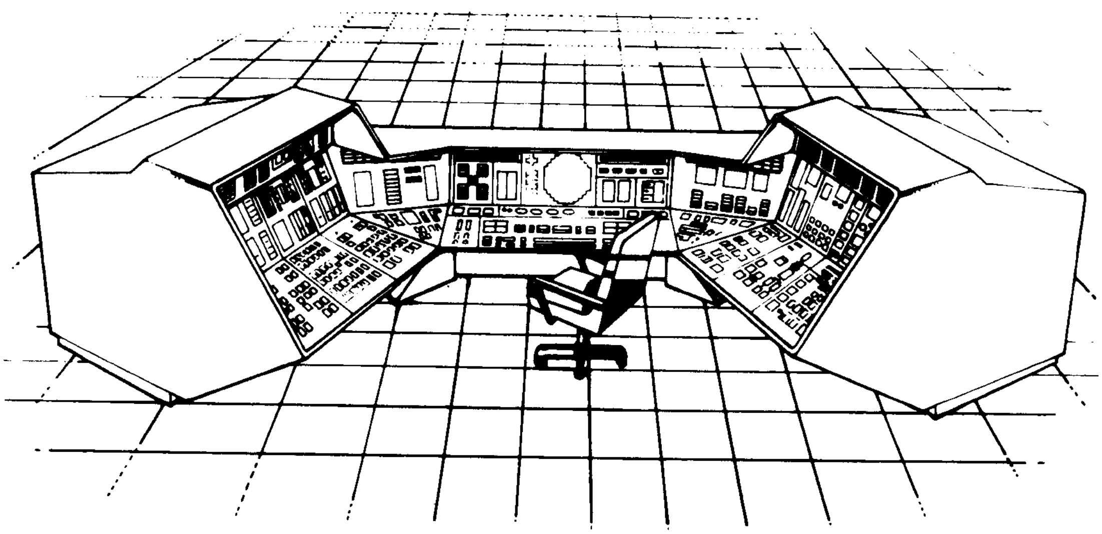


Figure 6-1. Control Room Benchboard

- The reactor water recirculation pumps are started. (Because the low frequency motor-generator sets cannot start the recirculation pump motors, the pump motors are started from auxiliary power and transferred to the low frequency motor-generator sets when the pump motor nears full speed and after the starting current has dropped).
- Control rods are manually withdrawn according to a predetermined schedule to achieve criticality of the reactor. They are further withdrawn to approximately 32% of rated power with the reactor water flow control valves fully open and the recirculation pumps operating at low speed (25%). (The rate at which power level is raised is usually limited by conditions of thermal expansion of the reactor vessel.)
- At approximately 32% of rated power, the reactor water flow control valves are closed and the recirculation pump transferred to auxiliary power and operated at rated speed.
- From approximately 30% to approximately 40% of rated power, the control of power level is by manual control of recirculation flow by changes in control valve position from minimum position.
- Above approximately 65% of rated core flow, the recirculation flow control is automatic.
- Between approximately 38% and approximately 75% of rated power, control rods are normally used to change power level.
- Above approximately 75% of rated power, change in reactor water recirculation flow is normally used to change power level.
- Neutron monitoring channels monitor the nuclear behavior of the reactor. (Counting channels are used in the subcritical range up through criticality. The intermediate range, from criticality to the power range, is monitored by the neutron counting channels and/or intermediate range monitoring channels. The power range neutron monitors are used throughout the power range usually above 21% of rated power.)

During initial power operation, an operating curve is established relating reactor power to recirculation flow. The first point of the curve is full flow and rated power. When a rod pattern is established for this point, recirculation flow is reduced in steps at the same rod pattern, and the relationship of flow to power is plotted. Other curves are established at lower power ratings and other rod patterns as desired. During operation, reactor power may be changed by flow control adjustment, rod positioning, or a combination of the two, while adhering to established operating curves. A rod withdrawal interlock is used to prevent unscheduled rod withdrawal which would result in an excessive power-to-flow ratio. The operating curves are evaluated periodically, usually during startups, to compensate for changing reactivity coefficients. Although control rod movement is not required when the load is changed by recirculation flow adjustment, long-term transient reactivity effects are normally compensated for by control rod adjustment.

Turbine Startup

While the reactor temperature is being increased, the turbine is rotated by the turning gear. When reactor steam is available, the shaft seal steam is applied and the mechanical vacuum pump is started. After a partial vacuum is established in the main condenser, heating of the turbine and steam flow from the reactor are accomplished by first establishing a flow of steam to the condenser through the bypass valves. This flow is gradually transferred to the turbine until rated speed is achieved after which the unit is synchronized with the system. The bypass flow is controlled by the initial pressure regulator during this initial period and the turbine is controlled by the governor. The initial pressure regulator assumes normal control of the turbine admission valve after the unit is synchronized and a small amount of load is applied.

POWER OPERATION

After the generator is synchronized to the electrical system and is producing a substantial output, the power output is adjusted to meet the system requirements by manual adjustment of control rods, manual or automatic adjustment of reactor recirculation flow, or a combination of these two methods.

Control Rod Adjustment

Withdrawing a control rod reduces the neutron absorption and increases core reactivity. Reactor power then increases until the increased steam formation just balances the change in reactivity caused by the rod withdrawal. The increase in boiling rate tends to raise reactor pressure, causing the initial pressure regulator to open the turbine admission valves sufficiently to maintain a constant pressure. When a control rod is inserted, the converse effect occurs.

The rate of power increase is limited to the rate at which control rods can be withdrawn. Control rods can be operated one at a time, or in groups of four rods in a symmetrical pattern. Single rods or rod groups can be withdrawn continuously or in incremental steps (notch steps). Continuous movement is usually limited to subcritical and heatup conditions. Control rod movement is the normal method of making large changes in reactor power, such as daily or weekly load shifts requiring reduction and increases of more than 25% of rated power.

Recirculation Flow Control

The BWR is unique in that reactor power output can be varied over a power range of approximately 25% of the operating power level by adjustment of the reactor recirculation flow without any movement of control rods. This is the normal method used for load following and maneuvering the reactor and allows for load following at rates of up to 1% of rated power per second.

Reactor power change is accomplished by using the negative power coefficient. An increase in recirculation flow temporarily reduces the volume of steam in the core by removing the steam voids at a faster rate. This increases the reactivity of the core which causes the reactor power level to increase. The increased steam generation rate increases the steam volume in the core with a consequent negative reactivity effect, and a new constant power level is established. When recirculation flow is reduced, the power level is reduced in a similar manner.

Reactor — Turbine Control

A schematic diagram of the flow control system is shown in Figure 6-2. The adjustment of the flow control valve changes the recirculation flow rate. To change reactor power, a demand signal from the operator or a load/speed error signal from the speed governing mechanism is supplied to the master controller. A signal from the master controller adjusts the position setting of the controller for each valve. This signal is compared with the actual position of the valve associated with each controller. The resulting error signal causes adjustment of the valve position to reduce the error signal to zero. The reactor power change resulting from the change in recirculation flow causes the initial pressure regulator to reposition the turbine control valves.

Automatic load control is accomplished by supplying a speed-load error signal from the turbine governor to the master controller. The energy storage capability of the water in the reactor system is used to increase the speed of response of the automatic load control system. An automatic, temporary change in the set point of the pressure regulator is produced when there is a demand for a change in turbine output. If an increase in load is demanded, the pressure set point is lowered and water in the reactor system flashes to produce extra steam flow to the turbine. If a decrease in load is demanded, the pressure set point is raised which causes the turbine control valve to move toward the closed position.

System Control

Control signals between the reactor and the turbine provide two functions required for normal operation. A signal from the initial pressure regulator is provided to the turbine admission valves to maintain a nearly con-

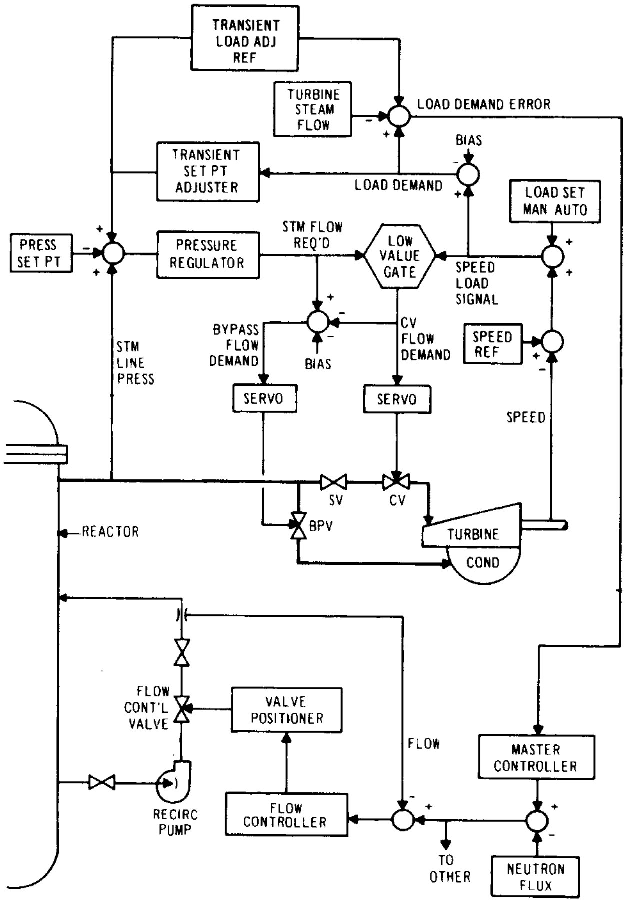


Figure 6-2. Reactor — Turbine Control

stant reactor pressure. A signal from the speed-load governing mechanism to the master flow controller establishes the necessary reactor recirculation flow required to meet the system power requirements.

If, while under normal load, the turbine speed decreases or the speed-load changer setting is increased, a positive speed-load signal is transmitted to the initial pressure regulator and the master flow controller. The increase in signal causes a momentary decrease in the pressure setting of the initial pressure regulator and causes the master controller to increase the flow demand to the recirculation system flow valve controller.

Decreasing the pressure setting of the initial pressure regulator causes a signal to be sent to the turbine admission valves, instructing them to open rapidly by an amount and for a length of time which is a function of the speed-load error. This gives a limited rapid initial response to a speed-load error by increasing the steam flow from the reactor vessel. The allowable duration of this transient increased steam flow is limited by the fact that increased steam flow tends to reduce the reactor pressure and power level.

The increased flow demand to the recirculation system flow valve controller causes the flow control valve to open wider, causing an increase in reactor recirculation flow. The increased flow increases the reactor power output by sweeping out steam bubbles from the core faster, thus raising the effective density of the moderator. The increased steaming rate causes a slight increase in reactor pressure. The increase of pressure is sensed by the initial pressure regulator, which sends a signal instructing the turbine admission valves to open sufficiently to increase the turbine output to a level that will cancel the speed-load error.

Daily Load Following — Essentially any practical daily load following profile can be followed. There are no restrictions due to spatial xenon oscillations. (Early ascent above 95% of rated power would be subject to xenon override considerations.) Power levels can be readily reduced to any level during daily load following, including the power level where the turbine-generator is supplying only house loads. Automatic load following provides the capability to accept large changes in load demand at operating power levels. The change in load demand may be initiated at any power level or reactor water recirculation flow combination in the automatic flow control range. This region lies between 28% and 75% of rated power and core flow rate ranging from approximately 65% to approximately 68% (constant flow control valve setting) and between 40% and rated power at rated core flow. For load reduction demands that exceed the range of the automatic flow control system, the main steam bypass system provides additional capability up to the bypass system capacity. The reactor operator would then establish a new control rod configuration to match the new power demand. For load increase demands that exceed the range of the automatic flow control system (assuming reduced flow initially), the power level will rise to that level corresponding to rated core flow and remain there until the control rods can be adjusted to increase power up to the desired level.

Step demands for up to 25% of the power at rated core flow are accommodated by automatic reactor water recirculation flow control.

Automatic Dispatch Operation — Automatic reactor water recirculation flow control in combination with ganged control rods allows full participation in an automatic dispatch system with the combined purpose of meeting tie line regulation, spinning reserve, grid load rejection, and daily load following requirements. During such operation, automatic reactor water recirculation flow control meets the rapid changes in load demand required by tie line regulation, while at the same time providing margins for spinning reserve or grid load rejection. The Unit operator would adjust control rods to preserve the desired automatic margins during the slower changes in base power level required by daily load following.

Turbine Bypass Valve

A fast response, modulating-type valve, controlled by the steam bypass pressure regulator system, is used to perform three basic functions. The primary function is to reduce the rate of rise of reactor pressure when the turbine admission valves are moved rapidly in the closing direction. To perform this function, the bypass valve needs about the same speed of response as the turbine admission valves to prevent a pressure-induced reactor scram from high neutron flux when the turbine load is suddenly reduced by partial or complete closure of the turbine admission valves.

The second function of the bypass valve is to control reactor pressure during startup of the turbine. This allows the reactor power level to be held constant while the turbine steam flow is varied as the turbine is brought up to speed under the control of its speed governor.

The third function of the bypass valve is to help control reactor pressure after the turbine has been tripped. It is used to discharge the decay heat to the condenser and to control the rate of cooling of the reactor system.

Pressure Relief Function

A pressure relief function is used to control large pressure transients. This system will operate safety/relief valves following closure of the main steam isolation valves or the sudden closure of the turbine admission or stop valves and failure of the turbine bypass system to relieve the excess pressure. For this function, the safety/relief valves discharge steam from the steam lines inside the drywell to the suppression chamber. Each safety/relief valve is operated from its own overpressure signal for the relief function, and by direct spring action for the safety function.

To limit the cycling of safety/relief valves to one valve subsequent to their initial actuation during a main steam line isolation event, two valves (one a backup to the other) have the feature of automatically changing normal set pressures (opening and closing) following their initial actuation at normal set pressures to a lower level, thereby limiting the pressure cycles to a level where the other relief valves will not reopen. In conjunction with these two valves, the set pressure for the closing of the other valves is changed automatically which allows for them to stay open longer before closing to accommodate pressure swings. Manual valve operation and resetting of valve set pressure to their normal levels following the transient is by the control room operator.

Reactor Feedwater Control System

The reactor feedwater control system automatically controls the flow of reactor feedwater into the reactor vessel to maintain the water in the vessel within predetermined levels during all modes of plant operation. The control system utilizes signals from reactor vessel water level, steam flow, and feedwater flow.

The reactor feedwater control system provides the signal for the reduction of reactor water recirculation flow to accommodate reduced feedwater flow caused by failure of a singe feedwater pump.

PLANT SHUTDOWN

For normal plant shutdown, reactor power and plant output are reduced by manual insertion of control rods. After turbine load is reduced to a minimum value, steam flow is maintained through the bypass valve and the generator is disconnected from the system. Reactor power is further reduced to a low level and the decay heat is rejected to the condenser through the turbine bypass valve. If the reactor is to be kept in the hot standby or steam condensing condition, criticality is maintained but fission power is reduced to a low level (about 0.01% of rated power is sufficient to maintain operating temperature). If refueling or other functions requiring access to the vessel are planned, all control rods are inserted and the reactor is cooled down by release of steam to the main condenser. The rate of cooldown is normally controlled by periodically lowering the setting of the initial pressure regulator. After vessel gage pressure has been reduced sufficiently [135 psi (930 kPa)], the heat sink can be switched from the main condenser to the Residual Heat Removal System heat exchangers in order to get the reactor to the cold shutdown condition.

REACTOR INSTRUMENTATION
NEUTRON MONITORING SYSTEM

Reactor power is monitored from the source range up through the power operating range by suitable neutron monitoring channels, with all detectors inside the reactor core. This location of detectors provides maximum sensitivity to control rod movement during the startup period and provides optimum monitoring in the intermediate and power ranges. Three types of neutron monitoring are used: source range counting; intermediate range, voltage variance method; and local power range, d-c ion chambers (see Figure 6-3). A traversing in-core probe system provides for periodic calibration of the neutron detectors.

Source Range Monitor (SRM)

In the source range, the neutron flux is monitored by fission counters which are inserted to about the midplane of the core by the drive mechanisms (Figure 6-4) which move each chamber into the core through inverted thimbles. A range from below the source level to 10^9 nv is covered.

As startup progresses and the count rate approaches the top of the meter range (about 10^6 cps), the counters are withdrawn downward to give a drop in apparent count rate. Criticality normally occurs before movement of the counters is necessary. The counters can be motor driven to any position within their limits of travel (see Figure 6-4); however, two or three selected positions will provide the necessary range to achieve criticality and provide overlap with the intermediate range monitors.

When the reactor reaches the power range, the counters are moved to a position approximately 2 feet (0.61m) below the core. This places the counters in a low neutron flux so that burnup and activation of the counters are minimized.

Intermediate Range Monitor (IRM)

The intermediate range is from about 10^8 to 1.5×10^{13} nv. In this range, the neutron flux is monitored by a system using a voltage variance method (also known as MSV or Campbell method). This method makes use of the a-c component of voltage which is due to the random nature of neutron pulses generated in a detection chamber. With small chambers located in the high temperature ambient of the reactor core, the a-c component is used to measure neutron flux at lower power levels because cable leakage and gamma radiation have relatively little effect on the signal.

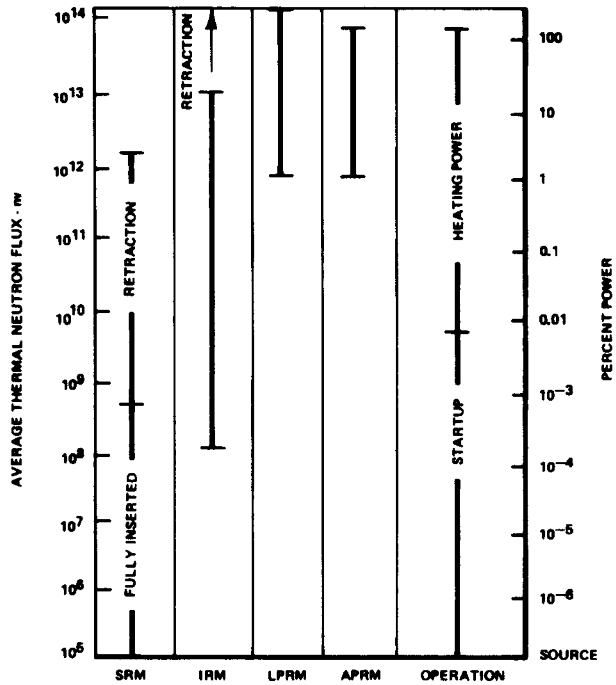


Figure 6-3. Ranges of Neutron Monitoring System

These fission chambers are also withdrawn during full power operation to maintain their expected life and to reduce activation. They are positioned with drive mechanisms similar to those used for the source range fission counters.

Local Power Range Monitor (LPRM)

In the power range, neutron flux is monitored by fixed in-core ion chambers which are arranged in a uniform pattern throughout the core. These chambers cover a range of about 1% to 125% of rated power on a linear scale. When a control rod or group of control rods is selected for movement, the readings from the adjacent detectors are displayed on the operator's control benchboard together with a display of the position of the rod or group of rods.

Detector assemblies each contain four fission chambers and a calibration guide tube for a traversing ion chamber. The chambers are uniformly spaced in an axial direction and lie in four horizontal planes. Each ion chamber is connected to a d-c amplifier with a linear output. Internal controls permit adjustment of the amplifier gain to compensate for the reduction of chamber sensitivity caused by burnup of its fissionable material.

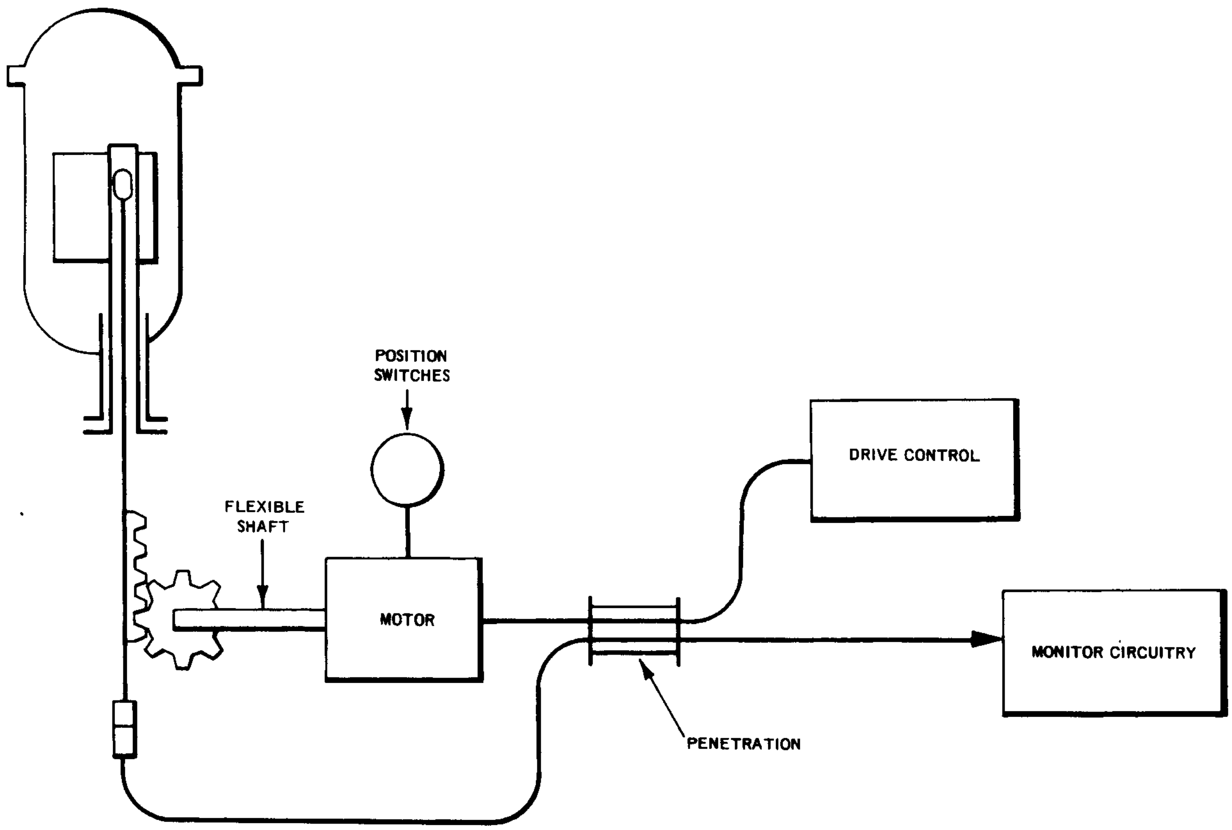


Figure 6-4. Detector Drive System

These detectors are individually replaced through the bottom of the reactor vessel as described in Section 8. The design life of the ion chambers is 1.1×10^{22} nvt before the neutron-to-gamma current ratio drops to 5:1.

Traversing In-Core Probe (TIP)

The calibration guide tube included in each fixed in-core assembly permits the insertion of a traversing ion chamber to obtain vertical flux profiles and to calibrate the chambers. Each calibration guide tube extends nearly to the top of the active portion of the core and is sealed at the upper end. The tubes pass through the nozzles and seals beneath the reactor vessel and connect to an indexing mechanism located inside the containment. The indexing mechanism permits the traversing ion chamber to be directed to many different detector assemblies.

Flux readings along the axial length of the core are obtained by fully inserting the traversing ion chamber into one of the calibration guide tubes, then taking data as the chamber is withdrawn. The data goes directly to the computer. One traversing chamber and its associated drive mechanism is used for each group of seven to nine fixed in-core assemblies (depending on reactor size).

Average Power Range Monitor (APRM)

The average power level is measured by four average power range monitors. Each monitor measures bulk power in the core by averaging signals from as many as 24 detectors distributed throughout the core. The output signals from these monitors are displayed and are also used to operate trips in the reactor protection system.

NUCLEAR SYSTEM PROTECTION SYSTEM

The nuclear system protection system is a four-channel electrical alarm and actuating system which monitors the operation of the reactor, which, upon sensing an abnormal condition, initiates action to prevent an unsafe or potentially unsafe condition. The system integrates the following functions:

- **Reactor Trip** — Monitors reactor operation and shuts down the reactor when certain limits are exceeded.
- **Nuclear System Isolation** — Isolates the reactor vessel and all connections of the primary pressure boundary that penetrate the containment barrier.
- **Engineered Safety Feature Actuation** — Actuates engineered safety feature systems.

The nuclear system protection system uses "solid state" electronic technology from sensor output to actuation device inputs which include sensors, signal conditioning, combinational logic and actuator logic. The system provides for the analog indication of major variables, separation of divisions and on-line testability.

Logic bases for the nuclear system protection system functions are as follows:

- Reactor trip initiation for automatic control and reactor shutdown is based on a two-out-of-four logic.
- Nuclear system isolation by isolation valve closure in process lines penetrating the containment barrier is based on two-out-of-four logic for main steam isolation valves, and a one-out-of-two taken twice logic for remainder of nuclear system isolation function.
- Engineered safety feature systems initiation is based on a one-out-of-two taken twice logic.

Sensors can be either analog (such as process control transmitters) or digital (such as pressure switches or limit switches). Analog inputs for important variables drive indicators which allow the operator not only to see the absolute value, but to compare readings in different channels. Both analog and digital signals are modified, if necessary, in signal conditioners to signals that are compatible with the solid-state logic. After conditioning, the digital signals go directly to the decision logic. Each conditioned analog signal is compared with the output of a set-point generator in a bistable trip unit. When the pre-set level is exceeded, the bistable puts out a signal to the decision logic.

The decision logic is made up of solid-state circuitry that compares with various inputs. When a combination of inputs requires action, the logic circuitry provides a signal that seals in to turn on a solid-state power gate that operates activation devices, such as contactors, circuit breakers, and solenoid pilot valves. The actuation devices in turn control power to the motors that operate valves and drive pumps, or control the air supply to pneumatically operated valves.

Simultaneous open and close manual switch conflicts are prevented by exclusive "OR" logic. Manual inputs may be either momentary or maintained. The identity of the most recent momentary input is retained. When a maintained manual input is removed, the input channel reverts to automatic status.

Upon loss of a-c power, functions which are normally energized (such as the reactor trip function) will provide fail-safe trip action. For such functions, loss of power to a sensor, its channel, or associated logic automatically produces a trip output. For normally deenergized functions (such as emergency core cooling), loss of power to a sensor, its channel, or associated logic leaves the state of the actuated equipment unchanged. Subsequent restoration of power will not introduce transients that could cause a change of state in the actuated equipment.

Reactor Trip Function

The nuclear system protection system initiates the rapid insertion of the control rods to shut down the reactor. The system is of the fail-safe design where it will trip on loss of electrical power, but will not trip and cause a scram on the loss of a single power source. The four trip channels are physically separated from each other and from other equipment precluding the possibility of interactions that could cause possible false scrams or failure to scram. The logic requires a manual reset by the operator which is automatically inhibited for 10 seconds. One reset switch is used for each trip channel. Failure of a single trip channel, division logic, or a system component will not prevent the normal protective action of the nuclear system protection system.

Nuclear System Isolation Function

The nuclear system protection system provides for the closing of valves to isolate the containment thereby preventing the release of steam and process fluids. The logic and equipment required for those valves which are required to be open during emergency core cooling are part of each of the separate emergency core cooling systems.

The lines which penetrate the containment and are required to be isolated during emergency core cooling consist of three groups:

- **Reactor Coolant Pressure Boundary Isolation** — These are lines which connect directly to the reactor vessel and penetrate the drywell and containment barrier.
- **Containment Isolation** — These are lines which do not connect to the reactor vessel but penetrate the drywell and containment atmosphere.
- **Closed System Isolation** — These are lines that penetrate the containment but are neither part of the reactor coolant pressure boundary nor are they connected directly to the containment atmosphere.

All isolation valves except non-testable check valves are capable of remote manual control from the control room. Automatic closure signals override manual control signals. Once isolation has been initiated, valves close fully and will not reopen automatically when the signal clears. Valve position (except non-testable check valves) is indicated in the control room.

Power and control systems associated with containment isolation are multichannel, fail-safe types. Failure of a single sensor circuit or system component does not prevent normal protective action. Two valves in the same line are fed by separate routes from different, reliable power sources. Control power and motive power for an electrically operated valve are supplied from the same source.

Engineered Safety Features Actuation Function

The engineered safety features include the emergency core cooling systems (high pressure core spray system, low pressure core spray system, low pressure coolant injection function of the residual heat removal system and the automatic depressurization function of the nuclear boiler system) and the reactor core isolation cooling system (see Section 4).

Divisional Separation

Four divisional separations are used for reactor trip, isolation and emergency core cooling inputs and outputs, both physically and electrically. Physical separation divisions are established by their relationship to the reactor vessel which is divided into four quadrants. The sensors, logic and output of the various systems are allocated to divisions.

Connections between divisions are isolated optically at the output of the originating cabinet or panel and buffered electrically at the receiving cabinet or panel. Connections to external devices, such as annunciators, indicators, and the computer, are similarly isolated and buffered.

Power Distribution — Both a-c and d-c power are required for the nuclear system protection system. Power distribution is divided into four divisions.

An inverter supplied from either the a-c emergency bus or the d-c battery bus provides a-c power for the scram solenoid pilot valves and the main steam isolation valves.

Reset and Annunciation

A momentary trip of any channel is annunciated and causes that channel to lock out until manually reset. Sufficient annunciation trip signals are used so that the operator can determine the particular sensor or sensors which caused the channel trip. The computer also prints out the identification of sensors which have caused scram and, if several variables are involved, it prints out the sequence of events in which they occurred.

Backup Protection

Two three-way normally deenergixed solenoid valves are used to remove the main instrument air supply from all scram valves. If any of the scram pilot valves failed to operate properly during scram, then the associated control rods would be scrammed by the loss of air supply due to operation of the back-up scram valve.

Conditions Monitored and Inputs That Activate The Nuclear System Protection System

High Pressure in the Drywell — Abnormal drywell pressure trips the reactor, initiates the automatic depressurization function, the high pressure core spray system, the low pressure core spray system, and the residual heat removal system.

Low Water Level in the Reactor Vessel — A low water level in the reactor vessel trips the reactor, causes nuclear system isolation, activates the automatic depressurization function, initiates the high pressure and low pressure core spray systems, and initiates the reactor core isolation cooling system.

High Pressure in the Reactor Vessel — High pressure in the reactor vessel will trip the reactor and initiate automatic depressurization function.

High Neutron Flux — High neutron flux will cause a reactor trip.

High Water Level in the Scram Discharge Volume — High water level in the control rod drive scram discharge volume will cause a reactor trip.

Turbine Stop Valve Closure and Turbine Control Valve Fast Closure — Turbine stop valve closure and fast closure of the turbine control valve will cause a reactor trip.

Main Steam Line Isolation — The closure of the main steam line isolation valves will cause a reactor trip.

High Radiation Activity Near Main Steam Line — High radiation levels near the main steam lines will cause a reactor trip and nuclear sytstem isolation.

Leak Detection — Excessive leakage will cause nuclear system isolation.

Low Pressure at the Turbine Inlet — Low pressure at the turbine inlet will cause nuclear system isolation.

Bypass and Interlocks

An operation mode switch on the reactor control panel controls the interlocking and bypassing of the protection system for the various operational modes. Following are the modes and interlocks provided.

Shutdown — This mode is for use when the reactor is to be shut down and maintenance work performed. All rods must be fully inserted and none can be withdrawn.

Refuel — This mode is for use during refueling operations. It allows a single control rod to be withdrawn for test purposes.

Startup and Standby — This mode is for starting up of the reactor and bringing it to a maximum of about 5% rated power. It also permits keeping the reactor critical while the turbine and associated equipment are being serviced with the main steam line isolation valves closed.

Run — This mode is for normal operation. The intermediate range flux scram is bypassed and all other function bypasses are removed. However, bypassing of some individual instruments for maintenance may be accomplished where permitted by operating procedures.

Interlocks are used on the intermediate range neutron monitors to ensure that all units are operating properly and on the proper range. Control rod withdrawal is blocked if the ratio of reactor power to recirculation flow exceeds a predetermined value.

ROD CONTROL AND INFORMATION (RC&I)

The primary purpose of the rod control and information function is to effect control rod motion as requested by the operator. It displays all information which is relevant to the movement of rods. In addition to enabling the operator to move rods, this function also enforces adherence to operating restrictions which limit the consequences of a potential rod drop accident. At higher power levels it limits rod movement so that rods cannot be withdrawn to the point of generating excessive heat flux in the fuel. Unit conditions are considered in determining which restrictions are applied to a given rod movement request.

Rod position is sensed by a series of sealed glass reed switches contained within a tube inside the drive piston. Two switches are spaced every 3 inches (76mm) with each of the dual switches feeding a separate channel. These signals are multiplexed inside the containment and transmitted to the control room. The rod position information function decodes these data and makes them available to other parts of the rod control and information function, to the process computer and to the operator. The detection of an invalid input caused by a failed reed switch is indicated. The status of the scram valves and accumulators on the hydraulic control unit is monitored and these data are available to the operator and the computer.

The speed and capacity of the rod control and information function permit the control of more than one rod at a time. Up to four rods can be operated simultaneously. The position of each rod in a gang is monitored.

PROCESS RADIATION MONITORING SYSTEM

Certain plant processes are monitored to detect radioactivity in excess of acceptable limits. Such monitoring is described below.

Main Steam Line Radiation Monitoring

The main steam lines located in the steam tunnel (downstream of outer isolation valves) between the nuclear system and the turbine are monitored continuously for gamma radiation for the purpose of detecting increased radiation levels as might be caused by gross fuel failures.

High radiation levels initiate reactor trip, closure of the main steam line isolation valves, termination of operation of the condenser mechanical vacuum pump, and closure of the condenser mechanical vacuum pump line valve, to preclude release of excessive radiation to the environment.

Four gamma-sensitive ionization chambers monitor the area of the steam tunnel as close to the drywell as practical. The detectors are arranged such that each detector views all steam lines with approximately the same exposure.

Each ion chamber detector and instrumentation channel feeds signals to a log radiation monitor. The monitor has two upscale trips, one downscale trip, and one trip indicating an inoperative instrument. The upscale trips and the instrument-inoperative trips are inputs to the nuclear system protection system.

Air Ejector Offgas Radiation Monitoring and Sampling — Prior to Treatment

Condenser offgases are continuously monitored for gross gamma radiation, and offgas samples are taken prior to treatment by the offgas treatment system. Indications of increased activity at this point permit the operator time (typically 15 minutes) to take appropriate corrective action and abort the automatic closure of the offgas isolation valve that would occur upon reaching off-limit levels. The radiation level of fission gases monitored is also an indication of the general level of fuel cladding defects in the core.

All instrument trips alarm in the control room.

An offgas vial sample panel permits the taking of offgas grab samples for laboratory analysis.

Air Ejector Offgas Radiation Monitoring and Sampling — Following Treatment

An offgas sample is drawn from the offgas process stream following the charcoal adsorbers. The sample stream passes in parallel through two shielded detector chambers in series and through filters for the collection of particulates and halogens. The sample stream is returned to the offgas process stream at a location prior to the charcoal adsorbers.

Redundant instrumentation channels (one monitoring each of the two shielded chambers) include a detector, a process log count-rate radiation monitor, trip auxiliaries, and a two-pen recorder (one pen for each channel). The upper-level upscale trip, in conjunction with the downscale trip, isolates the offgas treatment system, drain valves, and alarms. Trips alarm in the control room.

The operation of the instrument channel can be checked for preoperation by purging the sample line and then monitoring a check source included as part of the monitoring system.

An offgas vial sample panel permits the taking of offgas grab samples for laboratory analysis.

Offgas Vent Pipe Radiation Monitoring — Pre-Release

The gases discharged to the environs from the offgas vent pipe are continuously monitored for radiation and sampled for halogens and particulates. The monitoring of this effluent provides a permanent record of the gross gamma radiation discharged to the environs.

A gas sample is drawn through an isokinetic probe located high enough in the exhaust stream to ensure representative sampling. The sample stream passes in parallel through a shielded chamber which contains a scintillation detector, and through filters for the collection of particulates and halogens. The sample stream is returned to the offgas vent pipe.

The instrumentation channel includes a detector, a process radiation monitor, and a continuous strip-chart, single-pen recorder. The recorder and a five-decade meter provide readout in the control room. The process radiation monitor has three adjustable trips, two upscale and one downscale, which alarm in the control room.

Containment Ventilation Discharge Monitoring and Sampling

The ventilation air discharged to the environs from the containment is continuously monitored for radiation and sampled for halogens and particulates. The monitoring of this effluent provides a permanent record of the gross gamma radiation discharged to the environs.

A gas sample is drawn through an isokinetic probe located high enough in the exhaust stream to ensure representative sampling. The sample stream passes in parallel through a shielded chamber which contains a scintillation detector, and through filters for the collection of particulates and halogens. The sample stream is returned to the ventilation duct.

The operation of the instrument channel can be checked for proper operation by purging the sample line and then monitoring a check source included as part of the monitoring system.

Liquid Process Radiation Monitoring

The effluent from the site service water, the liquid radioactive waste treatment system, each of the redundant service water cooling systems for the residual heat removal system and the process fluid of the closed cooling water system for reactor service are continuously sampled and monitored for gross gamma radiation levels. Each instrument channel includes a gamma-sensitive scintillation counter which monitors the flowing sample and a radiation monitor. The instrument channel monitoring the residual heat removal system cooling water also includes a two-pen recorder, one pen for each of the two cooling loops.

Air Ejector Offgas Carbon Bed Vault Radiation Monitoring

The vault housing the charcoal adsorbers associated with the air ejector offgas system is monitored for gamma radiation by a single instrument channel which includes a sensor and converter, an indicator and trip unit, and a locally mounted auxiliary unit. The channel is powered from one of the power supplies used for the monitoring units associated with the containment ventilation exhaust plenum. The channel provides for readout both locally and in the control room. The indicator and trip unit has an adjustable upscale trip for indication of high radiation levels and one downscale trip for indication of instrument trouble. A trip alarms in the control room.

Containment Exhaust Monitoring

Air discharged to the environs from the containment is continuously monitored for radiation and sampled for halogens and particulates. The monitoring of this effluent provides a permanent record of the gross gamma radiation discharged to the environs.

A gas sample is drawn through an isokinetic probe located high enough in the exhaust stream to assure representative sampling. The sample stream passes in parallel through a shielded chamber which contains a Geiger-Muller tube-type detector and through filters for the collection of particulates and halogens. The sample stream is returned to the ventilation duct.

The instrumentation channel includes a detector, a log count rate meter which includes a power supply, and a continuous strip-chart single pen recorder. The detector is a beta sensitive Geiger-Muller tube whose output pulse is counted by the log count rate meter. The recorder and a five-decade meter with a maximum scale reading of 10^6 counts per minute provide readout in the control room. The rate meter has two adjustable trips, one upscale and one downscale, which alarm in the control room.

The operation of the instrument channel can be verified for proper operation by purging the sample line and then monitoring a check source included as part of the monitoring system.

AREA RADIATION MONITORING SYSTEM

A number of gamma-sensitive detectors are located throughout the plant and recorded on a multipoint recorder in the control room. Monitors are available in several sensitivity ranges and are chosen to match the expected radiation levels. Local indication and annunciation are also available for frequently occupied areas.

LEAK DETECTION SYSTEM

The detection of leaks occurring within the reactor drywell and leaks occurring outside the reactor drywell, including the containment, fuel, and auxiliary equipment buidings, and the steam line tunnel, uses sensing and monitoring instrumentation for indication of abnormal flows in process piping systems, abnormal pressure and temperature changes, increased equipment drain and floor drain sump flow activity, and abnormal radiation levels.

Small leaks are generally detected by temperature and pressure changes, and fill-up rate of drain sumps. Larger leaks can also be detected by changes in reactor water level and changes in flow rates in process lines.

COMPUTER

The computer system for the BWR is designed to reliably monitor plant parameters and perform a series of operations and calculations designed to allow increased plant operation efficiency.

Operation

Several significant operations are accomplished by the computer. Total reactor core power is computed. The core power distribution is ascertained by neutron flux measurements and analytical procedures. Limiting power densities and critical heat flux ratios within the core are determined. Fuel isotopic composition, local power range monitor sensitivities, and correction factors are also calculated, stored and made available for printout and utilization by other programs.

In order to obtain more comprehensive histories of the plant operation and function, many of the calculated results are printed out for final analysis and use by plant personnel. The exposures of reactor fuel, control rods, and fixed in-core detectors are all accumulated and stored in memory. Weighted moderator void fractions which must be computed for use in core calculations are also stored in memory.

Output

Logs — Information resulting from the running of a program or that stored in the computer's memories, is printed out either on direction of one of the controlling programs or upon request by an operator. Output logs are initiated as part of an on-demand program requested by the operator, or they occur as periodic logs which are printed by the computer according to its executive program. The historical logs are programmed by the computer to be printed out hourly, daily, and monthly. The logging of the event recall and sequence-of-events functions is initiated by the change of state of an input and provides the historical data and the sequential data for analysis by operating and maintenance personnel. Summary logs may be printed on demand.

Alarms — As part of the scanning function, the input analog signal is checked against reasonable value limits. The converted digital signal is limit tested according to operator established limits or program calculated limits. If the limits of either the inputs or converted values are exceeded, they will be alarmed. A change in state of preselected digital inputs will be alarmed. Others are not alarmed but merely stored in the computer memory for use in applicable programs when called upon. Alarm information is displayed to the operator on video terminals. Any change in state of a preselected range of digital inputs will initiate a sequence of events recording. Chronological change of state of these preselected inputs is logged by the computer.

The process computer is an integral part of the reactor instrumentation system in order to more quickly and accurately determine the local power density distribution throughout the reactor core. The increase in accuracy allows the core to be operated closer to limits without increasing the probability of fuel deterioration. The increase in speed allows a quicker response to operating conditions. The increased accuracy and speed of determining the core power distribution allow a smaller core to be used for a given power rating.

Function

The primary function of the computer is to perform reactor core calculations and provide the reactor operator with current core operating data. In addition to core calculations, it is the function of the computer to monitor, calculate, store, log, and alarm information which has been collected by the plant instrumentation.

Monitoring

The computer monitors analog, digital, and pulse inputs from the various nuclear system auxiliary systems. Neutron flux level inputs are received from the neutron monitoring system. Control rod position and movement are received from the control rod information functions of the rod control and information system. Reactor pressure, steam and feedwater flow, and reactor vessel water level are received from the feedwater control system. All input variables to the nuclear system protection system are also monitored by the computer. The computer analog inputs are scanned at high speed. Selected contact inputs are monitored for change of state. These are incorporated in a sequence of events function which is initiated by actuation of any input contact in this group. The balance of the contact inputs supply data for specific programs. Along with the NSSS contacts, the state of these contacts is read periodically.

The information gathered from the nuclear system instrumentation is recorded in the computer memories. It is held there and updated regularly so that current values are available when these data are needed in calculations. As the individual programs are executed, the input data are withdrawn from memory and operated upon by the computer. The results of these programs are either printed, displayed or stored as directed by the program.

A computer functional block diagram is presented in Figure 6-5. It displays the interconnections of the main components of the computer in its application to the nuclear boiler.

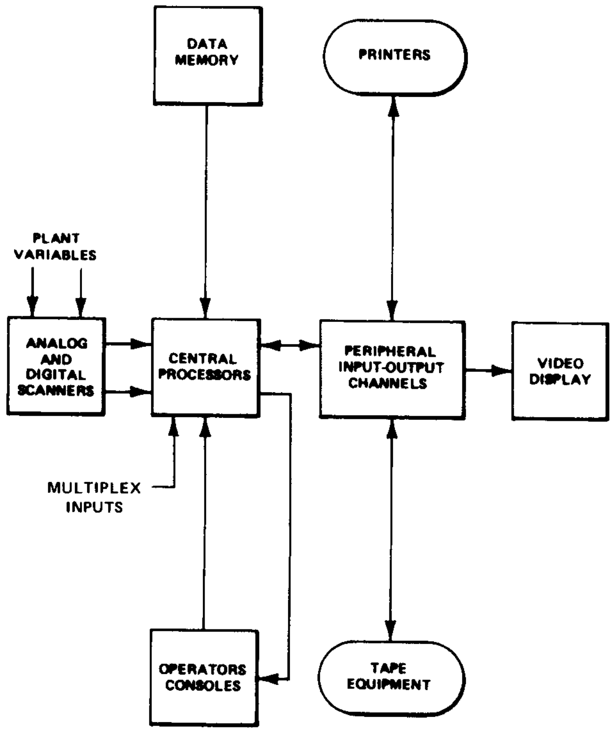


Figure 6-5. Computer Functional Block Diagram

CONTROL ROOM PANELS, CABINETS, AND RACKS

Main and auxiliary control panels, cabinets, and instrument racks with the nuclear boiler controls and instrumentation are located in the control room and elsewhere. The main panels of a compact control room design include equipment such as control switches, indicating lights, instruments, and annunciators, as well as the main controls for process instrumentation. The auxiliary panels, cabinets, and racks include the signal conditioning and sensing equpment for nuclear boiler instrumentation.

NUCLENET™ 1000 CONTROL COMPLEX

To further enhance the operator control, a compact control room design is available, the Nuclenet 1000 Control Complex. From one command position, a single operator can effectively control and monitor operation of the nuclear plant.

The Nuclenet Control Complex is an integrated approach to the design and scope of BWR control rooms. The concept integrates the conventional components of a control room into one system. The design functionally provides the complete engineering and hardware for the signal processing, recording, display, and control required to operate the BWR plant and forms a well-defined boundary between the control room equipment and field cabling.

The Nuclenet Control Complex is composed of three distinct subdivisions integrated to form the control complex. These are the operator's interface system, the performance calculation and monitoring systems, and the power generation control complex (PGCC).

The operator's interface system includes that instrumentation, equipment, and software necessary to provide the operator with information and control required to operate the plant. This includes benchboards, panels, consoles, and displays. (Refer to Figures 6-6 and 6-7, typical operator's console.)

The performance calculation and monitoring system includes the equipment and software required for supporting and supplemental information needed for overall plant operations. This includes the nuclear steam supply (NSS) and balance-of-plant (BOP) core and performance calculations, data logging functions, and other supplemental functions.

The Nuclenet Control Complex incorporates PGCC, a packaging scheme that utilizes factory fabrication to provide a prefabricated control room complex. The auxiliary cabinets, panels, benchboards, and consoles associated with the control room systems are built and mounted on floor sections. Factory fabricated cables provide interconnecting of the complex. The assembly components incorporate the use of termination cabinets for field inputs. Prior to shipment to the site, the entire complex, excluding the termination cabinets, is preassembled at the factory and a factory system test conducted (Figures 6-8 and 6-9).

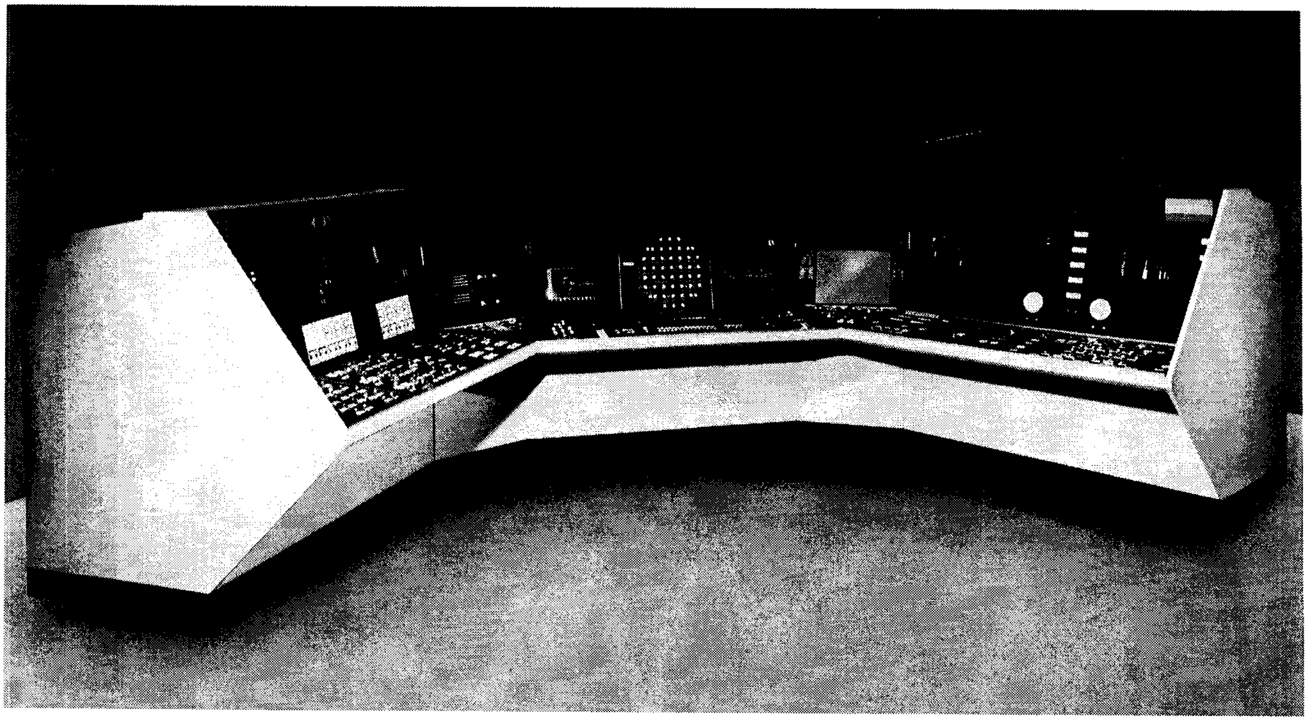


Figure 6-6. Nuclenet Control Complex Operator's Console

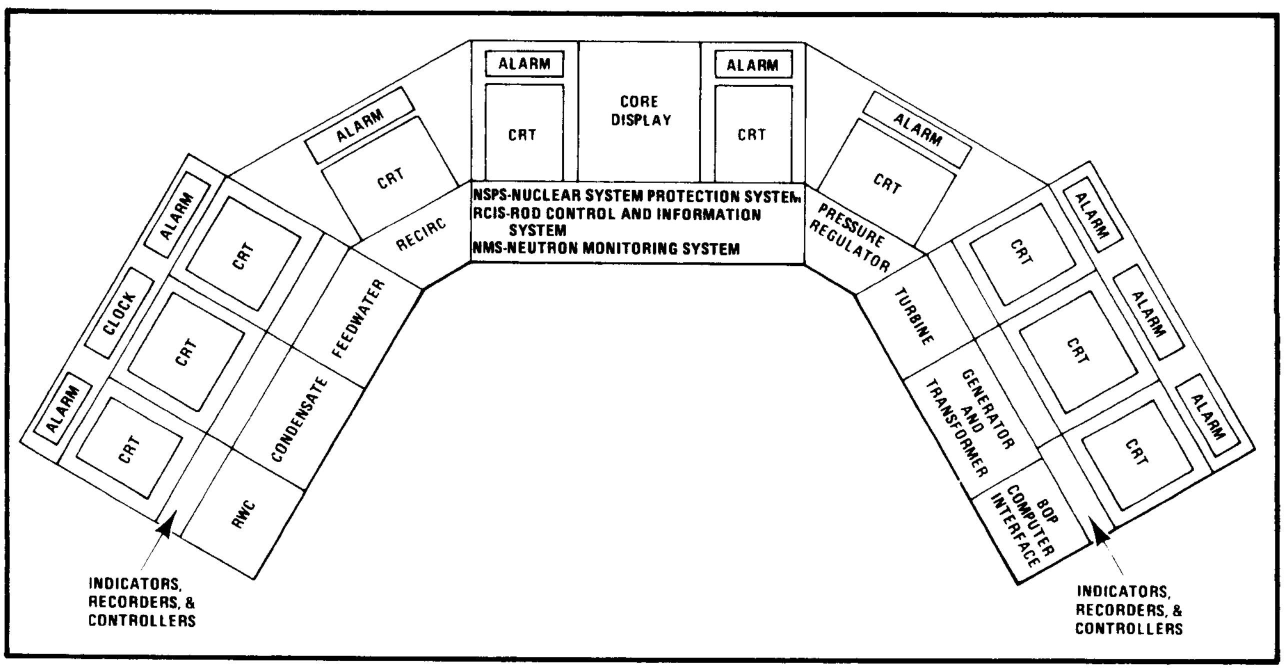


Figure 6-7. Operator Console (Functional Layout)

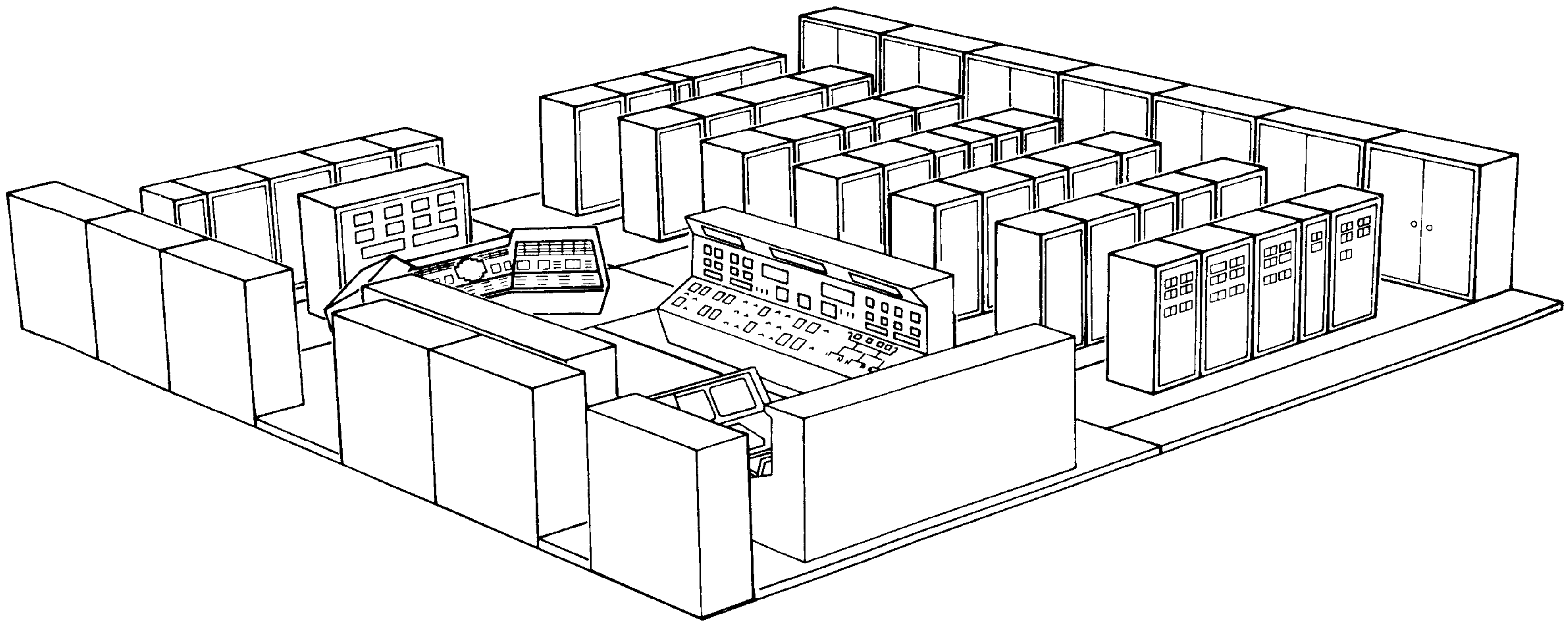


Figure 6-8. Power Generation Control Complex (PGCC)

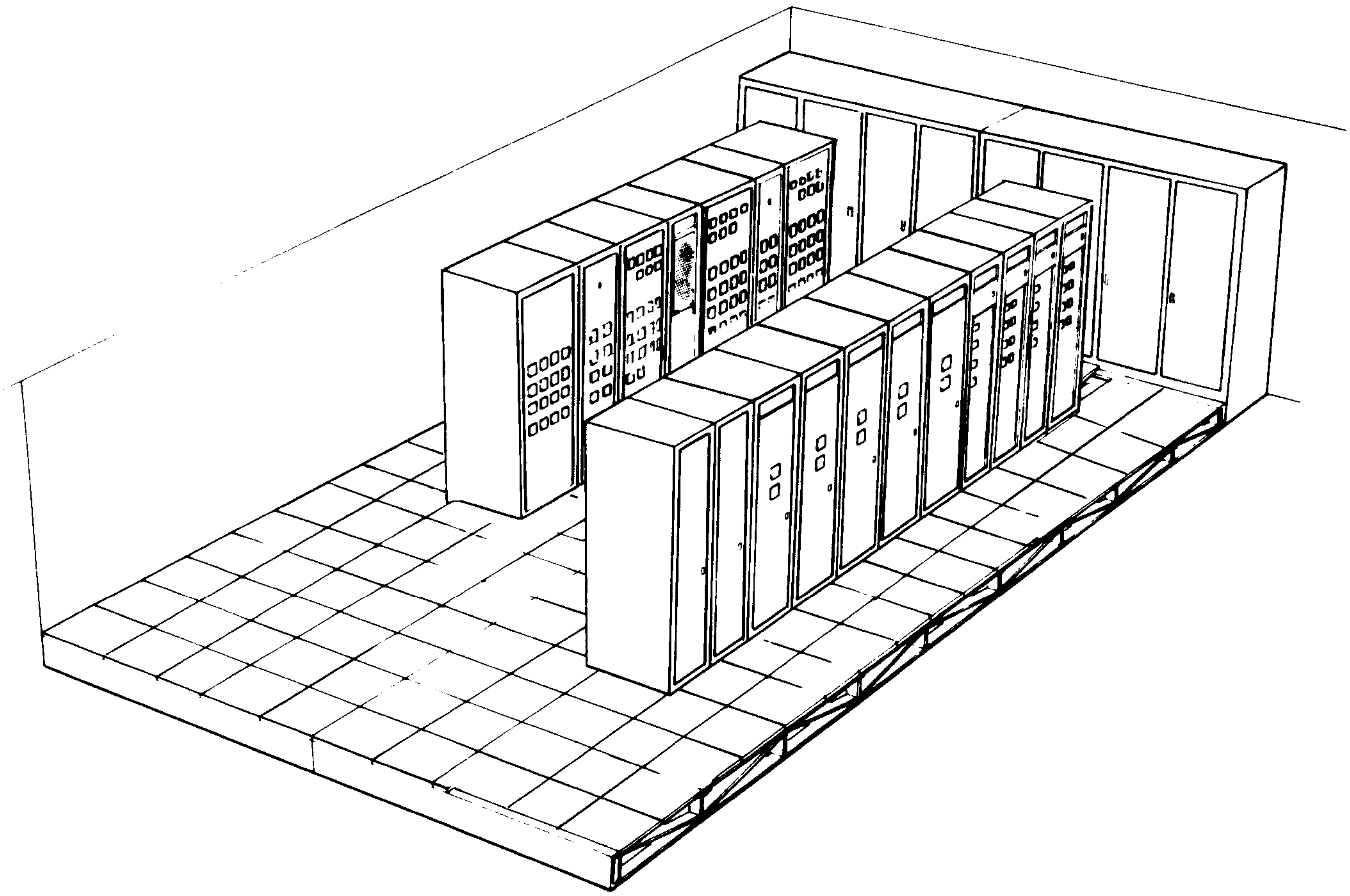


Figure 6-9. Typical Panel Module with Termination Cabinets

Section 7

Nuclear Plant Design

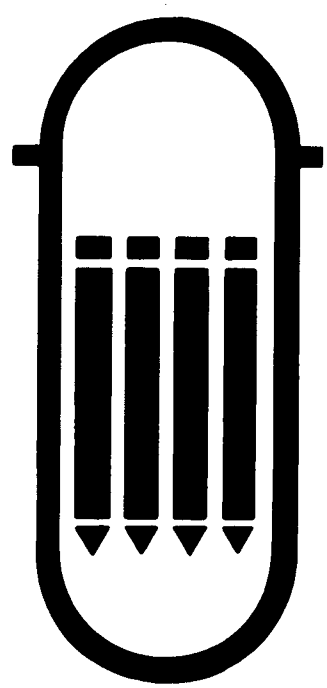
INTRODUCTION

The General Electric BWR nuclear system is used in a containment system which includes the pressure suppression feature to contain the energy and prevent significant fission product release in the event of a postulated design basis loss-of-coolant accident (LOCA).

The reference Mark III containment design is a single-barrier pressure containment and a multi-barrier fission containment system consisting of the containment vessel (pressure and fission barrier), and fission product barriers, which include the shield building, auxiliary building and fuel building, all of which are normally kept at a negative pressure relative to atmosphere, to further limit nuclear radiation to the environment.

REACTOR BUILDING

The reactor building structure, shown in Figure 7-2, includes the shield building, the containment system, the drywell, the reactor vessel pedestal, the reactor shield wall, drywell pipes support structure, the pressure suppression pool, weir wall, the upper containment pool structure, equipment rooms, the operating floor, and miscellaneous platforms. It also includes the steam tunnel which connects the drywell, located in the reactor building, and the turbine building which places part of the steam tunnel in the containment portion of the reactor building and part in the auxiliary building. The steam tunnel is divided into two parts by the shield building. A single, reinforced concrete foundation mat support all reactor building structures. Components located within the drywell include, but are not limited to, the reactor vessel, the reactor water recirculation system, the main steam lines, main steam line safety/relief valves and discharge piping, control rod drives and piping, piping and valves associated with reactor vessel, nuclear system instrumentation, and heating and ventilation. Components located outside the drywell, but inside the containment vessel include, but are not limited to, the control rod drive hydraulic modules, standby liquid control system components, reactor water cleanup system heat exchangers, auxiliary system piping, refueling bridge, polar crane, nuclear system instrumentation, and heating and ventilating. Except for the weir annulus, the vapor suppression pool itself is located inside the containment and outside the drywell.



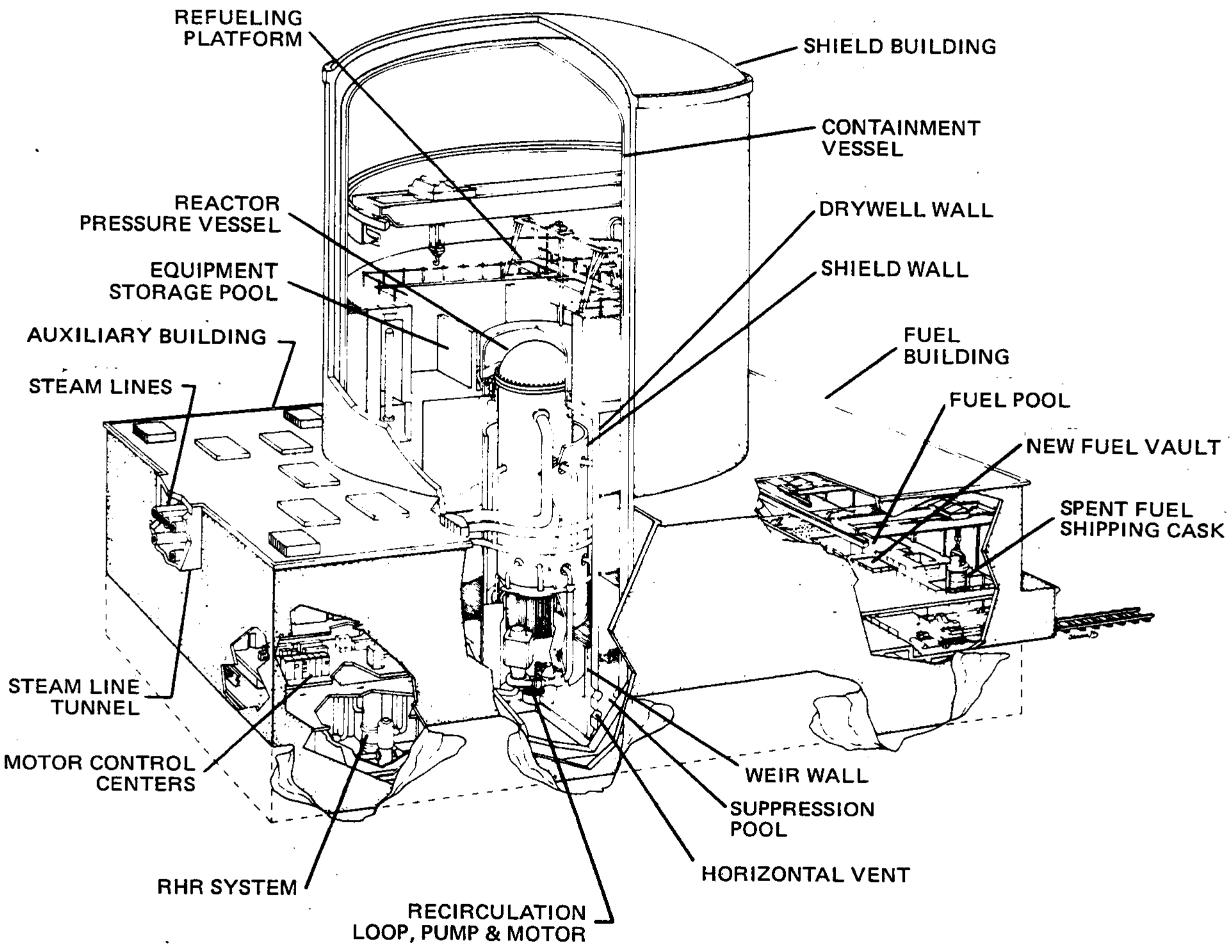


Figure 7-1. Reactor Building, Fuel Building, and Auxiliary Building

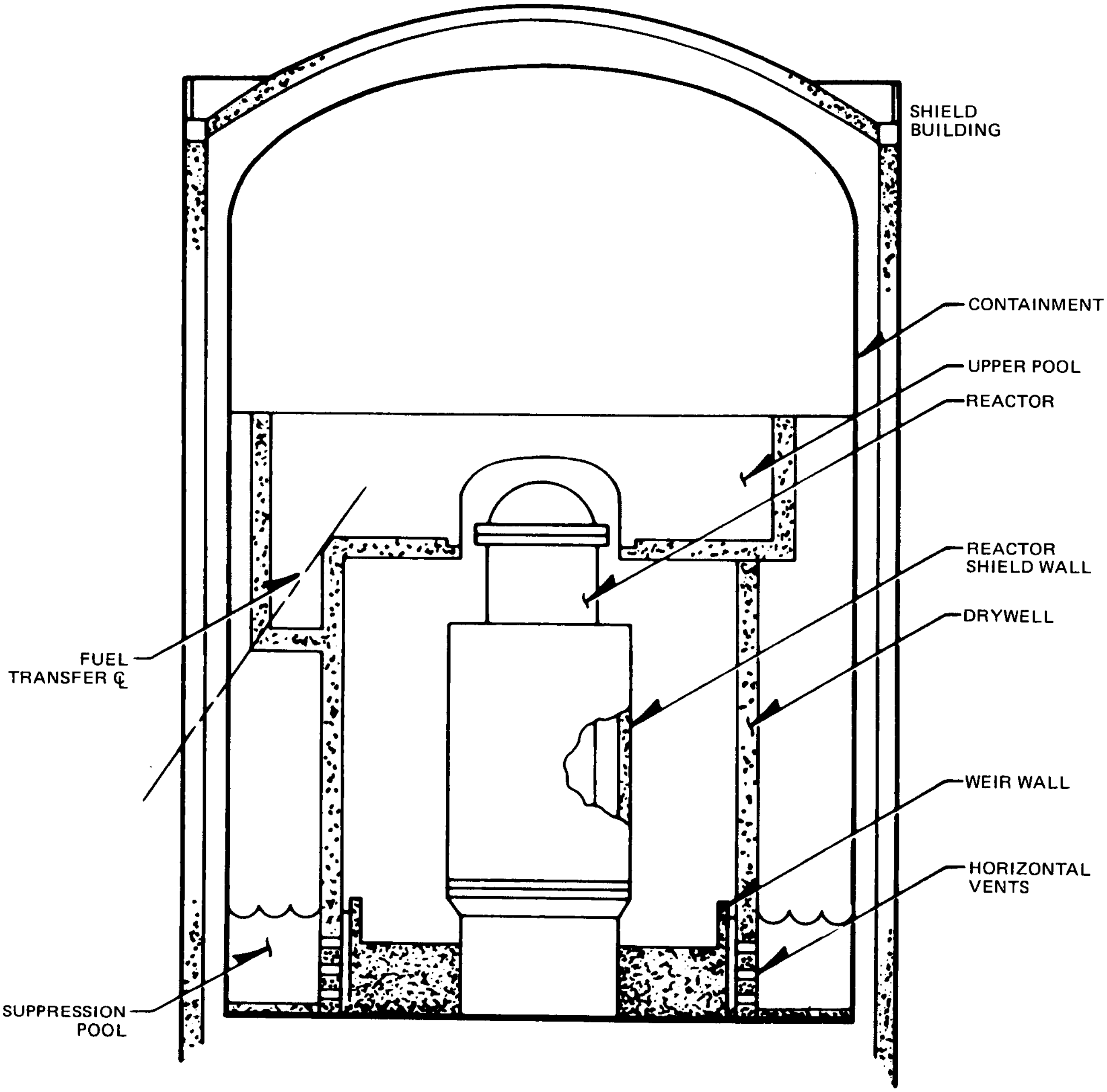


Figure 7-2. Reactor Building (Mark III Containment and Shield Building)

Shield Building

The shield building is a cylindrical shell of reinforced concrete with a shallow spherical dome roof. It is a seismic Category I structure completely enclosing the reference free-standing containment. A radial gap of approximately 5 feet (1.5m) between the containment and the interior face of the shield building permits construction operations and periodic visual inspection of the containment. The primary function of the shield building, the "secondary fission product barrier," is to further limit nuclear radiation to the environment in the event of an accident involving the release of fission products. The structure also protects the containment from adverse atmospheric conditions and external missiles.

The shield building annulus provides a plenum for the collection and filtration of fission product leakage from the containment that may occur following a design basis accident. The annulus is normally kept at a negative pressure relative to atmospheric pressure so any leakage through the shield building or containment is into this space. Under accident conditions, the ventilation exhaust from this space is automatically diverted through the filtered standby gas treatment system before release to the environs.

Containment Vessel

The containment is a steel leakage barrier which prevents significant fission product release to the shield building annulus in the event of an accident.

The reference containment, classified as a seismic Category I structure, is a free-standing, vertical, cyclindrical steel pressure vessel with an ellipsoidal head and a flat bottom steel liner plate. The cylindrical shell has horizontal external stiffeners and is anchored into the reactor building foundation mat. The flat, bottom liner plate, which is continuously supported by and integrated with the foundation mat, serves as a leak-tight membrane only and not as a pressure retaining part. The drywell, weir wall and reactor vessel pedestal loads are transferred through the bottom liner onto the foundation mat.

The containment, including all penetrations and welded attachments, acts as an independent structural component within the reactor building for the maximum temperature and pressure conditions that can occur as the result of a LOCA, and accommodates reactor blowdown through the safety/relief valve discharge piping to the suppression pool. The containment also supports an overhead polar crane which is used for servicing the reactor.

Among the major penetrations through the containment are two, standard, double-door, personnel access locks and an equipment hatch. The locks extend through the shield building and are supported by the containment vessel. One lock is located at the operating floor level and the other is one level above grade. The elevator/stair access tower of the fuel building serves both locks of the reactor building and all levels of the fuel and auxiliary buildings. The double-door locks permit personnel access to the containment during reactor operation for inspection and maintenance of equipment. Penetration seals are utilized at locations where piping, mechanical devices, and electrical conductors pass through the containment. The penetration seals maintain the containment leakage barrier.

Drywell

The drywell is a cylindrical shell of reinforced concrete with a flat roof slab containing a centrally-located, circular access cover above the reactor vessel for refueling and maintenance. It is a seismic Category I structure which (a) supports the upper containment pool; (b) contains the transient pressure resulting from a LOCA and channels the air-steam mixture to the suppression pool; (c) provides shielding to reduce radiation levels in the containment external to the drywell to levels which permit normal access during reactor operation; and (d) provides pipe whip, jet impingement, and missile protection for the containment. The drywell roof slab, which is also the base slab for the upper containment pool, transmits the weight of the pool to the foundation mat through the cylindrical drywell walls. The circular opening in the drywell roof slab over the reactor vessel is covered during operation with a removable steel drywell head which is a part of the pressure boundary.

The drywell head is unbolted and removed during refueling to provide access to the reactor vessel. A vessel-to-drywell seal maintains drywell-containment pool boundary integrity when the drywell head is not in place. Any leakage is drained off to the waste treatment system.

A standard, double-door, personnel lock penetrates the drywell and permits access to it during periods of reactor shutdown or hot standby. Another hatch, which also penetrates the drywell wall, can be removed during periods of reactor shutdown in order to transfer large equipment in and out of the drywell. A motor removal cart, which rides on rails, permits the transfer of the recirculation pump motors out of the drywell for maintenance.

Large diameter horizontal vent openings, used to conduct the LOCA steam to the suppression pool, penetrate through the lower section of the drywell. The vents are formed by sections of pipe which are welded to matching circular openings in the inner and outer face plates of the drywell. Three identical rows of vents are spaced circumferentially around the drywell, providing sufficient vent area for the size of the nuclear steam supply unit.

Suppression Pool and Weir Wall

The suppression pool is an annular pool of demineralized water between the drywell and the outer containment boundary. This pool covers the horizontal vent openings in the drywell to maintain a water seal between the drywell interior and the remainder of the containment volume.

The suppression pool provides (a) a means to condense any steam released in the drywell area during a hypothetical LOCA; (b) a heat sink for the reactor core isolation cooling system during hot standby operation until the decay heat can be piped directly to the residual heat removal (RHR) heat exchangers; (c) a heat sink for venting the nuclear system safety/relief valves; and (d) a source of water for the emergency core cooling systems (ECCS). The suppression pool also serves as a heat sink under normal operating conditions. Blowdown through the main steam safety/relief valves during anticipated reactor transients is routed to the suppression pool where the steam discharges through an "X-type" quencher at the end of the discharge piping and is condensed. The discharges are spaced to ensure uniform, dynamic load distribution and mixing in the pool during safety/relief valve operation.

Upper Containment Pool

The upper containment pool is a rectangular stainless-steel-lined pool crossing the top of the drywell area. The drywell area roof slab serves as the base slab for the pool. The long sides of the pool form two beams which stiffen the drywell roof slab and allow it to withstand internal pressure loading during a LOCA.

The water in this pool provides shielding for the operating floor during normal reactor operation. A portion of this water is also drained to the suppression pool following a LOCA. This makes up for the water inventory pumped into the vessel and the drywell by the emergency core cooling system. Interior crosswalls divide the pool into five areas: (a) steam separator storage; (b) reactor cavity; (c) steam dryer storage; (d) racks for temporary storage of fuel elements; and (e) fuel transfer.

Equipment Rooms

Equipment rooms within the containment provide radiation shielding for certain reactor water cleanup system components located outside the drywell. Reinforced concrete is used for both shielding and strength. The rooms are supported by the drywell and the upper containment pool walls and are separated from the containment.

AUXILIARY BUILDING

The auxiliary building, shown in Figure 7-1, is located adjacent to the reactor building and encloses the main steam tunnel. The lower level of the auxiliary building contains the pump rooms for the High Pressure Core Spray (HPCS), Low Pressure Core Spray (LPCS), Reactor Core Isolation Cooling (RCIC) and Residual Heat Removal (RHR) systems.

The second or grade level of the auxiliary building serves as the primary access to this structure. Cable trays leading to the control room and cable spreading room are located at this level.

The third level of the auxiliary building houses electrical switchgear, battery rooms, motor control centers and electrical penetrations. The top level of the auxiliary building houses the heating and ventilation supply systems and water chillers for the reactor and auxiliary building.

Steam Tunnel

The steam tunnel contains the four main steam lines, two feedwater lines and other process piping linking the drywell area with the turbine building via the auxiliary building. The steam tunnel is a rectangular reinforced concrete structure supported by the drywell and provides shielding for the contained process lines. A gap between the tunnel and the containment allows for differential movement.

FUEL BUILDING

The fuel building, shown in Figure 7-1 provides for the receiving of new fuel, the storage of new and spent fuel, the preparation and servicing of fuel, the transfer of new and spent fuel to the reactor servicing pools within the reactor building, and the preparation of spent fuel for off-site shipment. The fuel building is located adjacent to the reactor building and is accessible for fuel servicing during plant operation.

A lined fuel pool is used for the storage and servicing of spent fuel and the preparation of new fuel for insertion into the reactor. An area of the pool, separated by gates, is used for transfer of fuel to the reactor servicing pools located in the reactor building, and the receiving of spent fuel discharged from the reactor using a transfer tube. Another area of the fuel storage pool, also separated by gates, is used for the loading and decontamination of equipment and its containers for off-site shipping.

The process system equipment for maintaining the temperature and quality of water for the fuel building pools and for the upper containment pool is located within the fuel building.

Section 8

Operation, Training and Maintenance

INTRODUCTION

Commercial operation and maintenance of a nuclear power station requires a comprehensive organization of trained personnel as well as a program and documented procedures to support this organization. This section provides an outline of the typical requirements for an organization and supporting program to carry out a successful nuclear power plant operating and maintenance program. Organizational requirements, as well as the various aspects of operation and maintenance, are discussed.

NUCLEAR POWER STATION ORGANIZATION

The functional responsibility of a staff to operate a boiling water reactor power station is similar to that required for an equivalent-sized conventional, modern, fossil-fueled power generating station, with the additional requirement of special knowledge and skill in nuclear phenomena and fundamentals.

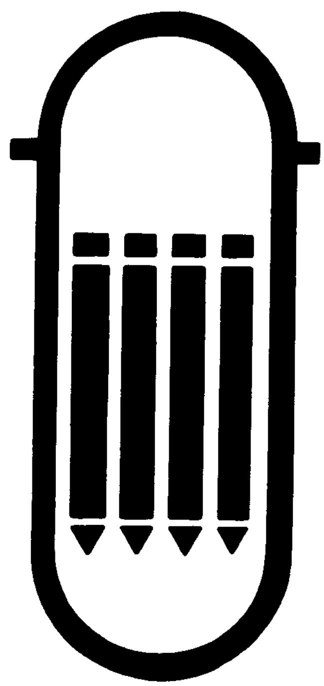
Staff and operating functions authorized to operate the nuclear reactor are subject to NRC examination and licensing. The station staff can, however, vary through a wide range, depending upon the utility's practices for staffing conventional stations and the planned use of nonstation personnel for major maintenance outages. The station staff should be self-sufficient in all operations. A minimum staff for a typical single-unit nuclear power plant is 97, and for a dual-unit plant, about 140.

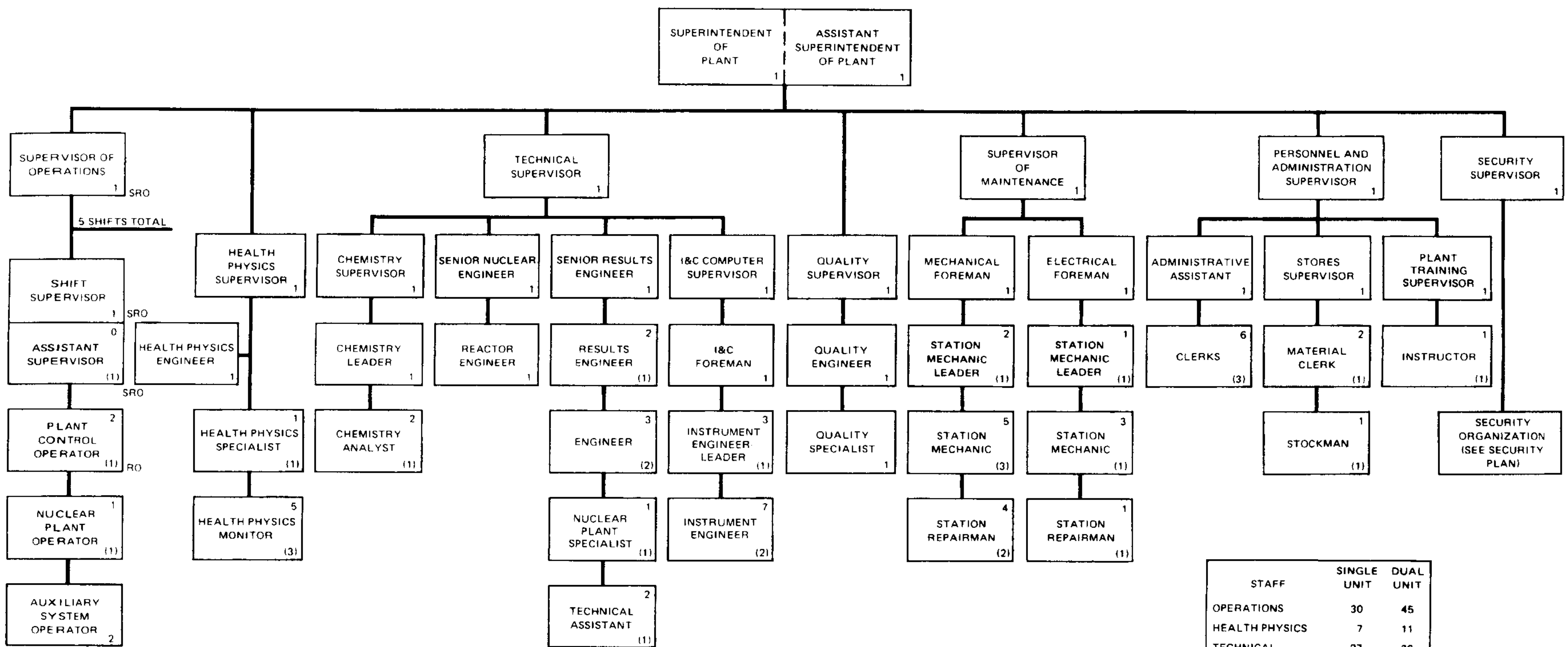
The organizational chart shown in Figure 8-1 represents the combined utility experience of General Electric in the operation and startup of more than 40 General Electric BWR units.

Plant Management

The **superintendent of plant** is responsible for adherence to the operating license and technical specifications, for safeguarding the general public and station personnel from radiation exposure. He is also responsible for the safe, efficient, and reliable operation, and maintenance of the facility, and for execution of the administration controls and quality-related activities within the station.

The **assistant superintendent of plant** assists the superintendent of plant in these duties and assumes these direct responsibilities in the superintendent's absence.





KEY: RO REACTOR OPERATING LICENSE
SRO SENIOR REACTOR OPERATING LICENSE
= ESTIMATED NUMBER OF EMPLOYEES IN POSITION FOR UNIT 1
(=) ESTIMATED NUMBER OF ADDITIONAL EMPLOYEES FOR UNIT 2

STAFF	SINGLE UNIT	DUAL UNIT
OPERATIONS	30	45
HEALTH PHYSICS	7	11
TECHNICAL	27	36
QUALITY	2	2
MAINTENANCE	18	27
PERSONNEL AND ADMINISTRATION	13	19
SECURITY – SEE SECURITY PLAN		
TOTAL	47	140

Figure 8-1. Typical Nuclear Power Station Organization

The **supervisor of operations** is responsible for the routine as well as abnormal or emergency operating situations that may arise. He is responsible for seeing that all operations are carried out in a safe, efficient manner, and that the plant is operated in conformance to the operating license, technical specifications, and in accordance with approved written procedures.

The **health physics supervisor** is charged with the responsibility of providing the superintendent of plant with the information necessary to establish compliance with regulations pertaining to radiation safety, uniform enforcement of station health physics requirements, and that every reasonable effort to minimize personnel exposures has been made.

The **quality supervisor** reports to the superintendent of plant, assuring the degree of independence necessary to obtain objective verification of quality-related activities. The quality supervisor has stop work authority on any nonconforming activity being performed by plant personnel or subcontractors.

The **personnel and administration supervisor** is responsible for development and implementation of programs and procedures in the areas of clerical, records management, training, and plant stores.

Operating Organization

The operations staff performs the operating functions at the plant. A functional description of duties follows. Operations staffing for single and dual plants is shown below, based on a five-platoon shift coverage.

Staff	Single Unit	Dual Unit
Shift Supervisor	5	5
Assistant Shift Supervisor	0	5
Plant Control Operator	10	15
Nuclear Plant Operator	5	10
Auxiliary System Operator	10	10
TOTAL	30	45

The **shift supervisors** are directly responsible to the supervisor of operations. They are in charge of and responsible for the shift crew assigned to their specific shift. Ths shift supervisors have the authority to shut down the plant, if, in their opinion, plant conditions indicate that a nuclear safety hazard exists or indicate approved procedures so direct. Shift supervisors direct plant operations, supervise and check the performance of control room and other operators, inspect equipment when required, authorize various forms such as blocking permits, radiation work permits, maintenance work requests, and radwaste release forms, and respond to plant or equipment abnormalities in accordance with approved plant procedures. Additionally, shift supervisors inform station management in a timely manner of conditions which may affect public safety, plant personnel safety, plant capacity or reliability, or cause hazard to equipment. Shift supervisors log significant plant operations and problems, and determine that sufficient manpower is available for the operations planned for the next shift in order to ensure adequate plant staffing at all times. Additionally, the shift supervisors may, at times, direct the activities of other personnel during tasks such as backshift maintenance.

The **assistant shift supervisors** assist the shift supervisors in these duties and assume these direct responsibilities in the event of the unavailability of the shift supervisors. The shift and assistant shift supervisors participate in operator training, retraining and requalification activities from the standpoint of providing guidance, direction, and instruction to shift personnel, as well as pursuing academic work to keep their SRO licenses current. As assigned by the supervisor of operations, the assistant shift and shift supervisors review procedures as they apply to startup, shutdown, power operation, load changes, fuel handling, emergency situations, surveillance activities, etc., from the standpoint of safety, accuracy, and experience gained as the result of operation.

Plant control operators (PCO's) perform operations directed by shift supervision, monitor control room instrumentation, respond to plant or equipment abnormalities in accordance with approved plant procedures, direct the activities of the other operators, and log plant operations, systems or equipment abnormalities, and plant data. Plant shutdowns or scrams may be initiated by these operators when observation of plant conditions and equipment indicate a nuclear safety hazard exists, or approved procedures so direct. The PCO's manipulate process controls as necessary to match load demand and to respond to other process changes.

The PCO's are responsible for taking immediate operator action necessary to maintain or bring the plant to a safe condition during abnormal and/or emergency conditions. The PCO's participate in the training of new PCO's and maintain their reactor operator licenses by taking part in the license requalification program.

The **nuclear plant operators** (NPO's) and **auxiliary systems operators** (ASO's) perform routine duties outside the control room as assigned and necessary for continuous, safe plant operation and are available to the shift and assistant shift supervisors for additional work assignments that arise from time to time. They assist in plant startup, shutdown, surveillance, and emergency response as directed by assistant shift or shift supervisors and/or the PCO's. For those tasks requiring the use of checklists or procedures, such as valving for plant startup, or data sheets used on routine equipment checks, they are responsible for performing these assigned evolutions and for making accurate entries according to the applicable procedure, data sheet, or checklist. The NPO's and ASO's are responsible for assisting in the training of new employees and for improvement and upgrading of their own performance by participating in the applicable sections of the training program.

Shift personnel will be trained in Health Physics

The shift personnel are instructed that should conditions exceed preestablished radiation levels or exposure limits, or if conditions occur that are deemed to be unsafe or hazardous, the shift or assistant shift supervisor will be informed. A health physics monitor or qualified alternate will be assigned to each shift, and if the assistant shift or shift supervisor feels that a radiological condition exists that warrants attention and investigation by health physics, he will direct this individual to assist as necessary.

Health Physics Organization

The **health physics supervisor** is responsible to the superintendent of plant. The health physics supervisor is charged with the responsibility of providing the superintendent of plant with the information necessary to establish compliance with regulations pertaining to radiation safety, uniform enforcement of station health physics requirements, and for determining that every reasonable effort to minimize personnel exposures has been made. In addition, the health physics supervisor is responsible for assuring that the staff who implement the health physics program is trained and retrained in operational health physics principles.

The **health physics engineer** is removed from the line function of day-to-day health physics activities to provide the latitude and time to develop and implement a station as low as reasonably achievable (ALARA) program that is responsive to plant status. The health physics engineer's major responsibility is to provide the health physics supervisor with the information necessary to establish that every reasonable effort has been made to minimize personnel exposures.

The **health physics specialists** assure implementation of the station health physics program by supervision of routine and special survey and evaluation programs required by applicable regulations and procedures. The health physics specialists' major responsibility is to provide the health physics supervisor with the information necessary to establish that survey and recordkeeping requirements are properly met and that plant activities receive appropriate health physics attention.

Staff	Single Unit	Dual Unit
Health Physics Engineer	1	1
Health Physics Specialist	1	2
Health Physics Monitor	5	8
TOTAL	7	11

Technical Organization

The **technical supervisor** reports to the superintendent of plant and is responsible for all activities of the Technical Support Organization.

The Technical Support Organization is responsible for performing functions in the areas of chemistry, radiochemistry, nuclear engineering, plant engineering and instrument and control/computer calibration, maintenance, and testing. The technical supervisor is responsible for development and implementation of programs and procedures in these technical areas. Reporting to the technical supervisor are the senior results engineer, senior nuclear engineer, instrumentation and control/computer supervisor, and chemistry supervisor.

Staff	Single Unit	Dual Unit
Chemistry Supervisor	1	1
Chemistry Leader	1	1
Chemistry Analysis	2	3
Senior Nuclear Engineer	1	1
Reactor Engineer	1	1
Senior Results Engineer	1	1
Results Engineer	2	3
Engineer	3	5
Nuclear Plant Specialist	1	2
Technical Assistant	2	3
I&C Computer Supervisor	1	1
I&C Foreman	1	1
Instrument Engineer-Leader	3	4
Instrument Engineer	7	9
TOTAL	27	36

The **senior results engineer** supervises the activities of the engineering support staff which consists of results engineers, engineers, nuclear plant specialists, and technical assistants. Activities of the engineering support staff include plant systems engineering support, plant equipment and systems performance evaluation, plant surveillance, procedure preparation and review, failure analysis, test development and implementation, preparation of routine and special reports, maintenance of technical files, initiating engineering recommendations for improved operation, and continued support of plant operations and maintenance.

The **senior nuclear engineer** supervises nuclear engineering activities and is assisted by reactor engineers. Activities of this group include on-site nuclear fuel management and performance activities, planning and coordinating fuel handling, inventory and inspection operations and coordination with the appropriate General Electric department on long-term fuel management. The senior nuclear engineer assists the integrated startup group in coordination of initial criticality, low power physics tests, power ascension tests, and provides technical direction to operations necessary for the safe thermal and nuclear operation of the reactor.

The **instrumentation and control/computer supervisor** supervises the activities of the instrument and control group which consists of an I&C engineer, I&C foreman, instrument engineer-leaders, and instrument engineers. The I&C/computer supervisor is responsible for development and direction of the instrument and control maintenance program for the station. Responsibilities include development of instrumentation and control surveillance procedures, resolution of instrument and control/computer problems, completion of instrumentation and control surveillance testing, and calibration and maintenance of instrumentation and controls, and control of I&C test equipment.

The **chemistry supervisor** supervises the activities of the chemistry and radiochemistry group which consists of chemistry leaders and chemistry analysts. The chemistry supervisor is responsible for activities in the chemistry laboratory, radiochemistry laboratory, and radiological counting room. The chemistry supervisor is also responsible for the development and implementation of station chemistry, radiochemistry, and appropriate programs that provide monitoring of plant processes and discharges.

Quality Organization

The **quality supervisor** reports to the superintendent of plant, assuring the degree of independence necessary to obtain objective verification of quality-related activities. The quality supervisor is assisted by engineers, specialists, and personnel from other sections as assigned. The quality supervisor will be responsible for the verification that on-site quality related activities are accomplished in accordance with the Operational Quality Assurance Program. This verification will be accomplished by review of documents and procedures, inspection, testing, or monitoring of plant activities or an appropriate combination of these methods. The quality supervisor works closely with each section supervisor to provide guidance in implementing QA requirements.

The quality supervisor has stop work authority on any nonconforming activity being performed by plant personnel or subcontractors. These stop work orders may be overruled by the superintendent of plant or assistant superintendent of plant. Activities associated with stop work action are documented and auditable.

The quality supervisor is responsible for keeping the superintendent of plant informed on all quality-related matters.

Staff	Single Unit	Dual Unit
Quality Engineer	1	1
Quality Specialist	1	1

Maintenance Organization

Plant mechanical and electrical maintenance activities are under the direction of the **supervisor of maintenance** who is responsible to the superintendent of plant. The supervisor of maintenance is responsible for close liaison between other station organizations to assure safe equipment operation, shutdown, startup, and functional test, and to assure that work is performed in accordance with station health physics procedures. He is also responsible for the development and periodic review of plant maintenance procedures and instructions pertaining to quality-related work activity of his group. He is responsible for the generation of appropriate maintenance records.

The **foreman — electrical repairs and foreman — mechanical repairs,** reporting directly to the supervisor of maintenance, direct, plan, and coordinate related maintenance activities. Maintenance crews consist of station mechanical leaders, station mechanics, and station repairmen.

Staff	Single Unit	Dual Unit
Mechanical Foreman	1	1
Station Mechanical Leader	2	3
Station Mechanic	5	8
Station Repairman	4	6
Electrical Foreman	1	1
Station Mechanical Leader	1	2
Station Mechanic	3	4
Station Repairman	1	2
TOTAL	18	27

Personnel and Administrative Organization

The **personnel and administrative supervisor** reports to the superintendent of plant and is responsible for activities of the personnel and administrative (P&A) organization.

The P&A organization performs functions in the areas of clerical, records management, training, and plant stores. The personnel and administrative supervisor is responsible for development and implementation of programs and procedures in these areas. The plant training supervisor, administrative assistant, and stores supervisor report to the personnel and administrative supervisor.

- The **plant training supervisor** is responsible for coordinating and scheduling the station training program. Additionally, the plant training supervisor is responsible for development, periodic review, and revision of procedures needed to implement and document the station training program.
- The **administrative assistant** directs clerical activities and Records Management System. The administrative assistant performs duties related to budgeting, data gathering and analysis, cost control, payroll, employment, employee relations and payment of invoices.
- The **stores supervisor** directs the operation of the storeroom and supervises the activities of all storeroom personnel.

The storeroom staff is responsible for requisitioning and receiving incoming materials, supplies and spare parts. They perform clerical work and associated recordkeeping required for handling safety-related parts.

Staff	Single Unit	Dual Unit
Administrative Assistant	1	1
Clerks	6	9
Stores Supervisor	1	1
Material Clerk	2	3
Stockman	1	2
Plant Training Supervisor	1	1
Instructor	1	2
TOTAL	13	19

NORMAL STATION OPERATION

The operating characteristics of the system are established by the program of power testing. Thereafter, normal startup and power generation operation are performed routinely and efficiently by use of procedures.

All plants operate under licenses issued by the NRC. Operating limits are contained in the Technical Specifications in accordance with Federal Regulations (10CFR50.36).

Detailed written operating procedures for all modes of plant operation are prepared prior to the initial startup and critical testing period. Appropriate changes in these procedures are made, according to the experience of the power operation test program. The following is an outline of the principal normal operating procedures which have a potential effect on the safe operation of the plant. This information is presented to indicate the general method of operation.

Cold Startup

The number of control rods which must be withdrawn to reach cold criticality increases as the exposure increases. The control rod withdrawal sequences for approach to criticality are specifically designed to produce an acceptable flux distribution and period anywhere in the sequence.

A procedure for a normal cold startup is summarized as follows. This information is presented to indicate the nature and scope of the program. Details may be modified as the final planning and execution progress.

- Prior to actual startup, a complete and detailed list is verified to ensure that all equipment and systems are in proper condition for operation, that all safety circuitry and interlocks are operative and in service where applicable, that valve and control settings are proper for startup, and that the reactor is properly filled with water. The reactor protection system and the neutron flux level scram sensors are set to operate to protect the reactor in the event of an off-standard condition. The actual startup then proceeds.
- Recirculation, condenser, circulating, and CRD pumps are started.
- Cleanup system is started.
- The reactor is brought to critical by withdrawal of control rods in a prescribed sequence.
- To avoid developing undesirable thermal stresses in the reactor vessel walls, power is adjusted to maintain a specific metal temperature rate of rise limit.
- Turbine is put on turning gear.
- The steam seal regulator is placed in service.
- Vacuum pumps are placed in service.
- Air ejectors are placed in service when sufficient steam pressure is available.
- Feedwater flow and/or blowdown is adjusted as required to maintain the proper water level.
- Warming of turbine valve chest begins at a gage pressure of 600 to 700 psi. (4137 kPa to 4826 kPa.)
- As system pressure increases, the pressure regulator is adjusted upward until the operating pressure is reached.
- The turbine-generator is brought to speed by admitting steam into the turbine.
- The turbine-generator is synchronized and lightly loaded.
- The electrical load is increased to the desired value. This is accomplished by control rod withdrawal and recirculation flow increase to establish the desired steam flow rate.

Hot Startup

Whenever the plant has been shut down for a short period of time and the reactor vessel and auxiliary systems remain at or near operating temperature, a hot startup procedure is followed to return the plant to service. This procedure is essentially independent of the cause of shutdown, assuming that the cause is recognized and all nonstandard conditions have been corrected. The reactor instrumentation is reset and downscaled as required, and a hot startup check list is completed prior to the withdrawal of control rods. The startup proceeds in a manner similar to the normal cold startup procedure outlined previously.

Full Power Operation

Normal power operation of the station, including adjustment to rapid load changes, is controlled by the manual or automatic adjustment of reactor recirculation flow. Also, manual adjustments of control rods are used to shape core power distribution and to change reactor power levels due to daily load requirements.

The main functions of the operating personnel during normal operations are as follows:

- Surveillance of plant equipment for proper functioning and performance of necessary adjustments and repairs. These functions include routine preventive and overhaul maintenance, process-steam sampling, observing and recording information provided by plant instrumentation, routine tests of control rod functioning, and periodic tests of critical station valves and all emergency core cooling systems.
- Manipulation of control rods and manual adjustment of recirculation flow control to accommodate changes in load demand and to optimize flux distribution and fuel burnup while remaining within operating limits.
- Receipt and preparation of new fuel for refueling, and the preparation of irradiated fuel for off-plant shipment. This latter operation includes the removal of fuel bundles from the channels and inspection of the channels before reuse.

- Evaluation of any abnormal conditions and initiation of emergency action as required to minimize the effects of equipment difficulties. Although the safety of the plant is protected by the inherent self-limiting power characteristics of the reactor and by the safety devices provided, alertness of the operating staff ensures proper and safe operation of the plant and continued proper functioning of the automatic features.
- Operating the waste disposal system.

Operational Shutdown

An operational shutdown of short duration can be accomplished while maintaining system pressure. The reactor steaming rate is reduced by control rod insertion. The turbine generator is then removed from the distribution grid in the customary manner. During this shutdown, the reactor will remain critical with fission power reduced to a low level (less than 0.01% of rated unless the decay heat is insufficient to accommodate system heat losses and the steam-operated auxiliaries). Steam is bypassed to the condenser only as required to maintain system pressure.

Isolation Shutdown

The reactor may be isolated from the turbine and balance of plant and yet be maintained at the pressurized condition. The reactor is normally not critical during this type of shutdown. The main steam line isolation valves are closed.

In this condition, core decay heat will initially cause the reactor relief valves to open and blow down to the suppression pool. Reactor water level is maintained by the reactor core isolation cooling system taking suction from the condensate storage tank.* Concurrently, reactor steam at reduced pressure and temperature can be introduced to the residual heat removal system heat exchangers, where it is condensed and ultimately returned to the reactor by way of the reactor core isolation cooling system.

Extended Shutdown

Normal plant shutdowns of long duration, such as those required for refueling and other plant maintenance, are performed in a similar manner. However, system pressure is reduced by progressive readjustment of the pressure regulator set point to ensure controlled cooling. The determining factor is the allowable cooling rate for the pressure vessel and critical equipment. After the turbine is removed from the line, steam is bypassed to the main condenser until the rate of decay heat is sufficiently low to be removed by the shutdown cooling function of the residual heat removal system.

* HPCS system can be considered as as backup to RCIC to provide makeup water in isolated conditions.

Emergency Shutdown

The plant is designed to accommodate abnormal shutdowns such as loss of generator load or turbine trip-out. Depending on the operating condition of the plant and the size of the load rejection, a portion of the steam may be bypassed to the condenser. When the bypass system is not capable of accommodating the excess steam generation, the reactor will automatically scram and the reactor core isolation cooling and HPCS systems will be actuated if required to maintain the vessel water level. Any of the foregoing shutdown modes of operation may be established after the emergency shutdown.

Power Recovery Time Following a Scram

The normal plant operating features that delay achieving power following a scram follow:

- Control rod withdrawal rates
- Cold plant heatup rates
- Turbine loading rates
- Xenon considerations

The exact recovery time, dictated by rod removal rates, is a function of the elapsed fuel cycle time. At the beginning of the fuel cycle, a shorter time could be realized, but toward the end of the fuel life, more rods must be removed to reach power and a longer time would be required.

If the reactor system is to be cooled down, a heating rate may be the limiting factor based on temperature differences existing in particular regions of the pressure vessel. The turbine loading rates are a function of turbine temperatures.

PROGRAM COURSES

The training of a staff to operate a boiling water reactor power station includes special training in nuclear phenomena and fundamentals. General Electric offers a complete training program for either the utility that has purchased its first nuclear plant or the utility that has an experienced nuclear staff. A typical training schedule is illustrated in Figure 8-2.

Operator training includes classroom courses, BWR Training Center courses, and special training programs and services. The courses that make up the standard training program may be taken either individually or in combination, depending upon the trainee's experience or need.

MAINTENANCE

Routine Maintenance

Routine maintenance is undertaken in much the same way as in a fossil-fueled power station. The basic principles under which maintenance is performed are as follows:

- Maintenance is performed in accordance with specific procedures to minimize the possibility of error damage to the components.
- Component and system maintenance records are maintained to facilitate scheduling and completion of all necessary maintenance.
- Maintenance of most facilities outside of biological shielding may be undertaken by direct contact during station operation.
- To permit maintenance of such major components as the reactor, or a recirculating water pump, a complete nuclear unit shutdown is required.
- Radiation monitoring is used during the initial approach to potentially radioactive equipment, and monitoring continued periodically during disassembly by use of portable monitors.
- Protective clothing is worn and other measures are observed as necessary to avoid the spread of radioactive materials.
- Components are decontaminated, as necessary, prior to performance of maintenance. Such decontamination measures assist in reducing exposure to personnel during the ensuring work.

Nonroutine Maintenance

Nonroutine maintenance is scheduled as necessary during periodic shutdowns of the station. Such shutdowns are scheduled on the basis of system requirements, the safety of continued operation, and the availability of maintenance manpower.

Maintenance Summary

A summary of the most important functions is given below. In the case of conventional components, such as pumps and heat exchangers, it is recommended that normal maintenance practices be followed. Usually, this work is accomplished during regular plant shutdown for refueling.

Other equipment is serviced as required by normal operating practice. As a general rule, primary equipment is serviced during refueling outages; auxiliary equipment would be serviced while the plant is operating.

Item	Maintenance
Recirculation pump shaft seals	Inspect at each refueling and replace if required.
Power range in-core flux monitors	Replace as necessary during refueling outage.
Control rods	Replace as necessary during refueling outage.
Fuel channels	Replace as necessary at operational convenience between refueling outages while plant is operating or during refueling.

RELIABILITY TESTS

Reliability tests are routine verifications of system or equipment performance to give added assurance that they will function as designed when required by plant operating conditions. Many of these tests can be conducted without interrupting plant operations, although some variation or reduction in power output may result during the test. Some examples are control rod drive movement, average power range monitors trip verification, and primary steam isolation valve operation.

Other systems can only be tested with the plant shutdown. Such tests are conducted on a planned basis during refueling outages. Some examples are liquid control system valve operation and scram dump tank high level trip.

REACTOR EQUIPMENT MAINTENANCE

The reactor internal equipment is designed so that most components can be easily removed. Exceptions are the upper core grid and the lower core plate. These items are structural members of the core assembly and normally would not be removed as a maintenance function. However, they may be removed in the reverse

MINIMUM RECOMMENDED AND REQUIRED COURSES

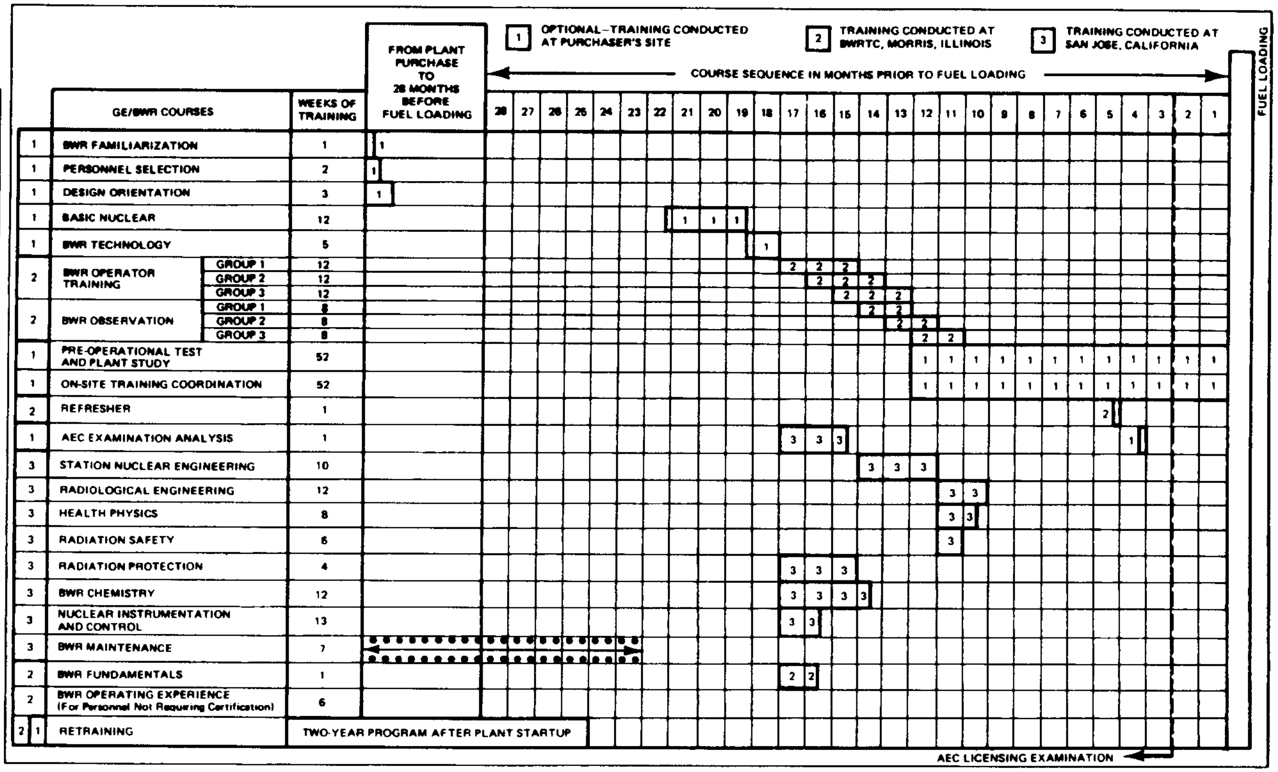
Course	4	5	6
REACTOR FUNDAMENTALS	4	5	
BWR TECHNOLOGY	4	5	6
BWR OPERATOR TRAINING	4	5	
BWR OBSERVATION	4	5	
STATION NUCLEAR ENGINEERING	4	5	6
BWR STARTUP TESTING		5	
BWR CHEMISTRY	4	5	6
RADIOLOGICAL ENGINEERING	4	5	
NUCLEAR INSTRUMENTATION AND CONTROL	4	5	6
BWR MAINTENANCE		5	
BWR OPERATING EXPERIENCE			6
REFRESHER (SIMULATOR)		5	
AEC EXAMINATION ANALYSIS		5	
BWR FUNDAMENTALS	4	5	6
RETRAINING	4	5	6
DESIGN ORIENTATION			
RADIATION SAFETY			
HEALTH PHYSICS			
RADIATION PROTECTION			
BWR OPERATIONAL PERFORMANCE			
BWR FAMILIARIZATION			
PERSONNEL SELECTION			
ON-SITE TRAINING COORDINATION			
VIDEO TAPES			

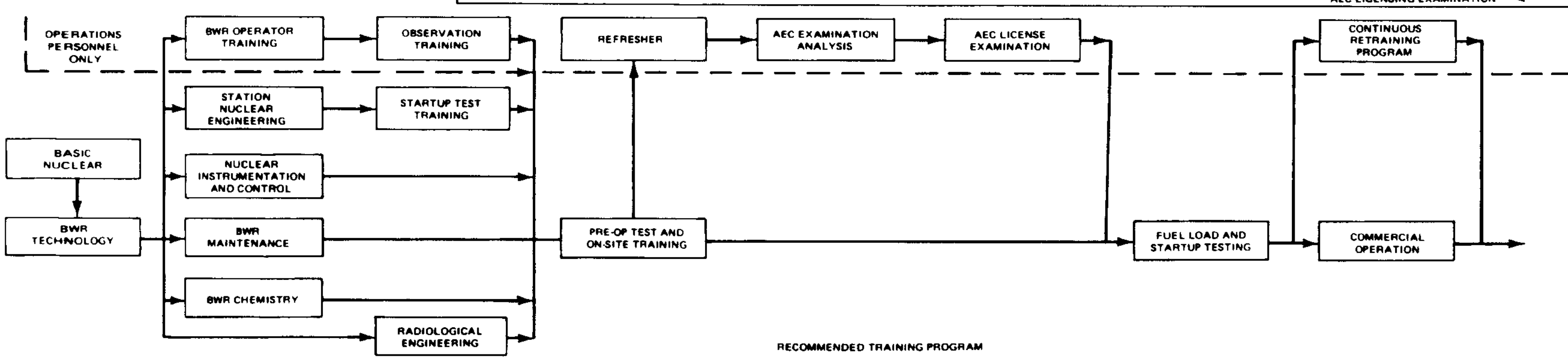
4 PROGRAM NECESSARY FOR THE UTILITY WHICH HAS PURCHASED ITS FIRST NUCLEAR PLANT

5 PROGRAM RECOMMENDED BY GENERAL ELECTRIC FOR THE UTILITY WHICH HAS PURCHASED ITS FIRST NUCLEAR PLANT

6 MINIMUM PROGRAM FOR THE UTILITY WITH NUCLEAR EXPERIENCE WHICH HAS PURCHASED ITS FIRST BWR AND PLANS TO STAFF IT WITH OPERATORS ELIGIBLE FOR AN AEC COLD LICENSE EXAMINATION

THESE COURSES AND SERVICES ARE DESIGNED FOR A SPECIFIC NEED. A COMPLETE TRAINING PROGRAM WILL SELECTIVELY INCLUDE VARIOUS PORTIONS OF THESE.





RECOMMENDED TRAINING PROGRAM

Figure 8-2. Typical Training Program

procedure of their installation or major modification work. The core shroud and the diffuser section of the jet pump are permanent parts of the vessel.

Although the probability of damaging reactor components is very low, the potential does exist because of the physical movement of components in the sequence of refueling or control rod replacement. Consequently, the need for convenient replacement is recognized and has been incorporated into the design of the fuel support pieces, guide tubes, orifices, rod drives, and jet pumps.

The fuel assemblies, control rods, and in-core flux monitoring assemblies are items that have a limited in-reactor exposure life. The design of these items has included, to a high degree, the concept of routine replacement. The following paragraphs are a general description of the procedures for control rod, rod drive, and in-core flux monitor replacement.

Control Rod Replacement

Control rod life depends on the time the control rod spends in the high flux region of the core. When the recorded irradiation history of the control rod indicates replacement is required, the fuel assemblies surrounding the rod are removed. The control rod is withdrawn and the fuel support casting is removed. The control rod is detached from the drive mechanism by actuating a spring-loaded coupling. The control rod is then moved to the holding pool for transfer to the fuel building storage pool. Irradiated control rods must be shielded, but the depth of water required is less than that for fuel.

Control Rod Drive Mechanism Removal and Replacement

It is expected that infrequent drive maintenance will be required to inspect wear surfaces and replace seals and other worn parts. Drive removal and replacement are performed in the space below the reactor vessel during plant shutdown. The control rod is withdrawn from the core until a seal surface on the rod coupling closes off the top of the control rod drive housing. (A penetration seal is used to seal from above if the rod backseat is ineffective.) The coupling is actuated either from above the reactor, if the vessel head has been removed, or from below the reactor by using an activating tool which is attached to the drive. Water in the drive is drained.

The mounting and removal of the control rod drives with the reactor vessel mounting flange and the transport of the drives in the under reactor vessel area are performed using power assisted equipment. A platform, mounted on circular track secured to the inner surface of the reactor pedestal, provides for the operation of the drive handling equipment and for the movement of the drives to the required under vessel location. (Targets, corresponding to the location of each control rod drive reactor vessel mounting flange, are located on the under vessel area floor for this purpose.) The drive handling equipment includes a cart which mounts on rails located on each side of a center slot in the platform, an elevator mechanic (which is part of the cart) and other power operated features.

The removal of a control rod drive is reverse of the mounting of a drive. Control rod drives enter the under-vessel area secured to the cart. Once the drive is at its intended location by rotating the platform and moving the cart on its rails, the drive is rotated from the horizontal to the vertical, aligning the elevator with the target on the floor. The drive is raised to meet the vessel mounting flange. A torquing tool is raised to secure the drive mounting bolts. The operation of the handling equipment is from a control station located on the equipment handling platform.

Power Range In-Core Neutron Monitor Servicing

The lifetime of the power range neutron monitors varies with the reactor neutron flux level at a particular core location. The power range detection assembly consists of multiple dry tube assembly and four neutron detectors. The assemblies lie in the interstice of four fuel assembly groups that are enclosed by four control rods. The flux monitor assemblies need not be disturbed during refueling.

Removal of the individual detectors is accomplished by automatic or manual winding into a shielded disposal cask. Installation of a detector is accomplished manually by insertion until the fitting is in position on the dry tube. When the fitting is tightened, the mechanical installation is complete. The signal cable is connected to the mating connection of the detector and the sensor is ready for use.

Section 9

Refueling Equipment and Procedures

INTRODUCTION

The BWR/6 is designed to provide fast and efficient refueling to minimize refueling outage time. The total refueling system is based on extensive experience with a large number of operating BWR's.

This section describes the refueling tasks as well as the refueling tools required to support a refueling of the nuclear boiler in a Mark III containment. The refueling concept shortens the reactor-related operations through the use of several series of operations: fuel handling over the reactor in the upper containment pool, fuel transfer between the upper pool and fuel building transfer pool, and fuel handling into the storage pool in the fuel building. The containment pool has a fuel holding capacity equal to 25% of initial core load, facilitating ease of the refueling operation.

The design provides for fuel preparation and storage in the fuel building and incorporates an inclined fuel transfer tube to effect transfer of the fuel between the upper containment pool and the fuel building pool.

The reactor building refueling floor is shown in Figure 9-1. The storage space allotted for the various equipment is outlined. A watertight gate separates the fuel storage and transfer area of the upper containment pool from the reactor well so that the reactor well can be drained when opening the reactor vessel.

The fuel building accommodates a dry new-fuel storage vault, a new-fuel inspection stand, a transfer-storage pool, a fuel storage pool, and a spent fuel cask shipping area.

An annual refueling outage includes these major tasks:

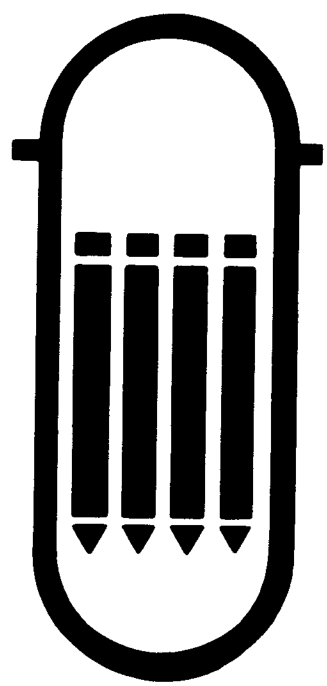
- Discharge 25% of the center core fuel assemblies
- Shuffle 25% of the core from outside-in
- Reload 25% of the peripheral locations with previously channeled new fuel

REFUELING TASKS

The refueling tasks are described in the following paragraphs.

Reactor Shutdown

The reactor is shut down according to normal station shutdown procedures (see Section 8). During cooldown, the reactor pressure vessel is vented and filled to



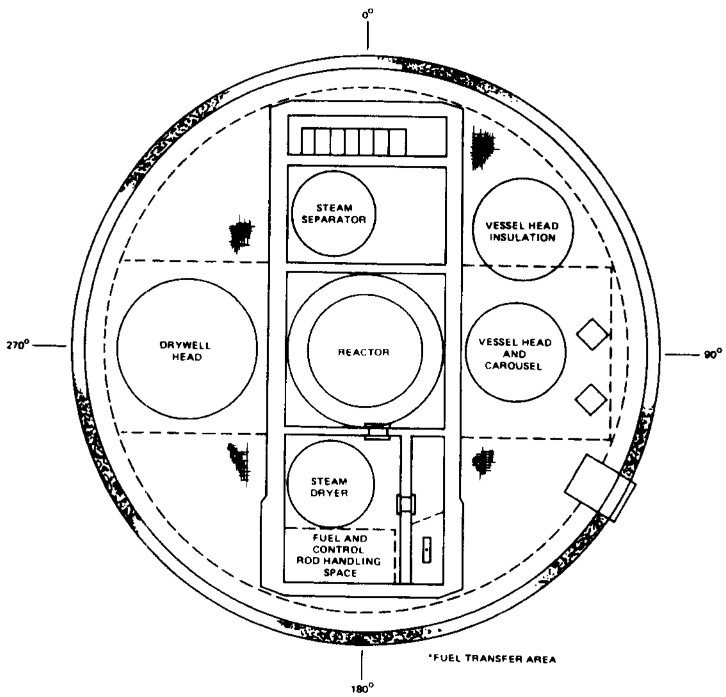


Figure 9-1. Reactor Building Refueling Service Floor Laydown Areas

above flange level to promote cooling. The drywell head cavity is drained and drywell head bolts removed during this time in preparation for head removal.

Drywell Head Removal

The drywell head typically utilizes swing bolts, which are loosened to clear the head and swing outward to lean on the reactor well wall. The unbolted drywell head is lifted by the polar building crane to its storage space on the refueling floor. The drywell seal surface protector is installed before any other activity proceeds in the reactor well area.

Vent Pipe and Vessel Head Insulation Removal

With the drywell head removed, an array of piping is exposed that must be serviced. Various vent piping penetrations through the reactor well must be removed and the penetrations made watertight. Vessel head piping is stored on the head insulation and a common lift is used to transport them to storage on the refueling floor.

Water level in the vessel is now lowered to flange level in preparation for reactor vessel head removal.

Reactor Vessel Head Removal

The combination of head strongback, and carousel stud tensioners is placed over the vessel head. The stud with multiple tensioners are used to loosen successive groups of head nuts until all are removed and stored in racks in the carousel. The head is attached to the strongback. The head, strongback, and carousel are transferred by the reactor building crane to the head holding pedestals on the refueling floor, keeping the vessel head elevated to facilitate inspection and O-ring replacement. The vessel studs in line with the fuel transfer canal are removed to provide a path for fuel movement.

Steam Dryer Removal

The dryer-separator strongback is lowered into the vessel opening and attached to the dryer lifting lugs. The dryer is lifted from the reactor vessel and transported underwater to its storage location in the dryer storage pool.

Steam Line Plug Installation

Prior to removal of the steam separator, the steam line plugs are installed in the four main steam nozzles from inside the vessel. This is accomplished using the installation tool suspended from a chain hoist attached to the building crane. The plugs are hand-guided into place in the steam nozzles and inflated. With these plugs in place, servicing of the safety/relief valves or main steam isolation valves can be accomplished without interference or holdup to refueling operations.

Steam Separator Removal

In preparation for the separator removal, the separator is unbolted and unlatched from the shroud, using the four shroud head bolt wrenches. The dryer-separator strongback is lowered into the vessel and attached to the separator lifting lugs. The separator is then transferred underwater to its storage place in the pool.

Fuel Bundle Sampling

During reactor operation, the core off-gas radiation level is monitored. If a rise in off-gas activity has been noted, the reactor core may be sampled during shutdown to locate any leaking fuel assemblies. The fuel assembly sampler is used to obtain water samples from the in-vessel fuel assemblies. The sampler permits the isolation of each fuel assembly in clusters of four for the taking of water samples to detect and identify failed fuel. Up to 16 fuel assemblies can be sampled at one time using 2 cabinets with a total of 4 heads, one cabinet on each side of the reactor wall. Suspect fuel bundles will be examined later for proper disposition.

Refueling

The gate isolating the fuel transfer area is removed, thereby interconnecting the reactor well and the fuel transfer area.

The refueling platform (Figure 9-2) is the principal device to handle and transfer fuel assemblies in and out of the reactor. To move fuel, the fuel grapple is aligned over the fuel assembly, lowered, and attached to the fuel bundle bail. The fuel bundle is raised out of the core, moved through the refueling slot, positioned over the storage rack, and lowered into the rack. Fuel is shuffled, and new fuel is moved from the storage racks to the reactor vessel in the same manner.

Concurrent with fuel shuffling, spent fuel assemblies are transferred from the containment pool storage racks to the fuel pool storage racks in the fuel building. Any identified defective fuel assembly is placed in a defective fuel container.

FUEL TRANSFER SYSTEM

An inclined transfer tube is used for the transfer of fuel assemblies, control rods, or other irradiated items between the fuel building pools and the upper pool in the containment (see Figure 9-3). The transfer tube assembly includes the transfer tube itself, a tilting container and carrier for two fuel assemblies that moves up and down inside the tube, upper and lower tilting mechanisms, and a mechanical transfer mechanism and associated controls.

The transfer tube is a 24 inch (0.61m) i.d. stainless steel tube encased by a carbon steel guard pipe. The tube is made up of sections joined by couplings and sealed against internal pressure. Valves at each end of the transfer tube keep the upper pool from draining through the transfer tube. Bellows provide a flexible seal to permit relative movement between buildings for settling or seismic displacement.

A blind flange mounted over the tube provides the required containment isolation during plant operation.

A carriage shuttles between the containment and fuel building pools. Individual inserts adapt the carriage to transfer fuel, control rods or other in-core components. A winch and dual cables are provided for moving the carrier through the transfer tube. Tilting mechanisms powered by hydraulic pistons rotate the container to the vertical position for loading and unloading of fuel bundles and other in-core components. Interlocks prevent rotation of the container unless the carrier and auxiliary or refueling platform are in the correct position.

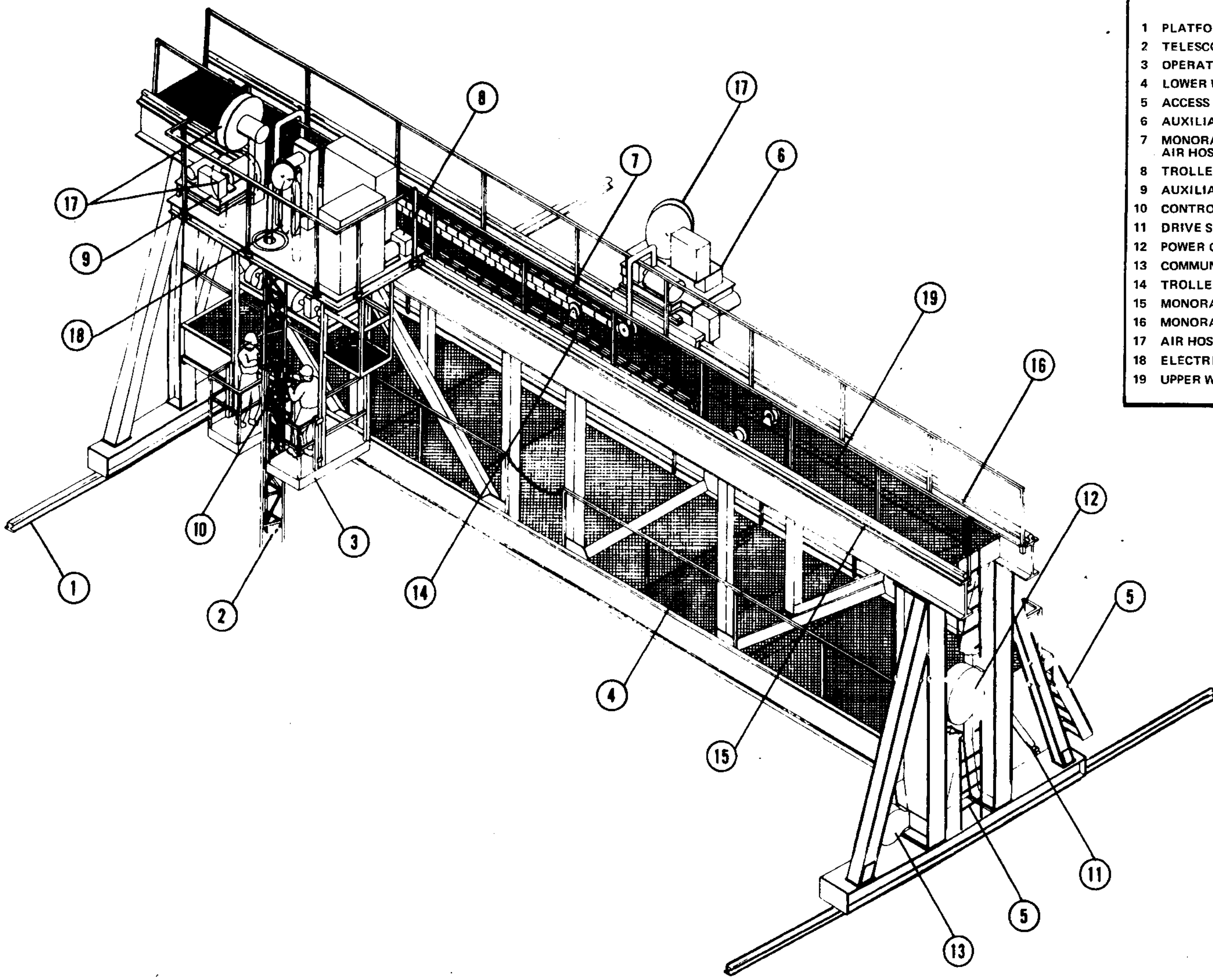
Control of the transfer system is from two control stations, one on the operating floor in the reactor building near the upper end of the transfer tube, and the other on the operating floor in the fuel building adjacent to the transfer pool and near the lower end of the transfer tube. Controls, instrumentation, and communications needed to operate and monitor the transfer system are grouped at these two locations.

The sequence of operations (some of which are automatic) to perform a transfer from the containment pool to the transfer pool in the fuel building (see Figure 9-2) is outlined as follows:

- Load fuel assembly into tilting tube in its vertical position.
- Tilt tube to the inclined position.
- Lower carriage to position just above lower valve in transfer tube and close flapper valve.
- Open exit valve and drain the water in the tube. (Water exit valve closes automatically.)
- Open the lower valve.
- Lower the carriage to its lower limit in the fuel building transfer pool.
- Tilt tube to the vertical position.
- Remove fuel assembly.

With the carriage as depicted in the lower pool,

- Unload the tilting tube and insert new fuel.
- Tilt tube to inclined position.
- Hoist the carriage to a position inside the transfer tube, just above lower valve.
- Close the lower valve.
- Open the water inlet valve and fill the transfer tube.
- Hoist carriage to upper limit in reactor building transfer pool.
- Tilt tube to a vertical position.
- Unload fuel assembly and transfer directly to core or to fuel holding pool in containment.



REFUELING PLATFORM

1 PLATFORM RAIL
2 TELESCOPING FUEL GRAPPLE
3 OPERATOR'S BASKET
4 LOWER WALKWAY
5 ACCESS LADDER
6 AUXILIARY HOIST (MONORAIL MOUNTED)
7 MONORAIL AUXILIARY HOIST ELECTRICAL CABLE AND AIR HOSES (POWER TRACK)
8 TROLLEY DRIVE
9 AUXILIARY HOIST (TROLLEY FRAME MOUNTED)
10 CONTROL CONSOLE
11 DRIVE SHAFT
12 POWER CABLE REEL
13 COMMUNICATION CABLE REEL
14 TROLLEY ELECTRICAL CABLE AND AIR HOSES (POWER TRACK)
15 MONORAIL
16 MONORAIL FOR AUXILIARY HOIST
17 AIR HOSE REEL
18 ELECTRICAL CABLE REEL
19 UPPER WALKWAY

Figure 9-2. Refueling Platform

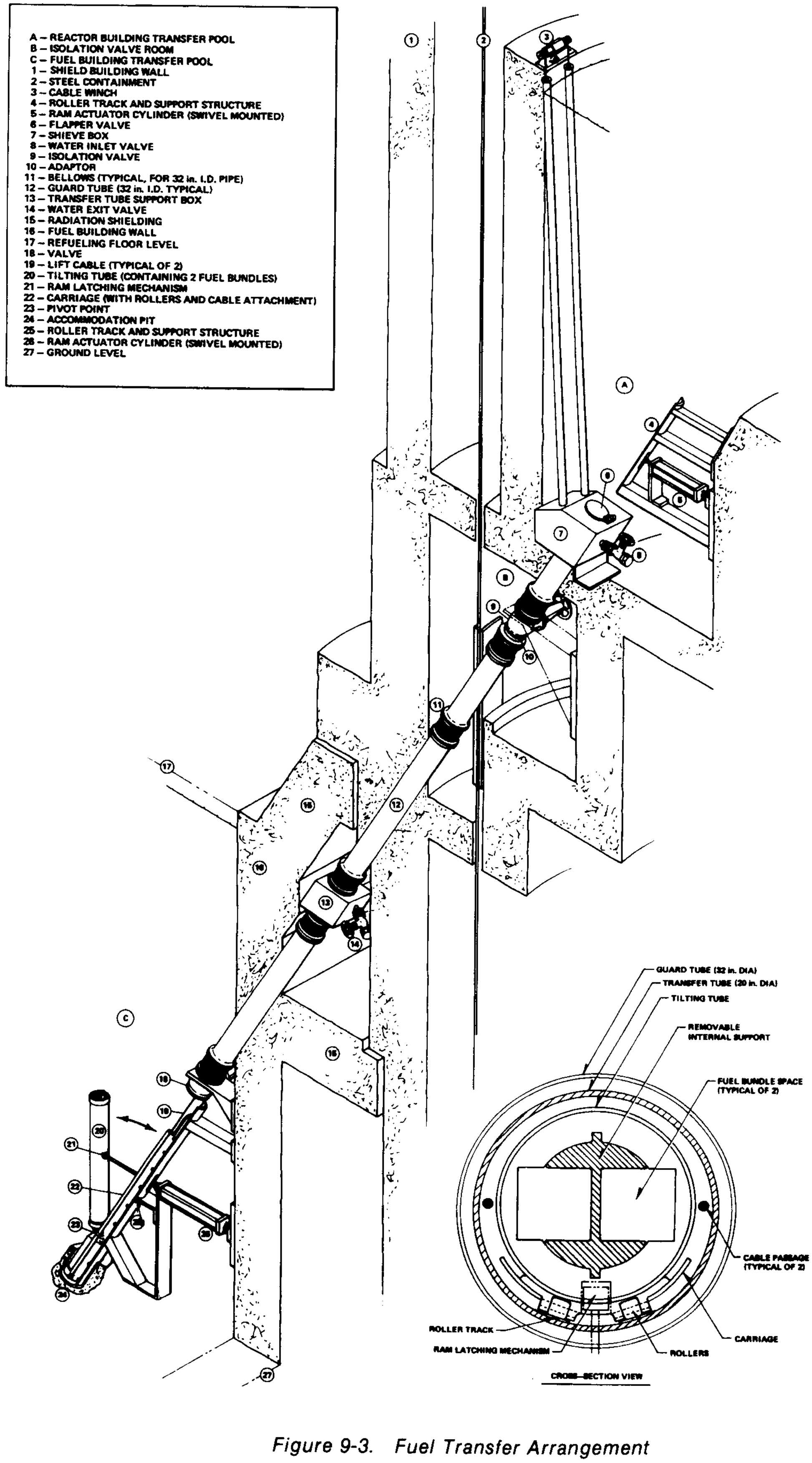


Figure 9-3. Fuel Transfer Arrangement

REFUELING EQUIPMENT

Refueling Platform

The refueling platform (Figure 9-2) is used to transport fuel assemblies in and out of the reactor. It also is equipped with two auxiliary hoists to handle and transport other core pieces, and it provides the operator with a work surface for most other servicing operations. The platform travels on tracks extending along each side of the upper containment pool. The fuel grapple is mounted on a trolley that can traverse the width of the platform.

The platform control system permits variable-speed, simultaneous operation of all three platform motions. One operator can control all the motions of the platform required to handle the fuel assemblies during refueling. Interlocks on both the grapple hoist and auxiliary hoists prevent hoisting of a fuel assembly over the core with a control rod withdrawn; interlocks also prevent withdrawal of a blade with a fuel assembly over the core attached to either the fuel grapple or hoists. Interlocks block travel over the reactor in the startup mode.

A digital position indication system is included. The readout, in the operator's cab, matches the core arrangement cell identification numbers and permits very accurate positioning of the refueling grapple.

Auxiliary Platform

The auxiliary platform is a low silhouette platform which rides on rails inside of those used for the refueling platform. It is used with the refueling platform for concurrent vessel or pool work and will accommodate an auxiliary hoist for general purpose work. The auxiliary platform may be lowered to the vessel flange to facilitate handling of jet pumps or other internal components.

Fuel Handling Platform

The fuel handling platform is a similar device mounted over the fuel building pools. The platform has a motorized trolley to which the grapple is mounted. The grapple is similar to that described for the refueling platform. An operator's bridge is suspended from the trolley. The combination of the bridge movement for the length of the pools and the trolley movement for the width of the pools provides complete access to these pools.

Fuel Pool Jib Crane

This jib crane is used in the fuel building for tranferring new and spent fuel assemblies between the storage racks and fuel preparation machines located in the fuel transfer pool.

Fuel Assembly Sipper

The fuel assembly sipper is used to sample suspect fuel assemblies in the fuel pool. It is used to evaluate and confirm the results obtained from the in-vessel fuel bundle sampler. The sipper circulates water in the fuel assembly and through the fuel bundle, permitting the desired buildup to isotopic concentration for sampling and testing. The evaluation is made in the fuel storage pool.

Miscellaneous Equipment

A complement of other equipment is used to facilitate the refueling process. These are items such as slings, grapples, actuating poles, wrenches, storage racks, lights, viewing aids, an underwater TV system, and vacuum cleaner.

NEW FUEL PREPARATION

New fuel is delivered to a receiving station within the fuel building. Each crate holds an inner metal container which contains two fuel bundles. Handling during uncrating is normally accomplished by the fuel building crane, extending down from the servicing floor through the equipment hatch. The inner container is lifted up through the equipment hatch to the servicing floor. The container is supported in a vertical position while the fuel bundles are unstrapped and then lifted into the new fuel storage racks located in the new fuel storage vault.

Inspection of the new fuel is normally deferred until all the reusable metal containers are emptied and the area around the new fuel vault is cleaned. The individual fuel bundles are then removed from the vault, inserted in the new fuel inspection stand, dimensionally and visually inspected, and returned to the storage vault to await affixing of channels.

New Fuel Inspection Stand

The new fuel inspection stand located in the refueling building is a frame structure which holds two fuel bundles in the vertical position adjacent to each other. The lower socket receptacle for the fuel bundle permits the rotation of the bundle. The upper receptacle is a retaining clamp to support the fuel bundle in the vertical position. A motor-operated lift raises and lowers a U-shaped working platform for inspection of each fuel bundle over its entire length.

Channeling New Fuel

Two fuel preparation machines are located in the fuel building pool: one for dechanneling spent fuel, and the other for channeling new fuel. New fuel bundles without channels are unloaded from the new fuel vault and transported to the fuel racks in the fuel pool. A spent fuel bundle is transported to the fuel preparation machine, using the fuel handling platform. The channel is unbolted from the bundle, and the channel handling tool is fastened to the top of the channel. The fuel preparation machine carriage is lowered, removing the fuel from the channel. The channel is then positioned over a new fuel bundle located in the second fuel preparation machine, and the process reversed. The channeled new fuel is stored in the pool storage racks ready for transfer and insertion into the reactor.

The preferred method is to channel a reload batch of fuel while the reactor is operating to ensure that the fuel is ready for use during the refueling operation. New fuel can then be transferred into the containment pool during reactor cooldown and refueling preparation tasks. Spent fuel can be transferred from the containment pool to the fuel building during vessel reassembly after refueling.

Dechanneling should immediately precede channeling, for maximum efficiency. Channel storage adaptors can be used to permit irradiated channels to be stored in the fuel storage racks. These adaptors fit in the fuel racks as spacers and allow the top of the stored channels to project above the top of the racks for convenient handling. A channel transfer grapple is used to transport channels.

Unirradiated spare channels are normally stored in the vacant racks in the new fuel storage vault. These channels would be removed only as needed to complete the assemblies required for the next refueling outage.

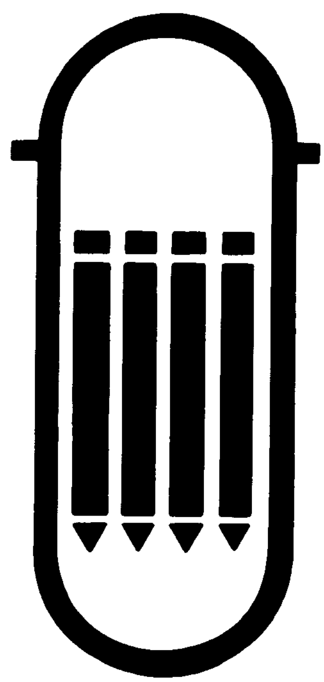
A wall-mounted channel accumulation rack is located between the two fuel preparation machines. The use of this rack permits some lag to occur between the dechanneling and channeling operations.

FUEL STORAGE AND SHIPMENT

Spent fuel removed from the reactor is stored in the fuel storage pool in the fuel building. High density fuel storage rack capacity to hold a normal discharge batch of fuel bundles plus a full core is normally available as a minimum storage capacity.

Spent fuel is stored for a time as determined by the schedule for shipment to the fuel reprocessing facility. For such shipments, fuel bundles are loaded into special design shipping casks which are generally made available by the fuel processor. The heavy casks, capable of holding many fuel bundles, arrive on special trucks or railroad cars which are brought into the car bay of the fuel building. The cask is lifted from the vehicle into the cask loading pool by the fuel building cask crane. This crane cannot traverse the spent fuel storage areas. Once fuel is loaded into the cask, it is sealed and lifted back onto the transport vehicle for its return trip to the reprocessing facility.

Section 10
Preoperational and Startup Test Programs



INTRODUCTION

The objectives of the Preoperational and Startup Test Programs are as follows:

- Verify that all equipment and systems meet design and operational requirements prior to fuel loading.
- Demonstrate that the NSSS and its associated emergency core cooling system meet all stated criteria and operational requirements over the full range of temperature, pressure, and power.
- Demonstrate that the reactor and all associated control systems meet stated criteria on stability, performance, and operability in all operating modes.
- Demonstrate that the NSSS meets or exceeds warranted performance.
- Demonstrate that all startup and surveillance tests can be performed within the NRC Technical Specifications as well as normal routine operation.
- Control and document the testing activities and create a permanent record of the results.
- Allow utility operating and technical personnel to obtain detailed knowledge of plant performance characteristics.

ORGANIZATION

The preoperational and startup testing is performed by plant personnel, with technical direction from General Electric Startup and Training personnel. Preoperational testing, with site technical direction by Startup Test Operations personnel, usually begins about 10 to 12 months before fuel loading. Startup testing ends with the successful completion of the steam output demonstration test, usually a 100-hour run of continuous operation at rated conditions.

TEST DOCUMENTS

As part of the startup technical direction, the following documentation is used:

- Preoperational test program specifications for equipment supplied by General Electric
- Preoperational test instructions for equipment supplied by General Electric
- Startup test program specifications
- Startup test instructions

The specifications define the minimum test programs which General Electric considers necessary to meet the objectives noted earlier. They include brief descriptions, criteria, and a suggested sequence for performing the tests. The instructions are primarily working documents to be used by operating and technical personnel performing the test work. They contain step-by-step instructions in enough detail for knowledgeable personnel to perform the tests; they also include data sheets, methods of data analysis, results of predictive analyses made in San Jose, and other supporting information.

PREOPERATIONAL TESTS

A program of preoperational testing of all equipment and systems is carried out to verify that the performance of plant systems meets design criteria and operational requirements and that the plant is ready for fuel loading and initial startup. This program includes the verifications, adjustments, and operations necessary to ensure that initial loading and subsequent operation can be safely undertaken. System components are tested, logic checks are performed, and sensor set points are verified. The system is then tested as a whole, to the extent feasible, to verify that the system performs in conformance with the design. For example, the system tests include, as appropriate, response and sensitivity measurements, wiring verification, air or hydraulic flow tests, equipment interlock checks, and vibration measurements. Finally, integrated functional tests are performed to check the interaction of related systems in response to simulated plant signals.

For systems that require elevated temperatures, pressures, or high pressure steam flows to complete a system checkout, preliminary checks are made in the cold condition, and final testing is accomplished during the initial nuclear heatup or power ascension portions of power testing.

The following is a brief summary of the preoperational test program. Four major system pre-op tests are described briefly; others are simply listed to indicate the nature and scope of the program for the equipment normally supplied by General Electric. Approximately 75 to 100 pre-op tests are also required on equipment supplied by others.

Reactor Water Cleanup

The vessel is cleaned and filled with high purity water and the reactor cleanup system is placed in operation to maintain water quality. In addition, all the process instrumentation required during the startup and "open vessel" testing is placed in operation.

Nuclear System Instrumentation and Control

Each primary element, transmitter, receiver, indicator, control element, and any other instrumentation device is inspected. Any damaged or inaccurate element is repaired or replaced. All interconnecting piping and wiring is verified. Each device is adjusted and calibrated in accordance with the manufacturer's recommendations. All control devices are exercised to ensure proper operation with the accuracy and response characteristics required by the system and specified by the manufacturer.

Reactor Protection System Circuits

Each circuit of the reactor protection system is tested to verify that a scram signal will initiate a reactor scram. For example, the reactor vessel pressure sensors in the reactor protection system as individually tested by varying a simulated pressure signal to the sensors. As a signal level corresponding to a scram condition is reached, scram functions and annunciation should appear for the particular channel under test. In this manner, all scram sensors except neutron flux sensors are tested.

Each intermediate and average power range monitor is tested to verify the scram function of these devices by an electronically generated signal. All reactor protection system coincidence features and the "fail-safe" feature of each component of this system are also tested for proper action.

Control Rod Drive System

The complete control rod drive hydraulic system (accumulators, scram valves, exhaust header, piping, pressure regulating valves, and interlocks) is tested without actuation of the drive mechanism. The control rods are then attached to the drives, and rod guides are installed to test each rod drive and rod assembly individually for shim control and scram operation. The operation is repeated as often as necessary to ensure that all the control rods are reliably operative.

Other Preoperational Tests

The following is a list by title of other preoperational tests performed:

Feedwater Control
Standby Liquid Control (SBLC)
Nuclear Boiler
Residual Heat Removal (RHR)

Reactor Core Isolation Cooling (RCIC)
Reactor Recirculation System
Fuel Handling and Vessel Servicing Equipment
Low Pressure Core Spray System (LPCS)
High Pressure Core Spray (HPCS)
Nuclear System Leak Detection
TIP Calibration System
Rod Control and Information System
Process Radiation Monitoring
Process Computer Interface
Containment Isolation

FUEL LOADING AND OPEN VESSEL TESTING

The preoperational testing program ends and the startup testing program begins when all items necessary for fuel loading have been completed. The general procedures to be followed during this period are outlined in the paragraphs which follow. Special attention is given to matters having a potential effect on nuclear safety.

At the start of fuel loading, all necessary equipment is in operable condition, and the reactor vessel is filled with water to the normal level. Normal refueling equipment and techniques are used for the installation of fuel. Minor modifications in the normal fuel handling procedures may be made since the fuel has not been irradiated. For example, the fuel pool and the pool above the reactor vessel need not be filled, and thus new fuel may be transported directly to the reactor since it can be handled more efficiently in this manner.

Operational neutron sources are installed in the vessel before fuel loading starts. These are Sb-Be sources that produce neutrons by the γ-n reaction. They are continually regenerated by reactor operation. The sources have sufficient neutron emission to provide significant readings on the special neutron-sensitive chambers used for initial fuel loading at all stages of the planned loading.

An Am-Be source is installed in the vessel before loading starts. This source produces neutrons by the α-n reaction. The source has a sufficient neutron emission rate to provide significant readings on the special neutron-sensitive chambers used for initial fuel loading at all stages of the planned loading.

The special neutron-sensitive chambers for initial fuel loading are installed in the vessel around the periphery of the volume to be loaded. These instrument channels are capable of initiating a reactor scram if reactor power exceeds preestablished limits.

All low level nuclear instrumention channels are tested for response to neutron flux, using the neutron sources, prior to the start of loading and/or at appropriate intervals during loading.

The following scram sensors must be operative:

- High water level in scram discharge volume
- Manual scram
- High flux on any of the flux monitors connected to the reactor protection system (SRM, IRM, APRM)
- Low water level in the reactor vessel

The standby liquid control system is available during loading and critical testing. At every stage during the initial loading and in the fully loaded configuration, the control rod system must provide the minimum cold shutdown margin.

No special tests are required to verify the nuclear characteristics of fuel assemblies and control rods. The BWR is characterized by the stability of its standard core design and core component design, and the extensive confirmation of the analytical methods applied in the design process. A manufacturing quality assurance program is provided to establish that the core components are fabricated to meet the physical properties specified. The core is loaded in prescribed steps, with a simple test performed for adequate subcriticality before and after each loading step. This also functionally tests each control rod and drive. The shutdown margin is measured or it is demonstrated that shutdown criteria are met.

HEATUP AND LOW POWER TESTING

After fuel loading is completed, the reactor is brought critical to verify adequate response of the startup range monitoring and intermediate range monitoring nuclear instrumentation systems. These systems are initially calibrated from nuclear data.

Following the open vessel tests, the reactor vessel is closed up and the system and plant brought to a condition of readiness for power operation (see Section 9, Refueling Equipment and Procedures). The nuclear system is hydrostatically pressurized and checked for leaks prior to starting the heatup test program.

- The recirculation pumps are operated to determine base flow rates and pressure drops in the cold condition.
- The reactor is brought critical again and system heating is begun; the intermediate range monitor-

ing and average power range monitoring systems are calibrated, based on thermal data. System pressurization begins when water temperature increases above 212°F (100°C).

- Reactor pressure vessel surface temperature, process temperatures, and thermal expansion are measured.
- The control rod drive system is tested at various intermediate pressures up to rated pressure.
- Radiation surveys are begun.
- The vacuum, condenser, and off-gas systems are brought into operation and verified.
- Reactor auxiliaries are operated and verified as appropriate (cleanup system, condensate systems, feedwater system, etc.).
- Operability of the reactor core isolation cooling and high pressure core spray systems are verified. The objective is to determine the ability to start from a cold condition and reach rated flow within the specified time.
- The turbine is rolled as soon as steam production is sufficient.

POWER OPERATION TEST PROGRAM

The reactor and plant are brought to full power in incremental steps. Comprehensive testing of the operational, transient, and safety characteristics of the nuclear system and the plant must be completed successfully at each line before power output is increased to the next higher line. Typically, the tests are performed at 25%, 65%, and 100% of full power rating.

This power test program, as further described below, is designed to demonstrate not only that the plant is properly designed and instructed and can be operated as reliable power producer in the utility system, but also that all of the operating safety provisions required by NRC regulations are functioning. This power test program, required of all light water reactors, is a vital but not highly publicized safety assurance program.

Ascent To Rated Power

- Power calibration of nuclear instrumentation by heat balance technique. The objective is to improve the power calibration accuracy of the nuclear instrumentation in the safety system. Conventional heat balance techniques are used.
- Radiochemical sampling and analyses are begun. The objectives are to establish a base condition from which to measure changes and to check out laboratory procedures.
- Local power range monitor (LPRM) calibration. The objective is to calibrate the power distribution instrumention system. The calibration procedure begins with flux measurements by moveable chambers (TIP) and results in a set of correction factors used to reset the LPRM amplifier gains.
- Core performance evaluations are made at each significantly different power, rod pattern, flow, or subcooling. The objective is to determine the core power distribution using the LPRM system and, in particular, to find the maximum heat flux and the minimum critical power ratio (MCPR) at each changed condition. Plant instrumentation and conventional heat balance techniques are used.
- The reactor relief valves are opened one at a time. The objective is to verify designed capacity and operating characteristics of each valve and check for proper reseating after operation.

Small Perturbation Tests

These tests are performed at each power flow plateau to test the dynamic response and verify stability. Control system adjustments are made when required. The objective is to demonstrate that the reactor response is well damped. Pressure power, core flow, and power distribution data are recorded.

- The pressure set point of the pressure regulator is moved quickly to ±10 psi (±69 kPa).
- A control rod is moved continuously (±) an amount sufficient to change the local power (measured by local in-core flux monitor) less than 10% of point.
- The recirculation flow is changed quickly (±) to affect power by less than 10% of point.
- The feedwater flow is changed quickly (±) to affect power by less than 10% of point.

Large Transient Tests

Large transient tests are performed at only a few power levels to demonstrate safe response to abnormal disturbances:

- The generator breaker is tripped which causes the turbine control valves to close rapidly and the bypass valves to open rapidly.
- The turbine stop valves are tripped closed which also causes the bypass valves to open rapidly.
- One and both recirculation pumps are tripped.
- By means of the flow control system, the recirculation flow is varied at maximum rate over its maximum range.
- The main steam isolation valves are closed.

WARRANTY DEMONSTRATION

Upon completion of all startup tests, the NSSS warranty performance test is performed in accordance with the NSSS contract at full power to demonstrate capability of the nuclear system to produce the steam flow rates at required pressure and quality.