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Realistic Simulation of Severe Accidents in BWRs—Computer Modeling Requirements

Sherrell R. Greene

Prepared for the U.S. Nuclear Regulatory Commission
Division of Engineering Technology
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FOREWORD

The Severe Accident Sequence Analysis (SASA) Program was established by the Nuclear Regulatory Commission in October 1980 for the purpose of studying potential nuclear power plant accidents beyond the design basis. Under the auspices of this program, boiling water reactor (BWR) studies have been conducted at Oak Ridge National Laboratory (ORNL) using Browns Ferry Unit One as the model BWR-4 MK I plant with the assistance and full cooperation of the plant owners and operators, the Tennessee Valley Authority. It is intended that some of the future studies will involve BWR-5 and BWR-6 plants with the MK II and MK III containment designs.

The primary analytical tool for analysis of the events of each severe accident that would occur after the core has been uncovered is the MARCH code, originally developed by Battelle-Columbus Laboratories.* The MARCH code incorporates the principal meltdown computer models used in the Reactor Safety Study† and various improvements and modifications added thereafter. A recent MARCH code assessment,‡ performed primarily from the standpoint of the application of MARCH to pressurized water reactor (PWR) accident analysis, points out that

> "The code's development, its structure, level of detail, etc., reflect the limited goals of early risk assessments. Thus, for example, relatively simple and fast-running models were needed so that many types and numbers of accident sequences could be investigated. Further, the uncertainties associated with using these simple models were not considered to be of major concern, in light of the large overall uncertainties present in risk assessment."

The original MARCH primary system thermal-hydraulic models are particularly crude; core flow is not modeled and the reactor vessel is modeled only as a two-node cylindrical volume with water at the bottom and steam at the top. The MARCH 2.0 version now becoming available comprises significantly improved modeling for PWR applications but most of the specific limitations of the original code with respect to BWR accident analysis remain. The significant BWR modeling deficiencies have been identified by the SASA Program at ORNL and have been documented

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*R. O. Wooten and H. I. Avci, *MARCH Code Description and User's Manual*, NUREG/CR-1711, Battelle Columbus Laboratories (1980).

†*Reactor Safety Study*, WASH-1400, NUREG-75/014, Washington, DC: U.S. Nuclear Regulatory Commission (1975).

‡J. B. Rivard et al., *Interim Technical Assessment of the MARCH Code*, NUREG/CR-2285, SAND 81-1672, Sandia National Laboratories (1981).

elsewhere.* Major items involve the representation of the core and reactor vessel internals and the modeling of the response of the secondary containment.

The SASA Program is not intended to involve code development. Nevertheless, since the MARCH code in its original form does not adequately represent the BWR, it has been necessary to perform modifications to the ORNL version of the code for each severe accident sequence studied. Assistance in this effort has been obtained through a subcontracted effort at Renessalaer Polytechnic Institute (RPI) for resolution of the modeling needs (such as the effect of core spray) that require a long-term development effort.

At ORNL the BWR modeling needs for severe accident analysis have been identified through the experience of attempting to apply the MARCH code in the SASA Program. Models have been developed to represent the BWR core and the effect of safety/relief valve actuation, and an arrangement has been effected with RPI to provide the long-term model development needed for adequate representation of the response of BWRs during severe accidents. However, as previously stated, the SASA Program should not be involved in code development and is so involved only out of sheer necessity.

The MELCOR code development program at Sandia is currently in the process of developing the successor to the MARCH code. It seems fitting that the ORNL SASA Program should terminate its code development efforts, making available all work completed to date and all assets available for future development of BWR-specific models to the MELCOR program in a cooperative effort. This report, completely funded by the SASA Program, is the first major milestone in this cooperative effort. It is intended that future BWR model development efforts will be primarily funded by the MELCOR project with significant support by the ORNL SASA Program for items of principal interest to SASA.

The primary purpose of this report is to identify the minimum modeling necessary to permit the analysis of the response of BWR plants under severe accident conditions. The BWR design and operation differ significantly from that of a PWR and, as shown in this report, special modeling is both appropriate and absolutely necessary.

S. A. Hodge
SASA Project Manager

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*S. R. Greene et al., *SBLOCA Outside Containment at Browns Ferry Unit One - Accident Sequence Analysis*, NUREG/CR-2672, ORNL/TM-8119/V1, Appendix B (1982).

REALISTIC SIMULATION OF SEVERE ACCIDENTS IN BWRs — COMPUTER MODELING REQUIREMENTS

Sherrell R. Greene

ABSTRACT

This report documents the results of an assessment performed at Oak Ridge National Laboratory to determine the reactor and containment hardware, systems, and phenomena which must be modeled in realistic boiling water reactor severe accident analysis computer codes. The scope of the assessment is limited to BWR-4, 5, and 6 reactors and Mark I, II, and III containment systems. The report presents a concise review of the subject reactor and containment designs, together with a description of the reactor and containment systems which have the capacity to impact the outcome of severe accidents. The results of recent BWR probabilistic risk assessments are briefly discussed, and a detailed visualization of a BWR core melt accident is presented. Recommendations are made regarding the type of phenomena which should be modeled and the level of modeling sophistication required for various stages of the core melt accident. Finally, the current availability of the necessary models is discussed along with the associated model development priorities.

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1. INTRODUCTION

1.1 Background

During the past several years, the NRC Office of Nuclear Regulatory Research (RES) has funded the development of several computer codes for analysis of the risk associated with operation of commercial light water power reactors. The MARCH, CORRAL, and CRAC codes were developed in conjunction with or as follow-ons to the Reactor Safety Study.[1.1] During the years since their development, these codes have been applied in a variety of LWR severe accident studies by both NRC contractors and public utilities. As a result of this experience and increased knowledge concerning severe accident phenomena, it has become clear that further improvements in severe accident modeling codes are essential.[1.2-1.3] The NRC, through its contractor, Sandia National Laboratory, is pursuing the development of the MELCOR integrated risk analysis code to address these problems.[1.4]

One of the intended applications of MELCOR is the addressing of recent severe accident source term controversies. The code is also intended to provide a structure which can be readily modified as new data become available. The results of this effort will be used for developing improved regulatory and licensing perspectives which could then evolve into new regulations.

The initial task in the MELCOR development plan is the identification of the minimum modeling capabilities which the code must have to facilitate its intended applications. The assessment of severe accident modeling needs for boiling water reactors (BWRs) was undertaken by the Severe Accident Sequence Analysis (SASA) Program staff at Oak Ridge National Laboratory (ORNL). The purpose of this report is to document the results of the ORNL assessment.

1.2 Approach

The major initial goal of the MELCOR program is to develop a modular code structure with sufficient flexibility to accommodate future improvements in LWR severe accident modeling methodologies. The purpose of this assessment is to identify the BWR plant hardware, systems, and phenomena that must or should ultimately be modeled to facilitate realistic simulation of severe accidents in boiling water reactors. The assessment effort will be successful only if it provides the information necessary to ensure that MELCOR is sufficiently flexible to accommodate important and appropriate accident phenomena, future improvements to phenomenological models, and a wide variety of accident types.

The approach adopted for this assessment is outlined in Table 1.1. Simply stated, the assessment is based on a review of actual plant hardware and system designs, and their possible performance characteristics during normal, design basis, and beyond design basis conditions. Based on this review, the significant physical phenomena that might occur in the reactor vessel and containment during a severe accident are identified and their overall importance is assessed. This system and phenomena data are then used to synthesize a MELCOR systems/phenomena modeling capabilities envelope.

1.3 Limitations of the Assessment

Table 1.2 is a list of domestic BWR power plants which are now operating or under construction. There are only two BWR-1 and two BWR-2 units still in operation. Both of the BWR-1s shown in Table 1.2 have reactor and containment system designs which differ significantly from later plant designs. While the two BWR-2 units shown in Table 1.2 do have standardized containments, both of these reactors employ external recirculation loops rather than the internal jet pumps which are utilized in the BWR-3 and later designs. The BWR-3 design is similar to that of the BWR-4 except that some BWR-3 units utilize isolation condensers to cool the core after main steam isolation valve closure, rather than reactor core isolation cooling (RCIC) systems which are utilized in BWR-4 and later designs.

The author has relied heavily on the experience gained and lessons learned during the past 3 years under the SASA program at ORNL — rather than new evaluation efforts. The ORNL SASA work has been oriented primarily towards BWR-4 MK I plants, with some associated work with BWR-4 MK II and BWR-6 MK III units. After consideration of the brief period of time available for the assessment (approximately 5 months), the experience of the ORNL staff, and the relatively small number of BWR-1, BWR-2, and BWR-3 isolation condenser plants, it was decided that the scope of the assessment should be limited to BWR-4, BWR-5, and BWR-6 plants.

The reader should bear in mind that BWR-3 units with RCIC systems are very similar to BWR-4 designs. Although this assessment does not specifically address BWR-3/RCIC design modeling issues, the results of the assessment should be applicable to these designs as well as to BWR-4, BWR-5, and BWR-6 facilities.

Chapter 1 References

1.1. *Reactor Safety Study*, WASH-1400, NUREG-75/014, U.S. Nuclear Regulatory Commission (1975).

1.2. J. B. Rivard et al., *Interim Technical Assessment of the MARCH Code*, NUREG/CR-2285, SAND 81-1672, Sandia National Laboratories (1981).

1.3. S. R. Greene et al., *SBLOCA Outside Containment at Browns Ferry Unit One - Accident Sequence Analysis*, NUREG/CR-2672, ORNL/TM-8119/V1, Appendix B (1982).

1.4. *Development of Improved Physical Process Computer Codes for Risk Assessment (MELCOR)*, U.S. Nuclear Regulatory Commission FY 82 Program Brief.

Table 1.1. BWR modeling needs
assessment approach

Step No.	Description
1	Define reactor/containment types of interest
2	Describe reactor/containment structures and hardware
3	Identify "impacting systems" — systems which can modify reactor/containment mass and energy balances during normal and accident conditions. Determine operating modes and control parameters for each system
4	Identify physical phenomena associated with "impacting systems" operations under normal and accident conditions
5	Assess magnitude of mass/energy balance impacts under normal and accident conditions
6	Specify desirable systems/phenomena modeling envelope

Table 1.2. Domestic BWRs operating
or under construction

Plant	Reactor type	Containment type	Power (MWe)	Startup date
Big Rock Point	1	dry	63	12/62
LaCrosse	1	dry	50	11/69
Nine Mile Point 1	2	MK I	610	12/69
Oyster Creek 1	2	MK I	620	12/69
Dresden 2	3	MK I	794	8/70
Millstone 1	3	MK I	660	12/70
Monticello	3	MK I	536	7/71
Dresden 3	3	MK I	794	10/71
Quad Cities 1	3	MK I	789	8/72
Quad Cities 2	3	MK I	789	10/72
Pilgram	3	MK I	670	12/72
Vermont Yankee	4	MK I	514	11/72
Duane Arnold	4	MK I	545	5/74
Cooper	4	MK I	778	7/74
Peach Bottom 2	4	MK I	1065	7/74
Browns Ferry 1	4	MK I	1067	8/74
Peach Bottom 3	4	MK I	1065	12/74
Browns Ferry 2	4	MK I	1067	3/75
Fitzpatrick 1	4	MK I	821	7/75
Brunswick 2	4	MK I	790	11/75
Hatch 1	4	MK I	797	12/75
Brunswick 1	4	MK I	790	3/77
Browns Ferry 3	4	MK I	1067	3/77
Hatch 2	4	MK I	806	8/79
Enrico Fermi 2	4	MK I	1100	11/83
Hope Creek 1	4	MK I	1070	12/86
Susquehanna 1	4	MK II	1050	5/83
Shoreham	4	MK II	820	9/83
Susquehanna 2	4	MK II	1050	1984
Limerick 1	4	MK II	1055	4/85
Limerick 2	4	MK II	1055	10/87
LaSalle 1	5	MK II	1078	10/82
LaSalle 2	5	MK II	1078	10/83
WNP 2	5	MK II	1100	2/84
Zimmer 1	5	MK II	810	1984
Nine Mile Point 2	5	MK II	1080	10/86
Grand Gulf 1	6	MK III	1250	1983
Perry 1	6	MK III	205	5/84
Clinton 1	6	MK III	950	8/84
River Bend 1	6	MK III	940	12/85
Perry 2	6	MK III	1205	5/88
Skagit 1	6	MK III	1288	1981

2. PLANT DESCRIPTION — REACTOR

2.1 Introduction

The BWR direct steam cycle (Fig. 2.1) begins with the reactor vessel which is part of the reactor coolant pressure boundary and contains the reactor core. The reactor core provides the heat source for steam generation and consists primarily of the nuclear fuel and control rods for regulating the fission process. The steam generated in the reactor vessel is routed to the steam loads and then condensed into water. The water is then purified, heated, and pumped back to the reactor vessel. Water from the reactor vessel is circulated through external pumping loops and then returned to the reactor vessel to provide forced flow circulation through the reactor core. Reactor water is continuously purified to minimize impurities. Should the reactor become isolated from its main heat sink, an auxiliary system such as the RCIC systems described in Sect. 2.6.6, automatically maintains the vessel water level above the reactor core. A reactor heat balance for a 3293 MW BWR 4 is provided in Fig. 2.2. The BWR primary and auxiliary systems are briefly discussed in the following paragraphs. The reader should consult Refs. 2.1 and 2.2 if more detailed information is required.

The reactor vessel houses the reactor core and internals. The reactor vessel also serves as part of the reactor coolant pressure boundary: supports and aligns the fuel and control rods; provides a flow path for the circulation of coolant past the fuel; removes moisture from the steam exiting the reactor core; and provides an internal floodable volume to allow for reflooding the reactor core following a loss of coolant accident.

The fuel generates energy from the nuclear fission reaction to provide heat for steam generation. The control rods control reactor power level, control axial and radial power (neutron flux) distribution to optimize core performance, and provide adequate excess negative reactivity to shut down the reactor from any normal operating or accident condition at the most reactive time in core life.

The Control Rod Drive System makes gross changes in core reactivity by positioning the neutron absorbing control rods in response to Rod Control and Information System (RCIS) signals and rapidly inserts all control rods to shut down the reactor in response to Reactor Protection System (RPS) signals.

The Recirculation System provides forced circulation of water through the reactor core, thereby allowing a higher power level to be achieved than with natural circulation alone.

The Main Steam System directs steam from the reactor vessel to the turbine generator, bypass valves, reactor feed pump turbines, and other selected balance of plant loads; directs steam to certain safety related systems under abnormal conditions; and provides overpressure protection for the reactor coolant pressure boundary.

The Condensate and Feedwater System condenses turbine exhaust or bypass valve steam; removes impurities from the water delivered to the reactor vessel, heats the feedwater, pumps the feedwater into the reactor vessel; and provides a means for the Reactor Water Cleanup (RWCU) System, Reactor Core Isolation Cooling (RCIC) System, Residual Heat Removal (RHR) System, and the high pressure coolant injection (HPCI) system to discharge water to the reactor vessel.

The Reactor Core Isolation Cooling System supplies high pressure makeup water to the reactor vessel when the reactor is isolated from the main condenser and the Condensate and Feedwater System is not available.

The Reactor Water Cleanup System maintains reactor water quality by removing fission products, corrosion products, and other soluble and insoluble impurities; provides a path for removal of reactor coolant from the reactor vessel in case of excess coolant inventory; and maintains circulation in the reactor vessel bottom head to minimize thermal stratification.

Descriptions of the reactor vessel and internals are given in Sects. 2.2 and 2.3. The design and operation of BWR safety/relief and main steam isolation valves are discussed in Sects. 2.4 and 2.5. Systems that are capable of injecting water into the reactor vessel are described in Sect. 2.6 and reactor vessel water cooling systems are discussed in Sect. 2.7. The operation of the automatic depressurization system is discussed in Sect. 2.8. The topic of BWR neutronics and power control is briefly discussed in Sect. 2.9. Section 2.10 discusses the design and function of the reactor protection system, while Sect. 2.11 describes those functions of the primary containment and vessel isolation control system that are related to the reactor. Finally, Sect. 2.12 presents a brief summary description of an imaginary generic BWR primary system which incorporates all the systems and features discussed in this chapter. Though such a reactor does not actually exist, it is believed that such a composite system description is a useful tool for evaluation of the ultimate BWR severe accident modeling capabilities one would wish to have availab e. In addition to the composite primary system description, the concept of system control and isolation matrices (which are also very useful tools for assessing the scope of desirable severe accident modeling capabilities) are introduced in this chapter. Major portions of the descriptive material in this chapter are excerpted from the NRC BWR-4 and BWR-6 Systems Manual (Refs. 2.1 and 2.2) and various plant Final Safety Analysis Reports (FSARs) as noted in the text.

2.2 Reactor Vessel

The BWR pressure vessel (Fig. 2.3) consists of a cylindrical shell with an integral hemispherical bottom head. The top head is also hemispherical in shape, but is removable to facilitate refueling operations. The lines that penetrate the reactor vessel and the vessel internal structures are discussed in later paragraphs of this section.

The reactor vessel is mounted vertically within the drywell and consists of a cylindrical pressure vessel of welded construction. The vessel base material is a low manganese-molybdenum, low carbon steel alloy. The vessel is designed and fabricated in accordance with the

ASME codes. The hemispherical top and bottom heads are fabricated to the same standards as the vessel shell.

The vessel shell and bottom head sections are clad on the interior with a 0.3 cm (1/8 in.) austenitic stainless steel weld overlay. The cladding is used to minimize corrosion which could adversely affect water clarity during refueling operations.

The overall height of the reactor vessel is ~22.2 m (~73 ft) with an inside diameter of 4.6 to 6.4 m (183 to 251 in.) (plant dependent). Vertical measurements are referenced to vessel zero, which is defined as the lowest point on the bottom head internal to the vessel. The thickness of the reactor vessel wall varies from 15.9 cm (6-1/4 in.) on the sides, to a maximum of 20 cm (8 in.) in the bottom head region. The top head is 8.6 cm (3-3/8 in.) thick at the top circular center piece, with the remainder being 12.4 cm (4-7/8 in.) in thickness. Typical reactor vessel design pressures and temperatures are 8.6 MPa and 575 K (1250 psig and 575°F). The total weight of the reactor vessel is 6.8 x 10^6 kg (~1.5 x 10^6 lbm) (BWR-4).

Penetrations in the pressure vessel top and bottom head and in the cylinder walls provide the means for effluent and influent reactor water, main steam lines, movement of internal components for reactor monitoring and control, reactor instrumentation, and accommodations for necessary safety devices. The vessel nozzle penetrations are identified in Fig. 2.3. Table 2.1 is a listing of these penetrations.

Six feedwater nozzles penetrate the reactor vessel. These penetrations, located at 60° intervals around the vessel, evenly distribute the feedwater via six feedwater spargers so that it mixes with the reflux liquid removed by the steam separation equipment higher in the reactor vessel.

Two recirculation suction lines penetrate the reactor vessel at the 4.3-m (170-in.) level. These 56-cm (22-in.) diam lines provide water from the vessel annulus region to the suction of the recirculation pumps.

Ten 30 cm (12 in.) recirculation inlet penetrations route water from the recirculation pump discharge to the inlet driving nozzles of the jet pumps. The jet pumps provide forced flow of the coolant and moderator through the reactor core.

The BWR-4 upper reactor vessel head incorporates penetrations for the head spray and vent/level instrumentation nozzles, and for a third nozzle that is normally unused.

The BWR-6 upper vessel head has two penetrations. The first connection is 76 cm (30 in.) radially off center and serves as a spare penetration with no planned use. The other penetration, directly at the center of the head, is for the flanged head vent and head spray lines. This dual purpose nozzle is a pipe within a pipe. The outer pipe, which is the head vent line, serves to vent noncondensable gases from the upper vessel region during startup, normal operation, and vessel floodup. During operation at temperatures less than the saturation temperature, noncondensable gases are vented to the drywell equipment drain sump through two motor operated valves. At temperatures above the saturation temperature, the vent is directed to the main steam line. The vent line is also used as a sensing point for a portion of the reactor vessel level instrumentation. The center portion of the upper penetration is the head spray line. The line sprays water from the Residual Heat Removal (RHR) System into the upper head region to collapse steam formed during vessel cool-down or vessel flooding.

The reactor vessel bottom head, which is welded to the vessel shell, contains numerous penetrations (Fig. 2.4) for control rods, local power range monitor (LPRM) detector strings, intermediate range monitor (IRM) detectors, and source range monitor (SRM) detectors.

Current BWR designs utilize between 137 and 193 control rod drive housing penetrations. In some BWRs, the stainless steel housings (Figs. 2.5 and 2.6) are welded to Inconel or stainless steel stub tubes which project up from the bottom of the vessel head. In other plants, the housings themselves are welded directly to the vessel head. The outside diameter of the housings is ~15.2 cm (~6 in.).

Current BWR designs also incorporate approximately 55 incore instrumentation tube penetrations in the bottom head. The penetrations generally consist of 2-in. diam tubing sections which are welded to the inside surface of the lower head. The wall thickness of the tubing is ~0.1 cm (~0.25 in.).

One penetration at the center of the bottom head provides a 5 cm (2 in.) low point drain. The drain line normally directs flow to the Reactor Water Cleanup (RWCU) System to aid in the removal of suspended solids, to provide a temperature measurement of water in the bottom head area, and to prevent cold water stagnation in the bottom head area. The line can also be used for a flushing connection during construction or a low point drain should the vessel ever have to be completely drained. Another penetration of the bottom head is a combination line used for the Standby Liquid Control (SLC) System and for pressure measurement below the core plate. The permanently installed line is connected to a distribution sparger for sodium pentaborate injection, and provides a path for the injection of the liquid control solution into the coolant stream.

2.3 Reactor Internals

2.3.1 Introduction

Unlike the PWR vessel, the BWR vessel internal volume is crowded with a wide variety of structures (Fig. 2.5). These reactor vessel internals are installed within the vessel to properly distribute the flow of coolant within the vessel, to locate and support the fuel assemblies, to provide an inner volume containing the core that can be flooded following a pipe break in the nuclear process system, and to increase the quality of the steam leaving the reactor vessel. The principal reactor vessel internals are the control rod guide tubes, the fuel/channel assemblies, control rods, shroud head, stand pipes and separators, and the steam dryers. Other major components include the core shroud, jet pumps, core plate, and the fuel support pieces. Except for the Zircaloy used in the fuel cladding and channels, the reactor internals are stainless steel or other corrosion-resistant alloys. The design and arrangement of these structures will be described in the remaining sections of Chap. 2.

2.3.2 Control rod drive housings, control rod guide tubes, and fuel support pieces

The control rod drive (CRD) housings, shown in Fig. 2.6, are extensions of the reactor vessel for mounting of the control rod drive mechanisms. CRD housings are ~4.3 m (~14 ft) long and provide vertical and lateral support for the control rod drives. They also transmit the weight of the fuel, the fuel support pieces, and control rod guide tubes to the reactor vessel bottom head.

The CRD housings, which are inserted from the bottom of the vessel, have flanges at the bottom for bolting of the CRD mechanisms and for the permanent attachment of the CRD hydraulic system insert and withdraw lines. Each housing is inserted through the bottom of the vessel, and welded to either an Inconel stub tube (which is welded to the inside of the reactor vessel bottom head) or directly to the vessel head. After installation and alignment, the top surfaces of the CRD housings are all at the same elevation. The CRD housings are constructed of austenitic 300 series stainless steel.

Each control rod guide tube, shown in Fig. 2.7, is ~0.28 m (~11 in.) in diameter and slightly over 4 m (13 ft) in length. The top portion has four 7.6 cm (3 in.) diam holes which direct inlet core flow from the below core plate area to the fuel assemblies through the flow orifices in the fuel support pieces. The bottom end of the guide tube is machined to mate with the CRD housing and is locked in place on top of the housing via the CRD thermal sleeve.

The guide tube performs the following functions: it guides the lower end of the control rod during rod movement; it forms a cylinder around the velocity limiter portion of the control rod so it can retard the free fall velocity under rod drop accident conditions (see Sect. 2.3.5); it supports and locates the orificed fuel support piece which, in turn, vertically supports the fuel; it provides a portion of the controlled reactor coolant leakage between the upper and lower plenum; and it provides the coolant flow passage into the orificed fuel support piece. The control rod guide tubes are constructed of austenitic 300 series stainless steel. The Browns Ferry reactors (BWR-4) employ 185 control rod guide tubes with a combined total weight of 20,974 kg (46,250 lbm).[2.3]

The fuel support pieces are of two basic types — peripheral and four-lobed. The peripheral fuel support pieces, which are welded to and supported by the core plate, are located at the outer edge of the active core and are not adjacent to control rods. Each peripheral fuel support piece will support one fuel assembly and contains a replaceable orifice assembly designed to assure proper coolant flow to the fuel assembly. The four-lobed fuel support pieces, shown in Fig. 2.8, will each support four fuel assemblies and are provided with orifice plates to assure proper coolant flow distribution to each fuel assembly. The four-lobed fuel support pieces rest in the top of the control rod guide tubes, which transfer the weight of the assemblies to the bottom head of the reactor vessel. The fuel support pieces are positioned laterally by the core plate. The control rods pass through slots in the center of the four-lobed fuel support pieces. The fuel support pieces are constructed of austenitic 300 series stainless steel. The Browns Ferry BWR-4 reactors employ 24 peripheral and 185 four-lobed support pieces with a combined weight of 5,125 kg (11,300 lbm).[2.3]

2.3.3 Core shroud, core plate, and top guide

The core shroud (Fig. 2.9) is a cylindrical stainless steel structure 5 cm (2 in.) thick, and weighing ~53,000 kg (~117,000 lbm) that surrounds the core and provides a barrier to separate the upward core flow from the downward flow in the annulus. The volume enclosed by the core shroud is characterized by three regions each with a different shroud diameter. The upper shroud has the largest diameter and surrounds the core discharge plenum which is bounded by the domed shroud head on top and the top guide below. The central portion of the shroud surrounds the active fuel and forms the longest section of the shroud. This section has the intermediate diameter and is bounded at the bottom by the core plate. The lower shroud, surrounding part of the lower plenum, has the smallest diameter.

A flange at the top of the shroud mates with a flange on the shroud head/steam separator assembly to form the core discharge plenum. The bottom of the shroud is welded to a rim on the baffle plate (Fig. 2.9). The outer diameter of the baffle plate is welded to the reactor vessel, and the inner diameter is supported by columns extending to the bottom head of the vessel. All of the vertical weight of the shroud, steam separator assembly, core plate and top guide, peripheral fuel assemblies, and the jet pump components carried on the shroud, is supported from the baffle plate supports. The baffle separates the annulus between the core shroud and the reactor vessel from the core inlet plenum. The diffusers of the jet pumps extend through holes in the baffle plate and are welded to the baffle plate.

The core plate, shown in Figs. 2.4, 2.10, and 2.11, consists of a circular horizontal stainless steel plate with vertical stiffener plate members below the horizontal plate. Tie rods serve to cross brace the stiffener members. The core plate has holes to accommodate the control rod guide tubes, alignment pins to ensure proper guide tube and fuel support piece orientation, holes for peripheral fuel support pieces, in-core guide tube holes for neutron instrumentation, and neutron source location holes.

The core plate acts as a partition to force the majority of the coolant and moderator into the holes in the upper portion of the control rod guide tubes, through the fuel support pieces and finally into the fuel assemblies. The core plate also provides vertical and lateral support for the peripheral fuel bundles via their fuel support pieces. It provides lateral support for all of the control rod guide tubes and hence lateral support for all of the fuel support pieces and fuel bundles. Vertical support for all of the fuel except the peripheral fuel bundles is provided by the fuel support pieces, the control rod guide tubes, and the bottom head of the reactor vessel via the control rod drive housing — not by the core plate. The core plate assembly is bolted to a support ledge between the central and lower regions of the core shroud (Fig. 2.4). The nominal weight of a typical BWR-4 core plate is ~9,297 kg (20,500 lbm).

The top guide, shown in Fig. 2.10, is set on a rim near the top end of the shroud and is bolted in place. The top guide is formed by a series of stainless steel plates joined at right angles to form a matrix of square openings. Each central opening accommodates four fuel assemblies and one control rod (this is defined as a fuel cell). Along the periphery are smaller openings which accommodate the peripheral fuel assemblies. Cutouts are provided on the bottom edge of the top guide at the junction of the cross plates to support the top end of the neutron instrument assemblies and neutron source holders. The top guide provides lateral support for the upper end of all fuel assemblies, neutron monitoring instrument assemblies, and the installed neutron sources. A nominal weight of a typical BWR-4 top guide is 6.893 kg (15,200 lbm).[2.3]

2.3.4 BWR fuel assemblies

A BWR fuel assembly (Fig. 2.12) consists of a fuel bundle and the zircaloy fuel channel that surrounds it. The fuel assemblies are arranged in the reactor core to approximate a right circular cylinder. Each fuel assembly is supported vertically and laterally by a fuel support piece at the bottom and laterally by the top guide at the top. The number of fuel assemblies employed in current BWR designs varies between 368 and 800.

More than 30 production fuel types have been designed, manufactured, and operated in BWRs. At this time both 7 x 7 and 8 x 8 (the numbers used to designate the fuel correspond to the number of individual rods on one side of the fuel assembly) are being used in production reactors. In the near future, the 8 x 8 will be the principle BWR fuel. There are three basic types of 8 x 8 fuel in use at this time. They are designated 8 x 8, 8 x 8R, and P8 x 8R. Table 2.2 lists the major fuel assembly design parameters for each of these three fuel bundle types.

A fuel bundle (fuel assembly without fuel channel) contains fuel rods and (in recent designs) water rods which are spaced and supported in a square array at the ends of the fuel bundle by the lower and upper tie plates. There are three types of rods in a fuel bundle: standard rods, tie rods, and hollow water rods.

The fuel rods are hollow cladding tubes fabricated from Zircaloy-2 alloy. High density (95% theoretical density) solid, right circular uranium oxide pellets are stacked inside each cladding tube. Following the loading of the fuel pellets, the fuel rod is heated and evacuated by means of a vacuum pump. It is then backfilled with helium gas at either one (8 x 8 and 8 x 8R fuel types) or three (P8 x 8R fuel types) atmospheres of pressure. The standard fuel rod (in 8 x 8 assemblies) is 4.06 m (160 in.) in length with an active fuel length of 3.81 m (150 in.). [The active length of 7 x 7 fuel rods is only 3.66 m (144 in.)]. The top 15 cm (6 in.) and bottom 15 cm of uranium oxide fuel is natural uranium. The remaining fuel pellets are enriched in U^{235} with some pellets having a urania-gadolina mixture. A free volume, provided in the top 25 cm (10 in.) of the fuel rod, contains a plenum spring to compress the fuel pellets axially and a small tube which contains a hydrogen getter. The getter is provided in the plenum space as assurance against chemical attack from any inadvertent admission of moisture from hydrogenous impurities in the fuel rod during manufacture. The end plugs of the standard rods have rounded shanks which fit into the upper and lower plates.

The number of fuel rods depends on the type of fuel (Table 2.2). The 7 x 7 fuel design utilizes 49 fuel rods. An 8 x 8 fuel bundle designated as 8 x 8 contains 63 fuel rods and 1 water rod. Fuel designated as 8 x 8R (retrofit) or P8 x 8R (prepressurized retrofit) contains 62 fuel rods and 2 water rods (Fig. 2.13). A fuel bundle is ~4.6 m (~15 ft) in length and weighs between 280 and 308 kg (617 and 679 lbs) depending upon bundle type. The fuel channel box is attached to a fuel bundle to form a fuel assembly. The entire assembly weighs between 324 and 340 kg (715 and 749 lbs) when assembled and measures 14 cm (5.5 in.) on a side.

The tie rods hold the fuel bundle together and support its weight during fuel handling operations when the fuel assembly is hanging from fuel handling equipment. Tie rods differ from standard fuel rods in that the lower end plugs thread into the lower tie plate casting and the upper end plugs are threaded and extend through the upper tie plate casting.

As previously noted, two rods in each 8 x 8R fuel bundle and 1 rod in each 8 x 8 fuel bundle are hollow water tubes, containing no fuel pellets. These water rods introduce moderator to the center of the fuel bundle to achieve a better thermal neutron flux and power distribution across the bundle. The outer diameter of the water rods is slightly larger than that of the standard rod. Several holes are drilled through the tube wall at the top and bottom of the rod to facilitate coolant flow.

The Zircaloy-4 channel box provides guidance and bearing surfaces for the control rod, permits control of flow distribution in combination with the orifice located in the fuel support plate, and protects the fuel assembly during fuel handling operations. Approximately 90% of the coolant flows within the fuel channels to remove heat from the fuel rods while 10% provides cooling flow in the interstitial region between fuel assemblies. Each channel box is ~4.1 m (~162 in.) long, with an average wall thickness of 2, 2.5, or 3 mm (0.080, 0.100, or 0.120 in.). Significant fuel channel design parameters are given in Table 2.2.

The lower tie plate, manufactured from a stainless steel casting, positions the fuel rods laterally and transfers vertical loads (weight of fuel assembly) to the fuel support piece. The nose piece of the lower tie plate fits precisely into the fuel support piece and directs coolant flow up through the fuel assembly.

The upper tie plate (manufactured from a stainless steel casting) provides alignment and support for the fuel rods at the top of the fuel bundle. The holes bored vertically through the upper tie plate position the fuel rods laterally at the upper end of the fuel bundle. A lifting handle is an integral part of the upper tie plate and is used for moving and handling the fuel bundle during initial core loading and subsequent refueling operations.

2.3.5 Control rods

As previously described, the fuel assemblies are arranged within the reactor core in fuel cells. Each of these cells consists of a control blade and four fuel assemblies (Fig. 2.13). With the exception of a few fuel assemblies around the outer edge of the core, each assembly is directly adjacent to a control blade.

The control rods (Fig. 2.14) perform the dual function of power shaping and reactivity control. Power distribution in the core is controlled during operation of the reactor by manipulation of selected patterns of control rods. The control rods are positioned in a manner which counterbalances steam void effects at the top of the core and results in significant power flattening.

The control rod consists of a sheathed cruciform array of stainless steel tubes filled with boron-carbide powder. The control rods are 24.9 cm (~9.8 in.) in total span and are located uniformly through the core on a 30.5 cm (12-in.) pitch.

Absorber tubes are made of stainless steel having an outside diameter of 0.48 or 0.56 cm (0.188 or 0.220 in.) and a wall thickness of 0.635 or 0.86 mm (0.025 or 0.027 in.). The tubes are sealed by a plug welded into each end. In order to prevent the formation of excessive void regions (which can be caused by settling of the B_4C powder), stainless steel balls spaced at 40.6 cm (16 in.) intervals are placed in the tubes. Should the boron carbide tend to compact further, the steel balls will distr[illegible] resulting voids over the entire length of the absorber tub[illegible] he contro rods are cooled by the fuel assembly bypass flow. The blade sheath is perforated to allow the coolant to freely circulate about the absorber [illegible]s.

2.3.6 Shroud head and steam separa[illegible]s

The shroud head and steam separator assembly is bolted to the top of the upper core shroud to form the top of the core discharge plenum. This plenum provides a mixing chamber for the steam-water mixture before it enters the steam separators. Refer to Fig. 2.15 for shroud head and steam separator assembly arrangement. The individual stainless steel axial flow steam separators (shown in Fig. 2.16) are attached to the top of the 16.3 cm (6.6 in.) OD standpipes which are welded into the shroud head.

The steam separators have no moving parts. In each separator, the steam-water mixture (10 to 13% quality), rising through the standpipe, passes turning vanes which impart a spin to establish a vortex which separates the water from the steam. The denser liquid is thrown radially outward by centrifugal force, forming a continuous wall of water on the inside wall of the inner pipe. Steam with a quality of at least 90% exits from the top of the separator and rises to the dryers. The separated water exits from under the separator cap and flows out between the standpipes, draining into the recirculation flow downcomer annulus. The combined weight of a typical shroud head-steam separator unit is ~63,490 kg (~140,000 lbm).[2.3]

2.3.7 Steam dryer

The steam dryer assembly, shown in Fig. 2.17, dries the wet steam from the steam separators, increasing its quality to greater than 99.9%. It also provides a seal between the wet steam area (steam exiting the separators) and the dry steam flowing to the main steam lines. The seal is formed by the steam dryer assembly seal skirt which extends below the normal reactor vessel water level. Abnormally low reactor water levels will result in loss of the water seal, opening a pathway for steam to bypass the dryer assembly.

The dryers are fabricated in a one piece assembly with no moving parts. The upper section of the assembly consists of steam dryers with portions of the sides cut away to permit steam flow to the main steam lines.

The dryers force the wet steam to be directed horizontally through the dryer panels. The steam is forced to make a series of rapid changes in direction while traversing the panels. During these direction changes, the heavier drops of entrained moisture are forced to the outer walls where moisture collection hooks catch and drain the liquid to collection troughs, then through tubes into the reactor vessel. An enlarged view of a steam dryer section is shown in Fig. 2.18.

The steam dryer assembly in a BWR-4 weighs ~40,800 kg (~90,000 lbm).[2.3] Upward movement of the steam dryer assembly is restricted by steam dryer hold down brackets attached to the reactor vessel top head.

2.3.8 Jet pumps

The jet pumps (Figs. 2.4, 2.9, and 2.19) provide forced flow of coolant through the reactor vessel to yield a higher reactor power output than is possible with natural circulation. The 20 jet pumps are located in two semicircular groups in the annulus region, with two jet pumps and a common inlet header combined to form a jet pump assembly.

Each stainless steel jet pump consists of a driving nozzle, suction inlet, throat or mixing section, and diffuser (Fig. 2.19). The driving nozzle, suction inlet, and throat are joined together as a removable unit while the diffuser is permanently installed. High pressure water from the recirculation pump discharge is directed downward into each pair of jet pumps through a riser pipe welded to the recirculation inlet nozzle thermal sleeve. A riser brace is welded to cantilever beams extending from pads on the reactor vessel wall.

The jet pump diffuser is a gradual conical section changing to a straight cylindrical section at the lower end. The diffuser is supported vertically by the baffle plate, a flat ring which is welded to the reactor vessel wall and to which is welded the shroud support cylinder. The throat section is supported laterally by a bracket attached to the riser. The jet pump diffuser section is welded to the baffle plate and provides a positive seal. This permits reflooding the core to the top of the jet pump inlet following a pipe break in the recirculation system. The combined weight of the jet pump assemblies is ~10,430 kg (~23,000 lbm).[2.4]

2.4 Safety/Relief Valves

The objective of the reactor vessel safety/relief valves (SRVs) is to prevent overpressurization of the nuclear system. The valves are located between the reactor vessel and the inboard main steam isolation valves on a horizontal run of the main steam lines within the drywell. This mounting location simplifies vessel head removal and makes the SRVs readily accessible during reactor shutdowns. The discharge from each safety/ relief valve is piped to the pressure suppression pool (see Sect. 3.2) with the line terminating below the pool water level to permit the steam to condense in the pool when the valve operates. Vacuum breakers are installed on the SRV piping inside the drywell to relieve pressure due to condensation in the tailpipe following actuation of a valve. The number of SRVs varies from plant to plant (i.e., 11 at Limerick; 24 at Nine Mile Point 2). Typical rated relief valve flows vary correspondingly.

Some operating BWRs are equipped with three stage Target Rock valves which have exhibited a higher probability to stick open in the past than other types of valves. Some operating BWRs are equipped with Dresser Electromatic relief valves. BWR-5 and BWR-6 plants are equipped with Crosby and Dikkers dual function safety relief valves.[2.5] Many utilities have replaced three-stage Target Rock valves with two-stage Target Rock valves in their BWR-4 plants.

All SRVs are capable of being operated either by pneumatic actuator (relief mode) or in direct response to steam pressure acting on a pilot valve or spring seated piston (safety mode). BWR safety/relief valves provide three main protective functions: (1) overpressure relief, (2) overpressure safety, and (3) depressurization. The primary purpose of this section is to discuss the relief and safety functions. The depressurization function of the SRVs is discussed more fully in Sect. 2.8.

All SRVs are fitted with pneumatic actuators which accommodate remote manual (BWR-4, 5, and 6) or remote automatic (BWR-5 and 6 overpressure relief) operation. All SRVs in the automatic depressurization system (ADS) (see Sect. 2.8) are fitted with high capacity pneumatic accumulators to ensure that these valves can be opened and held open following failure of the actuator air supply system. In some plants the remaining non-ADS valves are also fitted with smaller accumulators to ensure some degree of operability following failure of the actuator pneumatic air supply system. It is important to note that remote operation of these valves is possible only as long as the pneumatic actuator air supply system pressure exceeds the containment pressure by some minimum amount.

The lowest pressure setpoint is associated with the overpressure relief mode of operation. In a typical BWR-4 this mode is accommodated by utilizing multiple SRVs with different self actuating set points. In most BWR-5 and BWR-6 plants the overpressure relief function is accommodated by an automatic control system which senses reactor pressure and opens selected valves via their pneumatic actuators whenever the designated relief valve set pressure is attained.

The higher pressure set point associated with SRVs is the safety set point. All valves operate in a self actuated mode for safety functions. SRVs are typically arranged into multiple groups, each with different relief and safety pressure set points. Once open in the self actuated or remote automatic relief mode of operation, an SRV will reclose only after the reactor pressure has decreased 50 to 100 psi (plant dependent) below its opening pressure set point.

2.5 Main Steam Flow Restrictors and Isolation Valves

Immediately downstream of the last SRV and upstream of the inboard main steam isolation valve (MSIV) in the horizontal run of main steam discharge piping, a venturi flow nozzle is welded into each main steam line. The functions of these flow restrictors are to limit the steam flow in a severed main steam line to ~200% of rated flow for that line, thus limiting the rate of loss of coolant to protect the fuel barrier; to limit the differential pressure caused by high steam flow rates across the steam dryer and other reactor internal structures; and to provide steam line flow signals.

Each main steam line contains two MSIVs in series. These valves are welded in the horizontal pipe run with the inboard valve located inside but as close as possible to the drywell wall and the outboard valve just outside the primary containment boundary. Each MSIV is equipped with two independent position switches which provide a signal to the Reactor Protection System (RPS) scram trip circuit when the valve closes. A reactor scram will result from isolating more than one main steam line with the reactor plant at full power. The MSIVs are 'Y' pattern, pneumatic opening, spring and/or pneumatic closing, globe values designed to fail closed on loss of pneumatic pressure to the valve operator.

The MSIVs function in conjunction with the steam line flow restrictors to limit the release of radioactive materials to the environment or to limit reactor vessel inventory loss in the event of an accident. The MSIVs are automatically closed by any of the following conditions:

1. Main steam line high radiation
2. Main steam line turbine area high temperature (BWR-6)
3. Steam tunnel high temperature
4. Steam tunnel high differential temperature (BWR-6)
5. Main steam line high flow
6. Reactor low water level
7. Low main steam line pressure in the run mode (BWR-6)
8. Low condenser vacuum (BWR-6)
9. Main steam line low pressure with reactor mode switch in the run position (BWR-4)
10. Manual.

It is clear that the MSIVs would be closed under most severe accident situations. The reader should be cautioned, however, that several BWRs have encountered persistent and significant problems with steam leakage past closed MSIVs. Significant efforts are underway by both individual utilities and the BWR Owners Group to remedy these problems.[2.6] Many of the newer plants incorporate Main Steam Isolation Valve Leakage Control Systems (MSIV-LCS) which pull a vacuum on the section of the MSIVs between the inboard and outboard valves and beyond the outboard valves, and exhaust to the standby gas treatment system. MSIV leakage during core melt accidents, coupled with failure of the MSIV-LCS, could result in leakage of fission products into the turbine building atmosphere.

2.6 Vessel Water Injection Systems

2.6.1 Introduction

The purpose of this section is to briefly describe those systems that might be employed during a severe accident to inject water into the reactor vessel. Both traditional engineered safety feature (ESF) systems and non-ESF systems are discussed.

2.6.2 Condensate and Feedwater System

The Condensate and Feedwater System, shown in Fig. 2.1, is an integral part of the BWR regenerative steam cycle. The steam exhausted from the low pressure turbines is condensed in the main condenser and collected in the condenser hotwell, along with various equipment drains. A condensate transfer system is available to provide makeup to the condenser hotwells from the condensate storage tank in the event low hotwell condensate levels occur.

The condensate that is collected in the hotwell is removed by condensate pumps. These pumps provide the driving force for the condensate which flows through the steam jet air ejector (SJAE) condensers, steam packing exhauster condenser, and offgas condenser, performing a heat removal function. At this point the condensate is directed to the condensate demineralizers which remove impurities through the process of ion exchange. After the demineralizers, booster pumps are used to maintain the driving force of the condensate flow through strings of low pressure feedwater heaters. The feedwater pumps then take the condensate flow and increase the pressure to a value above reactor pressure.

During normal operation, the amount of feedwater flowing to the reactor vessel and, in turn, vessel level, is controlled by varying the speed of the turbine or motor driven reactor feed pumps. The discharge of the feedwater pumps is directed to the high pressure feedwater heater strings for the final stage of feedwater heating. Two feedwater lines penetrate the primary containment and then further divide into a total of six penetrations which enter the reactor vessel, each supplying feedwater to a feedwater sparger. The feedwater spargers distribute the flow of feedwater within the vessel annulus area. Each of the three condensate pumps normally take suction from the hotwell. These pumps are typically 33 to 50% capacity, vertical, two stage centrifugal, motor driven devices. Typical rated pump capacities range between 0.63 and 0.88 m^3s (10,000 and 14,000 gpm) at discharge pressures of 0.96 MPa (125 psig).

The condensate booster pumps normally take suction on the demineralizer outlet header and provide the required net positive suction head (NPSH) to the reactor feed pumps. The booster pumps are typically 33 to 50% capacity, horizontal, centrifugal, motor driven pumps, with rated capacities of 0.63 and 0.88 m^3/s (10,000 to 14,000 gpm) (plant dependent) at discharge gauge pressures of ~2.4 MPa (~350 psi).

The reactor feedwater pumps are single stage, turbine or motor driven centrifugal pumps. The pumps are typically 33 to 50% capacity units, with design flows of 0.69 and 1.32 m^3/s (11,000 to 21,000 gpm) with discharge pressures of 11.03 MPa (1,600 psia).

The condensate transfer pumps, which supply makeup water to the condenser hotwell, are low capacity centrifugal pumps rated at 0.006 m^3/s (100 gpm) against a head of 1 m (200 ft) (Browns Ferry).[2.3]

During most accident situations, the MSIVs would be closed and there would be no steam supply available to drive the feedwater pump turbines. Turbine driven reactor feedwater pumps would not, therefore, be available to inject water into the vessel under such conditions. However, the condensate booster pumps are capable of injecting water into the vessel through the stopped feed pumps whenever reactor vessel pressure drops below the condensate booster pumps shutoff head.[2.7] In many accident scenarios the condensate booster pumps would continue to run unless they are stopped by the operator. Some plants also have emergency operator procedures for realigning the condensate transfer pumps to inject directly into the reactor vessel via the core spray header, head spray nozzle or recirculation lines.[2.8]

2.6.3 Control Rod Drive Hydraulic System (CRDHS)

BWR control rod drive mechanisms are double acting, mechanically-latched hydraulic cylinders that use water from the condensate storage tank as operating fluid. Control rod movement is accomplished by admitting water under pressure from the Control Rod Drive Hydraulic System (CRDHS) to the region above or below the piston while draining water from the opposite side of the piston. A detailed description of the operation of the CRDHS is given in Ref. 2.8. The present discussion is limited to a description of the two major functions of the CRD hydraulic system.

The CRD hydraulic system (Fig. 2.20) consists of two electric motor driven centrifugal pumps (one operating, one spare), filters, valves and instrumentation necessary to convey water from the condensate storage tank to the control rod drive mechanisms. Although not a safety grade system, in many plants the CRDHS can be operated from emergency power supplies. The two major purposes of the system are to:

1. Maintain cooling water flow to each CRD mechanism during normal operating conditions, and
2. Provide the operating fluid and pressure necessary for movement of the control rods (both for individual rod movements and for system scram).

The CRD hydraulic piston seals are constructed of Graphitar, which is inert and has a low friction coefficient when water lubricated.

Because Graphitar loses strength at higher temperatures, the CRDHS supplies cooling water to hold the seal temperatures below 394 K (250°F) during normal operation. This cooling water flow of 1.9 x 10^{-5} m^3/s (~0.3 gpm) per mechanism [3.8 x 10^{-3} m^3/s (60 gpm) total for a typical BWR-4] is supplied at a pressure 138 KPa (~20 psi) above the reactor operating pressure.

When the Reactor Protection System (Section 2.10) issues a scram signal, the scram inlet and outlet valves open and the scram accumulators discharge to insert the control rods (see Fig. 2.20). Following control rod insertion, the total vessel inleakage past the CRD seals increases to ~110 to 200 gpm (plant specific and reactor pressure dependent). This increased injection flow will be maintained as long as the CRD hydraulic pumps are running and a scram signal is present. It is important to note that initiation of this increased injection flow requires no operator action of any type. In most plants the operator can take any one of several actions to further increase the CRD hydraulic system injection flow. Many accident scenarios would not involve CRDHS disfunction. The CRDHS is, therefore, an important injection mechanism which can influence many accident sequences.

2.6.4 High Pressure Coolant Injection (HPCI) System

BWR-4 plants are equipped with a High Presssure Coolant Injection (HPCI) System which is designed to supply makeup coolant to the reactor vessel (via a feedwater line) over a broad range of pressure from normal operating pressure to some predetermined depressurized condition (Fig. 2.21). The HPCI System consists of a steam turbine driven pump, piping, valves and controls necessary to provide a complete and independent emergency core cooling system. The principal HPCI equipment is installed in the reactor building. The HPCI System does not rely on auxiliary ac power, plant service air or external cooling water systems. The normal HPCI flow is ~0.31 m /s (~5000 gpm).

The normal source for HPCI injection water is the condensate storage tank, however, the pressure suppression pool is an alternative source of water. The HPCI pump suction will automatically and irreversibly shift from the condensate storage tank to the suppression pool if either a low condensate storage tank level or a high suppression pool level signal is received.

The HPCI turbine is driven by steam extracted from one of the main steam lines upstream of the main steam line isolation valves. In some plants the HPCI System can also be driven by steam from the auxiliary boiler system after installation of piping spool pieces. The turbine exhaust steam is discharged to the pressure suppression pool. Both the turbine oil and the gland seal condenser are cooled by the water being pumped.* Typical HPCI turbine steam demands range between 5.0 and 23.2

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*This is an important point, since the automatic shift of HPCI pump suction from the condensate storage tank to the pressure suppression pool has no logic to consider the suppression pool temperature.

kg/s (40,000 and 184,000 lbm/h). Typical system initiation, trip, and isolation control parameters are given in Table 2.3.

2.6.5 High Pressure Core Spray (HPCS) System

BWR-5 and BWR-6 plants are equipped with a High Pressure Core Spray (HPCS) System rather than a HPCI System. The purpose of the HPCS System is to maintain reactor vessel inventory after small breaks which do not depressurize the vessel, to provide spray cooling for line breaks which result in the reactor core becoming uncovered, and to backup the Reactor Core Isolation Cooling (RCIC) System during situations in which the reactor vessel is isolated.

The HPCS System, shown in Fig. 2.22, is a single loop system and consists of a suction shutoff valve, one motor driven pump, a discharge check valve, a motor operated injection valve, a minimum flow valve, a full flow test valve to the suppression pool, two high pressure flow test valves to the condensate storage tank, a HPCS spray sparger (inside the vessel above the core shroud), and associated piping and instrumentation. The HPCS System takes suction from the condensate storage tank and pumps the condensate into a sparger mounted in the core shroud. Spray nozzles mounted on the spargers are directed at the fuel bundles. The suppression pool is an alternate source of water for the HPCS System. The HPCS logic switches the HPCS pump suction from the condensate storage tank to the suppression pool upon either high level in the suppression pool or low level in the condensate storage tank.

As noted above, the HPCS System can take suction from the condensate storage tank (CST) or the pressure suppression pool. Normal suction is from the CST on a line common with the RCIC System suction line. The HPCI/HPCS/RCIC suction line from the CST is located lower than all other system suction lines to ensure a reserved volume of water in the CST exclusively for the HPCI, HPCS, and RCIC Systems.

The HPCS pump is a vertical, centrifugal, motor driven pump capable of delivering at least 0.098 m^3/s at 7.9 MPa (1550 gpm at 1147 psi) reactor pressure, 0.385 m^3/s at 1.38 MPa (6110 gpm at 200 psi) reactor pressure, and a maximum of 0.49 m^3/s (7800 gpm) at runout flow conditions.

The HPCS can be powered from both normal and emergency ac power systems. Major HPCS System components are located in the auxiliary building. Table 2.4 is a list of HPCS initiation, trip, and isolation control parameters.

2.6.6 Reactor Core Isolation Cooling (RCIC) System

The purpose of the reactor core isolation cooling (RCIC) System (Fig. 2.23) is to supply high pressure makeup water to the reactor vessel when the reactor is isolated from the main condenser and the condensate feedwater system is not available. Although the RCIC is not part of the BWR emergency core cooling package, it can and would be used during many accidents to inject water into the reactor.

The RCIC System consists of a steam turbine driven centrifugal pump (with pump lubrication system cooled by the pumped water) and associated valves and piping capable of delivering water to the reactor vessel. The turbine is driven by steam extracted from a main steam line upstream of the MSIVs. In some plants, the RCIC can also be driven by steam generated by the auxiliary boiler system. Water is taken from either the condensate storage tank (preferred) or the pressure suppression pool and injected into the reactor vessel via a feedwater line. Unlike the HPCI System, there is presently no provision for automatic shift of RCIC pump suction from the condensate storage tank to the suppression pool.* RCIC turbine exhaust is directed to the suppression pool. The normal set point for RCIC pump flow is ~0.038 m^3/s (600 gpm), however, the operator can manually control RCIC flow if desired. The steam demand of the RCIC turbine is ~0.96 kg/s (7600 lbm/h).

The RCIC System can also be used in conjunction with the steam condensing mode of the Residual Heat Removal (RHR) System in some plants of later design (BWR-5s and 6s). Steam directed to one or both RHR heat exchangers where it is condensed. The RCIC pump is aligned to take suction from the heat exchanger and return the condensate to the reactor vessel. RCIC pump speed is controlled in the manual mode and adjusted to maintain level in the RHR heat exchanger. As the suppression pool water heats up after extended operation, resulting from condensing RCIC turbine exhaust steam, one of the RHR heat exchangers is used to cool suppression pool water. This system is available on late model BWR-4s, BWR-5, and BWR-6 plants and is limited to use only when HPCI is not isolated — as the steam supply line penetration for steam condensing mode operation is downstream of the HPCI steam isolation valves.

All components normally required for initiation of the RCIC System are completely independent of ac power, plant service, and external cooling water systems. The principal RCIC equipment are located outside primary containment. Table 2.5 summarizes the RCIC initiation, trip, and isolation control parameters.

2.6.7 RHR Low Pressure Coolant Injection (LPCI) Mode

The low pressure coolant injection (LPCI) mode of the RHR System (Sect. 3.5.2) is the dominant operating mode and normal valve lineup configuration of the residual heat removal system (Fig. 2.24). All other modes of RHR System operation are submissive to LPCI, and the RHR System will automatically align to the LPCI mode when ECCS initiation conditions are sensed (see Table 2.6). LPCI is designed to restore and maintain the reactor coolant inventory after a loss of coolant accident in which the reactor is depressurized or after Automatic Depressurization System (ADS) actuation. The LPCI System is, therefore, a low head, high flow injection mode [0.1 MPa (290 psid) shutoff head, 1.26 m^3/s (20,000 gpm) (BWR-6) or 2.52 m/s (40,000 gpm) (BWR-4) rated flow at 138 KPad (20 psid) typical]. The capacity of the LPCI system is large enough to completely fill an intact reactor vessel in less than 5 minutes.

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*The NRC is currently examining such alternatives.

During LPCI operation, the RHR pumps take suction from the suppression pool and discharge into the reactor vessel via the recirculation loops (BWR-4) or directly into the region between the outermost fuel assemblies and the inside of the core shroud (BWR-5 and -6). The LPCI (RHR) pumps are motor driven centrifugal pumps. BWR-4 plants have the capability to draw suction from the condensate storage tank, and in some plants (BWR-6s), the RHR pumps can be realigned to take suction on the Fuel Pool Cooling and Cleanup System. BWR-5 and -6 plants have three pump/3 loop LPCI systems while BWR-4s have four pump/4 loop systems. BWR-5 and BWR-6 plants do not utilize the RHR heat exchangers to cool the LPCI water.

2.6.8 Low Pressure Core Spray (LPCS) System

The purpose of the low pressure core spray LPCS system (Fig. 2.25) is to protect against overheating of the fuel in the event the core is uncovered by the loss of primary coolant following a break or rupture of the primary system. This cooling effect is accomplished by directing spray jets of cooling water directly onto the fuel assemblies from spray nozzles mounted in sparger rings located within the shroud just above the reactor core.

Each loop of the core spray system consists of one or more motor driven centrifugal pumps, a spray sparger in the reactor vessel above the core, and such piping, valves, and control logic as are necessary to convey water from the pressure suppression pool to the reactor vessel. BWR-4s employ two 50% capacity core spray loops, each with its own sparger. BWR-5 and -6 plants utilize a single pump, single loop system. Typical total rated LPCS injection rates are 0.38 m^3/s (6000 gpm) (BWR-6) or 0.789 m^3/s (12,500 gpm) (BWR-4) at suppression pool-to-reactor vessel pressure differentials of 1.03 MPa (150 psi). The shutoff pressure of the LPCS pumps is typically 2.07 MPa (300 psig) gauge. BWR-4 plants have the capability to take LPCS suction from the condensate storage tank as an alternative to the pressure suppression pool. LPCS initiation, trip, suction switch, and isolation control parameters are given in Table 2.7.

2.6.9 Standby Liquid Control (SLC) System

The purpose of the Standby Liquid Control (SLC) System is to inject enough neutron absorbing solution into the reactor vessel to shut down the reactor from a full power condition, independent of any control rod motion, and to maintain it in a subcritical condition as the plant cool the plant down. The system is not a backup for rapid scram. Under a simulated injection test mode this system is capable of injecting small quantities of demineralized water into the reactor vessel. It is this test mode which will be described in this section.

The SLC System (Fig. 2.26) consists of two 100% capacity positive displacement pumps, a heated storage tank, and the associated valves, piping, and instrumentation necessary to inject the neutron absorbing boron solution into the reactor vessel. In current designs all SLC injection enters the reactor vessel through a line which penetrates the reactor vessel through a stub tube in the bottom head and terminates just below the core plate (Fig. 2.4). Inside the vessel the injection line has circumferential holes drilled throughout its entire length. In the future, some reactors may be modified to allow the SLC system to inject into the core spray, feedwater lines, or the diffuser throats of the jet pumps. The goal of such modification is to improve the mixing and dispersion of the boron solution throughout the reactor core.

Use of the SLC System for emergency vessel makeup water injection requires operator action to isolate the SLC storage tank, valve the demineralized water tank supply test tank into the SLC pump suction lines, and fire the explosive valves to inject the water into the reactor. Makeup water to the SLC test tank is supplied by the demineralized water system. Present SLC designs would accommodate an injection flow of 0.004 to 0.006 m^3/s (60 to 100 gpm) via this mode of operation. The total manual control of this system is illustrated by Table 2.8.

2.6.10 Miscellaneous systems with injection capabilities

In addition to the reactor injection systems previously described, there are various other systems that could be utilized in an emergency to inject water into the reactor vessel. In general, these systems were not designed specifically for reactor injection applications, however, emergency operating instructions for their use in this fashion do exist in some plants.[2.8] The availability of these alternative injection systems is highly plant specific.

In some plants the RHR drain and RHR service water systems can be utilized to inject water into the reactor vessel if the primary system pressure is sufficiently low. The RHR drain pumps can take suction on either the suppression pool or the condensate storage tank and inject into either the recirculation loops or the vessel head spray nozzle. The rated flow of these pumps is 0.10 m^3/s (1600 gpm) with a differential shutoff pressure of 0.45 MPa (65 psi). The RHR service water pumps take suction from the ultimate heat sink (river, etc.) and can inject into either the recirculation loops or the vessel head spray nozzle. The rated flow of typical RHR service water pumps is 0.57 m^3/s (9000 gpm) with a differential shutoff pressure of 1.12 MPa (162 psi).

2.6.11 Summary of vessel injection capabilities

The object of this section has been to briefly review those BWR systems which might be available to inject water into the reactor vessel under accident conditions. Table 2.9 summarizes the suction sources, injection points, and operating pressure ranges for the injection systems described in this chapter.

2.7 Vessel Water Cooling Systems

2.7.1 RHR shutdown cooling

The shutdown cooling and reactor vessel head spray subsystem is an operating mode of the Residual Heat Removal (RHR) System (Sect. 3.5.2). Operation of the RHR System in this mode (Fig. 2.27) necessitates a manual override of the normal RHR LPCI configuration, and is possible only after reactor pressure has been reduced to less than 0.93 MPa (135 psi) (gauge). The shutdown cooling mode must be manually aligned by the operator. During this mode of operation, reactor water is removed from one of the recirculation lines by the RHR main system pumps, circulated through the RHR heat exchangers, and returned to the reactor vessel via the same recirculation loop or a feedwater line. Part of the flow [0.03 m^3/s (~500 gpm)] can be diverted to the head spray nozzle.

2.7.2 RHR steam condensing mode

The RHR system steam condensing mode, shown in Fig. 2.28, is used in conjunction with the RCIC system to remove decay heat and minimize the makeup water requirements in late design BWR-4, BWR-5, and BWR-6 plants.

The flow path for the steam condensing mode is as follows: reactor steam passes through the combined RCIC turbine/RHR heat exchanger steam line to the RHR heat exchanger(s); condensate from the RHR heat exchanger(s) is forced (by heat exchanger pressure) to the suction of the RCIC pump; condensate is pumped by the RCIC system to the reactor vessel via a feedwater line. This mode must be manually aligned by the control room operator, since the LPCI mode is the dominant operating mode of the RHR system.

2.7.3 Reactor Water Cleanup (RWCU) System

The purpose of the Reactor Water Cleanup (RWCU) System (Fig. 2.29) is to maintain reactor water quality by removing fission products, corrosion products, and other soluble and insoluble impurities; to provide a path for removal of reactor coolant from the reactor vessel in case of excess coolant inventory; and to maintain circulation in the reactor vessel bottom head to minimize thermal stratification.

The RWCU System consists of a pumping system which takes suction on both recirculation loop suction lines and the reactor vessel bottom head, pumps the water through heat exchange and ion exchange facilities, and pumps the water back to the reactor vessel via the feedwater piping.

The flow path of the RWCU System includes high pressure flow through two 50% capacity pumps, three regenerative heat exchangers and two nonregenerative heat exchangers. Depending on desired system operation, flow can be routed through two 50% capacity filter demineralizers. The RWCU System has the capability to direct flow to the main condenser, the Liquid Radwaste System, or to the reactor vessel via the feedwater lines. Flow through the filter demineralizers and/or heat exchangers can be bypassed as desired depending on plant operating conditions.

The three regenerative heat exchangers, connected in series, provide the first stage of temperature reduction for the reactor influent to the RWCU System. The water is cooled from 550 to 383 K (~530 to 230°F). The two nonregenerative heat exchangers, connected in a series string, are the final stage for cooling the reactor water to 322 K (~120°F) for filter/ demineralizer service. After passing through the demineralizers, the RWCU water flows again through the regenerative heat exchanger, where it is heated from ~322 to 494 K (~120 to 430°F) before returning to the reactor via a feedwater line. The RWCU System is normally operated continuously during all phases of reactor operation, startup, shutdown, and refueling. RWCU control parameters are listed in Table 2.10.

2.8 Automatic Depressurization System (ADS)

The purpose of the Automatic Depressurization System (ADS) is to depressurize the reactor vessel so that the low pressure emergency core cooling systems (ECCS) can inject water to mitigate the consequences of a small or intermediate sized loss of coolant accident should the high pressure emergency core cooling systems fail.

The Automatic Depressurization System consists of redundant signal logics arranged in two separate channels that control separate solenoid operated air pilot valves on each safety/relief valve (SRV) assigned to the ADS function. The number of ADS equipped SRVs varies from plant to plant. The ADS associated SRVs open automatically if required as part of the emergency core cooling system (ECCS) logic to provide reactor vessel depressurization for events involving small breaks in the nuclear system process barrier. The ADS is initiated by coincidence of low reactor vessel water level and high drywell pressure, provided that one of the low pressure emergency core cooling systems is operating.

Each of the SRVs provided for automatic depressurization is equipped with an accumulator and check valve (to isolate the accumulator from the air supply upon loss of supply air pressure). These accumulators assure that the ADS valves can be opened and held open following failure of the instrument air supply to the ADS valves. The accumulators are supplied 1.03 MPa (150 psi) (gauge) pneumatic pressure from the Service and Instrument Air System. The accumulator is sized to be capable of opening the valves and holding them open against a maximum drywell pressure of 0.16 MPa (23 psi) (gauge) and contain sufficient air for one additional activation at 70% of the maximum drywell pressure rating. With normal drywell pressure, each ADS accumulator will provide sufficient air pressure for five SRV actuations. ADS control parameters are listed in Table 2.11.

2.9 BWR Neutronics and Power Control

2.9.1 Neutronics*

BWR designs utilize a light-water moderated core, fueled with slightly enriched uranium-dioxide. The use of water as a moderator produces a neutron energy spectrum in which fissions are caused principally by thermal neutrons. At normal operating conditions, the moderator boils, producing a spatially variable distribution of steam voids in the core. The BWR design provides a system for which reactivity changes are inversely proportional to the steam void content in the moderator. This void feedback effect is one of the inherent safety features of the BWR system. Any system input which increases reactor power, either in a local or gross sense, produces additional steam voids which reduce reactivity and thereby reduce the power. Conversely, any system input which raises reactor pressure or directly reduces core voiding will result in a power increase.

The fuel for the BWR is uranium-dioxide enriched to ~3 wt % in U^{235}. Early in the fuel life the fissioning of the U^{235} produces the majority of the energy. The presence of U^{238} in the uranium-dioxide fuel leads to the production of appreciable quantities of plutonium during core operation. This plutonium contributes to both reactivity and reactor power production (i.e., ~50% at end-of-life). In addition, direct fissioning of U^{238} by fast neutrons yields ~7 to 10% of the total power and contributes to an increase of delayed neutrons in the core. Since the U^{238} has a strong negative Doppler reactivity coefficient, the peak power during an excursion is limited.

The reactor core is arranged roughly as a right circular cylinder containing a large number of fuel assemblies and control rods. At each refueling period, ~25% of the fuel bundles are discharged from the core and replaced with an equivalent number of fresh fuel assemblies. The fuel bundles having the highest exposure (i.e., the lowest reactivity) are discharged starting with the highest exposure and moving toward less exposure. The bundles are then shuffled in order to minimize radial power peaking and maximize the end-of-cycle reactivity. This is accomplished by loading the lowest reactivity fuel on the periphery, loading the relatively high reactivity fuel in a region next to the periphery toward the core center, and loading the medium reactivity fuel in the central region of the core. Within each of these zones, the fuel bundles are arranged in a nearly homogeneous manner in order to minimize reactivity mismatch.

The bundle reactivity is a complex function of several important physical properties. The important properties consist of the average bundle enrichment, the gadolinia rod location and gadolinia concentration, the void fraction and the accumulated exposure.

The radial power distribution is also a complex function of the control rod pattern, the fuel bundle type, the loading pattern and the

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*Major portions of this section are excerpted from Ref. 2.4.

void condition for that bundle. The radial power distribution is influenced by both the radial reactivity zones and the control rods. The control rods are selectively inserted, or withdrawn, to flatten the radial power distribution consistent with the reactivity control needed. Near the end-of-cycle, the region of high reactivity adjacent to the periphery provides the necessary radial power flattening without recourse to control rods.

The effect of voids (during normal operation) is to skew the power toward the bottom of the core; the effect of the bottom entry control rods is to reduce the power in the bottom of the core; and the effect of the exposure distribution is to flatten the power. Since the void distribution is determined primarily from the power shape, the mechanism available for further optimizing the axial power shape is the control rods.

There are three primary reactivity coefficients which characterize the dynamic behavior of BWRs over all operating states: (1) Doppler reactivity coefficient; (2) moderator temperature reactivity coefficient; and (3) moderator void reactivity coefficient. Also associated with the BWR is a power reactivity coefficient; however, this coefficient is merely a combination of the Doppler and void reactivity coefficients in the power operating range. The most important of these coefficients is the void reactivity coefficient. The void coefficient must be large enough to prevent power oscillation due to spatial xenon changes yet small enough that pressurization transients do not unduly limit plant operation. In addition, the void coefficient in a BWR has the ability to flatten the radial power distribution and provides ease of reactor control due to the void feedback mechanism. The overall void coefficient is always negative over the complete operating range since the BWR design is undermoderated. The reactivity change due to the formation of voids results from the reduction in neutron slowing down due to the decrease in the water-to-mass fuel ratio. A typical value for the moderator void coefficient is -1.6×10^{-3} $\Delta k/k/\%$ void.

The moderator temperature coefficient is the least important of the reactivity coefficients since its effect is limited to a very small portion of the reactor operating range. Once the reactor reaches the normal operating range, boiling begins and the moderator temperature remains essentially constant. As with the void coefficient the moderator temperature coefficient is associated with a change in the moderating power of the water. The temperature coefficient is negative for most of the operating cycle; however, near the end-of-cycle the overall moderator temperature coefficient becomes slightly positive. This is due to the fact that the uncontrolled BWR lattice is slightly overmoderated near the end-of-cycle; this, combined with the fact that more control rods must be with-drawn from the reactor core near the end-of-cycle to establish criticality, results in the slightly positive overall moderator temperature coefficient.

The range of values of moderator temperature coefficients encountered in current BWR lattices does not include any that are significant from the safety point of view. Typically, the temperature coefficient may range from $+4 \times 10^{-5}$ $\Delta k/k°F$ to -14×10^{-5} $\Delta k/k°F$, depending on base temperature and core exposure. The small magnitude of this coefficient, relative to that associated with steam voids and combined with the long time-constant associated with transfer of heat from the fuel to the coolant, makes the reactivity contribution of moderator temperature change insignificant during rapid transients.

The Doppler reactivity coefficient is the change in reactivity due to a change in the temperature of the fuel. This reactivity change is due to the broadening of the resonance cross sections as the fuel temperature increases. At beginning-of-life, the Doppler contribution is primarily due to U^{238}; however, the buildup of Pu^{240} with exposure adds to the Doppler coefficient.

The power coefficient is determined from the composite of all the significant individual sources of reactivity change associated with a differential change in reactor thermal power assuming xenon reactivity remains constant. At end-of-equilibrium-cycle, the power coefficient at 105% steam flow conditions is approximately $-0.05\ \Delta k/k \div \Delta P/P$. This value is well within the range required for adequately damping power and spatial-xenon disturbances.

2.9.2 BWR power generation control

After the generator is synchronized to the utility's transmission grid reactor power output can be adjusted to meet the grid system requirements by adjustment of control rod position, manual or automatic adjustment of reactor recirculation flow, or a combination of these two methods.

Withdrawing a control rod reduces the neutron absorption and adds 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 pressure regulators to open the turbine control valves sufficiently to maintain a programmed throttle pressure. When a control rod is inserted, the reverse effect occurs. The rate of power increase is limited by the rate at which control rods can be withdrawn. When the reactor is operating above 30% power, rod withdrawals are restricted to two notches at a time to prevent local fuel damage.

Reactor power output can be varied over a power range of ~35% of rated power by adjustment of the reactor recirculation flow, while maintaining a nearly uniform power distribution. Reactor power change is accomplished by using the negative void coefficient. An increase in recirculation flow temporarily reduces the volume of steam in the core by raising the boiling boundary. This addition of reactivity of the core causes the reactor power level to increase. The increased steam generation rate then returns the steam volume in the core to approximately its original value, and a new constant power level is established. When recirculation flow is reduced, the power level is reduced in a similar manner.

During initial power operation, the operating curve or power/flow map (Fig. 2.31) 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 for steady state conditions. 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.

Although control rod movement is not required when the load is changed by recirculation flow adjustment, the long-term reactivity effects of fuel burnup can be compensated for by control rod adjustment. The reader should note from Fig. 2.30 that BWRs are capable of operating at significant power levels under natural circulation conditions.

2.10 Reactor Protection System

2.10.1 System description*

The purpose of the Reactor Protection System (RPS) is to prevent excessive fuel cladding or reactor coolant pressure boundary damage by generating signals to automatically shutdown the reactor when necessary, via rapid insertion of all control rods.

The Reactor Protection System is a solid state protective system which monitors various reactor plant process variables which indicate whether safe operating conditions exist in the reactor plant. In the event that these process variables depart excessively from their normal operating values resulting in potentially unsafe operation of the plant, a reactor scram occurs automatically. A reactor scram is the deenergization of scram pilot solenoid valves (which results in rapid insertion of all control rods) and isolation of the scram discharge volume.

The Reactor Protection System includes power supplies, sensors, trip circuitry, bypass circuitry and switches that generate the scram signals that cause rapid insertion of the reactor control rods (scram) to shutdown the reactor.

Each reactor plant process parameter is monitored by at least four sensors, one for each of the four RPS logic divisions. Each sensor output is sent to the logic of all four RPS divisions. When a parameter reaches its scram setpoint and a sufficient number of sensors reach this unsafe condition, a scram signal is generated from the RPS logic. The scram signal causes electrical power to be interrupted to the scram pilot solenoid valves on each control rod drive (CRD) hydraulic control unit (HCU), and all control rods are rapidly inserted into the reactor core, shutting down the reactor.

In addition, the RPS provides a backup scram method which operates to scram the control rods in the event of a failure to scram by normal means (failure of the scram pilot solenoid valves to properly reposition). This backup scram method is accomplished by energizing two dc solenoid operated valves, each of which requires two scram signals (one each from RPS channels A and B) to reposition (energize) and bleed air

*This section is excerpted from Ref. 2.2.

off all scram inlet and outlet valves. Repositioning of either backup scram valve will accomplish this purpose.

The capability to manually scram the reactor is also provided by two means. Manual scram pushbuttons, located on the control console, can be used to scram the reactor if proper combinations of two switches are operated. Placing the reactor mode switch in the shutdown position also inserts a scram signal into the RPS logic which causes a reactor scram.

Once a scram is initiated, it goes to completion. All control rods are fully inserted, and deliberate operator action is required to return the reactor plant and RPS to normal operation. Conditions which will cause an automatic RPS scram are described in the following sections.

2.10.2 SDIV high water level scram

The scram discharge instrument volume (SDIV) receives the water displaced by the motion of the control rod drive pistons during a scram. Should the scram discharge volume fill up with water to the point where insufficient space remains for the water displaced during a scram, control rod movement would be hindered in the event a scram were required. To prevent this situation, the reactor is scrammed when the water level in the discharge volume attains a value high enough to verify that the volume is filling up, yet low enough to ensure that the remaining capacity in the volume can accommodate a scram.

2.10.3 Drywell pressure scram

High drywell pressure may be caused by a break in the reactor coolant pressure boundary. It is, therefore, prudent to scram the reactor in such a situation to minimize the possibility of fuel damage and to reduce the energy transfer from the core to the coolant, which in turn minimizes the energy that the primary containment would be required to absorb. The high drywell pressure scram setting is selected to be as low as possible without inducing spurious scrams.

2.10.4 Vessel low water level scram

Low water level in the reactor vessel indicates that the reactor core is in danger of being inadequately cooled. Should the water level decrease excessively, fuel damage could result as a steam blanket forms around fuel rods. A reactor scram protects the fuel by reducing the fission heat generation within the core. The scram setting is far enough below normal operational levels to avoid spurious scrams, but high enough above the top of active fuel to assure that enough water inventory is available to account for evaporation loss and displacement of coolant following the most severe abnormal operational transient involving a level decrease in order to preclude uncovering the reactor core.

2.10.5 Vessel high water level scram

During reactor plant power operation, a significant increase in water level indicates excessive feedwater flow. This water level increase adds significant positive reactivity to the reactor core which increases the fission heat generation from the fuel. The reactor scram on high water level anticipates the positive reactivity addition by this increased feedwater flow and prevents possible fuel damage caused by the excessive heat generation rate. The scram setpoint is high enough above the normal operating level to prevent spurious scrams, but low enough to initiate a reactor scram prior to flooding the main steam lines.

2.10.6 High reactor pressure scram

Excessively high pressure within the nuclear system threatens to rupture the reactor coolant pressure boundary. A nuclear system pressure increase during reactor operation collapses steam voids and results in a positive reactivity insertion. This causes increased core heat generation that could lead to fuel failure and system overpressurization. A scram counteracts a pressure increase by rapidly reducing the core fission heat generation. The nuclear system high pressure scram setting is chosen slightly above the reactor vessel maximum normal operating pressure to permit normal operation without spurious scrams, yet provide adequate margin to the maximum allowable nuclear system pressure. The high pressure scram works in conjunction with the safety/relief valves to prevent nuclear system pressure from exceeding the maximum allowable pressure. The high pressure scram setting also protects the core from exceeding thermal and hydraulic limits that may result from pressure increases during events that occur when the reactor is operating below rated power and flow.

2.10.7 Main steam line high radiation scram

High radiation in the vicinity of the main steam lines may indicate a gross fuel cladding failure in the reactor core. A scram is initiated to limit the release of fission products from the fuel. This condition also signals the Primary Containment Isolation System (BWR-4) or the Nuclear Steam Supply Shutoff System (NSSSS) to initiate an isolation of the containment to prevent release of the fission products to the environment. The high radiation setpoint is high enough above background radiation levels to avoid spurious scrams and isolations, yet low enough to rapidly detect gross fission product leakage from the reactor core.

2.10.8 APRM scrams

Scram signals are generated by the average power range monitor (APRM) logic circuits under three different conditions: inoperable APRM circuit, high neutron flux, and high thermal power (high heat flux) for the existing recirculation loop driving flow. If one of the above listed conditions exists, a trip signal from the APRM channel (or channels) detecting this condition is generated.

2.10.9 IRM scrams

An intermediate range monitor (IRM) scram signal is generated if reactor power exceeds a preselected setpoint or if an IRM becomes inoperable. All IRM scram signals are disabled with the reactor mode switch in the run position.

2.10.10 MSIV closure scram

A reactor scram will result from isolating more than one main steam line (closure of one or more MSIVs in more than one main steam line) with the reactor plant at power (reactor mode switch in the run position). The main steam isolation valves have stem mounted limit switches which are used for valve position indication. One limit switch per valve sends a signal to the Reactor Protection System when the valve is partially (>10%) closed. If either valve in a steam line closes and the individual MSIV closure scram bypass switch for that steam line is not in the bypass position, a trip signal is generated. A main steam line isolation can result in a significant addition of positive reactivity to the core from void collapse as nuclear system pressure rises. The main steam line isolation scram setting is selected to give the earliest positive indication of isolation valve closure to limit the resultant pressure rise.

2.10.11 Turbine control valve fast closure scram

Turbine control valve fast closure sends inputs to the Reactor Protection System from oil line pressure switches on each of the four fast acting control valve hydraulic mechanisms. These hydraulic mechanisms are part of the turbine control valve and they are used to effect fast closure of the turbine control valves in the event this action is called for, in the case of a generator load rejection.

The turbine control valve fast closure scram provides additional margin to the nuclear system pressure limit. With the reactor and turbine generator at power, fast closure of the turbine control valves can result in a significant addition of positive reactivity to the core because of void collapse as nuclear system pressure rises. The turbine control valve fast closure scram is required to provide a satisfactory margin to core thermal hydraulic limits for this transient. This scram is automatically disabled when turbine first stage pressure is below 30% of rated conditions.

Turbine stop valve closure inputs to the Reactor Protection System are generated from valve stem position switches mounted on the main turbine stop valve. Each of the switches (one per valve) opens before the valve is more than 10% closed to provide the earliest indication of valve closure. Closure of the turbine stop valves with the reactor at power can result in a significant addition of positive reactivity to the core as the nuclear system pressure rise causes steam voids to collapse. The turbine stop valve closure signal initiates a scram earlier than either the neutron monitoring systems or nuclear system high pressure scram logic. It is required to provide a satisfactory margin below core thermal hydraulic limits for this transient. This scram is automatically disabled when turbine first stage pressure is below 30% of rated conditions.

2.10.12 Reactor mode switch scram

Placing the reactor mode switch to the shutdown position initiates a reactor scram. This scram is not required to protect the fuel or nuclear system process barrier, and it bears no relationship to minimizing the release of radioactive material from any barrier. The scram signal is removed after a 10 s time delay, permitting the scram to be reset which restores the normal valve lineup for the hydraulic control units and scram discharge instrument volume.

2.11 Primary Containment and Vessel Isolation Control System — Reactor Functions

The purpose of the Primary Containment Isolation System (PCIS) [Nuclear Steam Supply Shutoff System (NSSSS) in BWR-5 and BWR-6 plants] is to isolate the reactor vessel and various reactor plant systems which carry radioactive fluids or gases from the primary containment in order to prevent the release of radioactive materials to the environment in excess of specified limits. Only those PCIS/NSSSS functions which are associated with the reactor will be discussed in this section.

The PCIS determines, from information provided by reactor plant process instrumentration, which systems should be isolated and provides isolation signals to these. Isolation demand signals are generally divided into isolation signals for systems considered to be within the Nuclear Steam Supply System and for balance of plant (BOP) systems. Local sensor elements provide information concerning selected reactor plant parameters to the PCIS solid state logic in digital or analog form. The PCIS logic decides whether the need exists for an isolation based on the available input data, and either remains passive (no isolation), or provides an isolation demand signal to the appropriate reactor plant valve. Once the system is initiated, the isolation will proceed until completion, and a return to normal operation after the isolation will require deliberate operator action.

The PCIS/NSSSS isolation logic is divided into five discrete groups. These groups are associated with the system isolation listed below:

Group 1	Main steam isolation
Group 2	RWCU isolation
Group 3	Reactor water sample line isolation
Group 4	RHR isolation
Group 5	Balance of plant isolation

Parameters which control the isolation decision for Groups 1 through 5 are described below.

The main steam system isolation (Group 1) is provided to control the loss of coolant from the reactor vessel and release of radioactive materials to the environment. The NSSSS logic responds to signals that indicate a breach of the reactor coolant pressure boundary (RCPB), a breach of the fuel cladding, a failure of the Electro Hydraulic Control System or a loss of the primary heat sink (main condenser). When selected process parameters reach preset levels, isolation demand signals generated by a portion of the NSSSS cause the main steam isolation valve and main steam line drain valves to close.

The Reactor Water Cleanup System (Group 2) isolates on signals from the Leak Detection System when the symptoms of a RCPB leak, RWCU System leak or a malfunction resulting in a loss of RWCU System loop cooling are detected. In addition, contacts on the Standby Liquid Control (SLC) System pump control switches cause a RWCU System isolation upon SLC System initiation.

Isolation of the reactor water sample line (Group 3) occurs due to sensed main steam line high radiation or low reactor vessel water level. The reactor water sample line provides an alternate path for sampling the primary coolant via the Recirculation System. In performing this function, the piping must penetrate the reactor coolant pressure boundary, and therefore is a potential path for leakage from the RCPB. Because of this, when water level in the reactor vessel decreases to Level 2, the reactor water sample line isolation occurs which automatically closes the sample valves to isolate this possible source of leakage. Radiation levels in accessible plant areas could increase to very high levels if sampling through this line was in progress and a gross failure of the fuel barrier occurred. An excellent indication of gross fuel barrier failure is main steam line high radiation. Therefore the reactor water sample line valves isolate on a main steam line high radiation signal.

The Residual Heat Removal (RHR) System (Group 4) isolates in response to low reactor water level, high drywell pressure, high reactor pressure and RHR System area high temperature or high RHR System room ventilation differential temperature signals. On indication of low reactor water level, high RHR System area temperature, or high RHR System room vent differential temperature, the shutdown cooling suction and discharge valves, the head spray valve, the RHR System process sample valves, and the RHR System discharge valves to the Liquid Radwaste System all close. High drywell pressure causes isolation of only the sample valves and radwaste discharge valves. Finally, if reactor pressure increases to a value such that the saturation temperature corresponding to that pressure is approaching the temperature rating of the RHR system pumps, the RHR System isolation logic closes the shutdown cooling suction and discharge valves and the head spray valve to protect the RHR pumps.

The balance of plant (Group 5) isolation signals are generated in response to reactor plant parameter levels indicative of a breach in the RCPB. These are low reactor water level or high drywell pressure. These signals isolate BOP systems at the primary containment boundary.

The systems affected are the Service and Instrument Air System, Demineralized Water System, Standby Service Water System, Closed Cooling Water System, Chilled Water System, Plant Equipment and Floor Drain System, Fire Protection System, and the Containment Combustible Gas Control System. In addition, some of the Fuel Pool Cooling and Cleanup System valves are isolated by the BOP logic.

2.12 Reactor Systems Summary

This chapter has presented an abbreviated description of those BWR primary structures and systems which have the capacity to influence the outcome of severe accidents in BWR-4, BWR-5, and BWR-6 facilities. Briefly, the important components are in-vessel structures, vessel water injection systems, vessel water cooling systems, vessel pressure control systems, power control systems, reactor protection (scram) systems, and isolation control systems.

Figures 2.31—2.34 are a representation of a generic boiling water reactor that incorporates all of the component structures and system described in this chapter. The reader is cautioned that no single existing BWR would incorporate all of the features shown in Figs. 2.31—2.34.

The utility of Figs. 2.31—2.34 can be seen by examining the HPCI system drawing included in Fig. 2.32. The drawing indicates that the HPCI system consists of a single turbine driven pump which normally draws suction from the condensate storage tank. The pressure suppression pool is a secondary source of water. HPCI flow is routed to the feedwater line. The turbine normally draws steam from a main steam line but can be driven by the plant auxiliary boiler, and turbine exhaust is routed to the pressure suppression pool. These figures are useful for assessing the ultimate analytical capabilities which a computer code must possess to ensure that all of the significant features of any BWR-4, 5, or 6 vessel injection systems can be represented. The figure displays in a simplified fashion the various vessel water injection systems described in this chapter, together with the locations from which their suction is taken and their injection is routed. Most of the basic information necessary for incorporation of injection system models in BWR analysis codes is displayed in these figures and Tables 2.3—2.11. These figures can, therefore, be used by BWR code developers to identify desirable vessel noding schemes and system interaction points.

Chapter 2 References

2.1. *BWR 4 Boiling Water Reactor Systems Manual*, U.S. Nuclear Regulatory Commission.

2.2. *BWR 6 Boiling Water Reactor Systems Manual*, U.S. Nuclear Regulatory Commission.

2.3. *Browns Ferry Nuclear Plant Final Safety Analysis Report*, Tennessee Valley Authority.

2.4. *General Electric Standard Safety Analysis Report*, GESAR II, General Electric Inc., 1981.

2.5. *LaSalle County Generating Station Final Safety Analysis Report*, p. L.60-14, Commonwealth Edison Inc.

2.6. Letter from R. C. Lewis, Director of the Division of Project and Resident Program, NRC, to A. G. Parris, Manager of Power, TVA, Jan. 5, 1983.

2.7. Hodge, S. A. et al., *SBLOCA Outside Containment at Browns Ferry Unit One - Accident Sequence Analysis*, NUREG/CR-2672 Vol. 1, ORNL/TM-8119/V1, Oak Ridge National Laboratory, November 1982.

2.8. Browns Ferry Nuclear Plant Emergency Operating Procedure No. 41, Tennessee Valley Authority, November 1979.

Table 2.1. Reactor vessel penetrations

Type	Number and size (in.)
Recirculation outlet	2 - 36 to 22
Steam outlet	4 - 26
Recirculation inlet	10 - 12
Feedwater inlet	6 - 12
Core spray inlet	2 - 10
Low pressure coolant injection inlet	3 - 12
Instrument (one of these is Head Spray)	2 - 6
Control rod drive mechanism stub tubes	137 to 193 - 6
Jet pump instrumentation	? - 4
Vent	1 - 4
Instrumentation	6 - 2
Control rod drive hydraulic system return	1 - 2
Core differential pressure and liquid control	1 - 2
Drain	1 - 2
In-core flux instrumentation	53 to 55 - 2
Head seal leak detection	2 - 1

Table 2.2. Fuel assembly design specifications

Fuel assembly			
Fuel bundle	8 x 8	8 x 8R	P8 x 8R
Geometry	8 x 8	8 x 8	8 x 8
Rod pitch (in.)	0.640	0.640	0.640
Fuel rods			
Fill gas	Helium	Helium	Helium
Fill pressure (atm)	1	1	3
Getter	Yes	Yes	Yes
Number of fuel rods	63	62	62
Fuel			
Material	Sintered UO_2	Sintered UO_2	Sintered UO_2
Pellet diameter (in.)	0.416	0.410	0.410
Pellet length (in.)	0.420	0.410	0.410
Pellet immersion density (%TD)	95.0	95.0	95.0
Cladding			
Material	Zr-2	Zr-2	Zr-2
Outside diameter (in.)	0.493	0.483	0.483
Thickness (in.)	0.034	0.032	0.032
Water rod			
Material	Zr-2	Zr-2	Zr-2
Outside diameter (in.)	0.493	0.591	0.591
Thickness	0.034	0.030	0.030
Number of water rods	1	2	2
Fuel channel			
Material	Zr-4	Zr-4	Zr-4
Inside dimension (in.)	5.278	5.278	5.278
Wall thickness (in.)	0.080 or 0.100	0.080 or 0.100	0.080 or 0.100

Table 2.3. HPCI control parameters

Control parameter	Function			
	Initiation	Trip	Suction switch	Isolation
Low reactor water level	X			
High drywell pressure	X			
High reactor water level		X		
High HPCI turbine exhaust pressure		X		
Low HPCI booster pump suction pressure		X		
HPCI turbine mechanical overspeed		X		
Low condensate storage tank level			X	
High suppression pool level			X	
High HPCI equipment space temperatures				X
High HPCI turbine steam flow				X
Low reactor pressure				X
High turbine exhaust diaphragm pressure				X
Manual	X	X	X	X

Table 2.4. HPCS control parameters

Control parameter	Function			
	Initiation	Trip	Suction switch	Isolation
Low reactor water level	X			
High drywell pressure	X			
High reactor water level		X[a]		
Low condensate storage tank level			X	
High pressure suppression pool level			X	
Manual	X	X	X	X

[a]If a high drywell pressure signal exists in conjunction with a high reactor water level signal, HPCS injection will continue until manually stopped by the operator.

Table 2.5. RCIC control parameters

Control parameter	Function			
	Initiation	Trip	Suction switch	Isolation
Low reactor water level	X			
High reactor water level		X		
Electrical turbine overspeed		X		
Mechanical turbine overspeed		X		
High RCIC turbine exhaust pressure		X		
Low RCIC pump suction pressure		X		
RCIC equipment space high temperature				X
RCIC turbine high steam flow				X
Low reactor pressure				X
High turbine exhaust diaphragm pressure				X
Manual	X	X	X	X

Table 2.6. LPCI control parameters

Control parameter	Function			
	Initiation	Trip	Suction switch	Isolation
Low reactor water level	X			
High drywell pressure and low reactor pressure	X			
High drywell pressure (BWR-5 and -6)	X			
Manual	X	X	X	X

Table 2.7. LPCS control parameters

Control parameter	Function			
	Initiation	Trip	Suction switch	Isolation
Low reactor level	X			
High drywell pressure and low reactor pressure	X			
High reactor pressure (BWR-5 and -6)	X			
Manual	X	X	X	X

Table 2.8. SLC control parameters

Control parameter	Function		
	Initiation	Trip	Isolation
Manual	X	X	X

Table 2.9. BWR injection systems summary

System	Suction source			Pump type		Injection point						Operating pressure range (MPa)		
	PSP	CST	Other	Steam	Electric	Feedwater	Recirc line	Lower plenum	Shroud or bypass	Steam dome	Upper head	<2.76	2.76-5.51	>5.51
HPCI	X	X		X		X						X	X	X
HPCS	X	X			X					X		X	X	X
RCIC	X	X		X		X						X	X	X
LPCI	X	X	X		X		X		X		X			
LPCS	X	X			X					X		X		
SLC			X		X			X				X	X	X
RHR Drain	X	X			X		X				X	X		
RHRSW			X		X		X				X	X		
Condensate and feedwater		X		X	X	X						X	X	X
CRDHS		X			X				X			X	X	X

Table 2.10. RWCU control parameters

Control parameter	Function		
	Initiation	Trip	Isolation
High NRHX[a] outlet water temperature		X	
Low RWCU pump suction flow		X	
High RWCU pump cooling water temperature		X	
High pressure drop across filter/ demineralizer unit or effluent strainer			X[b]
Low reactor water level			X
High RWCU differential flow			X
High steam tunnel temperature			X
High steam tunnel ventilation differential temperature			X
Loss of leak detection logic			X
High RWCU area temperature			X
High RWCU area ventilation differential temperature			X
Initiation of Standby Liquid Control System			X
Manual	X	X	X

[a]Nonregenerative heat exchanger.

[b]Filter/demineralizer unit isolation only.

Table 2.11. ADS control parameters

Parameter	Function
≥ 1 Low pressure ECCS system running	Initiation permissable
Low reactor water level + high drywell[a] pressure	Initiation
Manual	Initiation

[a]As a result of NUREG-0737, "Clarification of TMI Action Plan Requirements," some utilities have committed to eliminate the high drywell pressure requirement when reactor water level remains low for a predetermined time period.

Table 2.12. PCIS/NSSSS control parameters

<u>Group 1 isolation</u>

1. High main steam line flow
2. High main steam tunnel area temperature
3. Main steam tunnel ventilation differential temperature
4. High steam line area temperature
5. Low reactor water level
6. Main steam line high radiation
7. Low main steam line pressure
8. Low main condenser vacuum

<u>Group 2 isolation</u>

1. SLC initiation
2. Low reactor water level
3. High RWCU differential flow
4. High RWCU area temperature
5. High main steam tunnel temperatures
6. Loss of power to the leak Detection System
7. High RWCU area vent supply and exhaust duct differential temperature

<u>Group 3 isolation</u>

1. High main steam line radiation
2. Low reactor water level

<u>Group 4 isolation</u>

1. Low reactor water level
2. High drywell pressure
3. High reactor pressure
4. RHR system area high temperature
5. High RHR system room ventilation differential temperature

<u>Group 5 isolation</u>

1. Low reactor water level
2. High drywell pressure

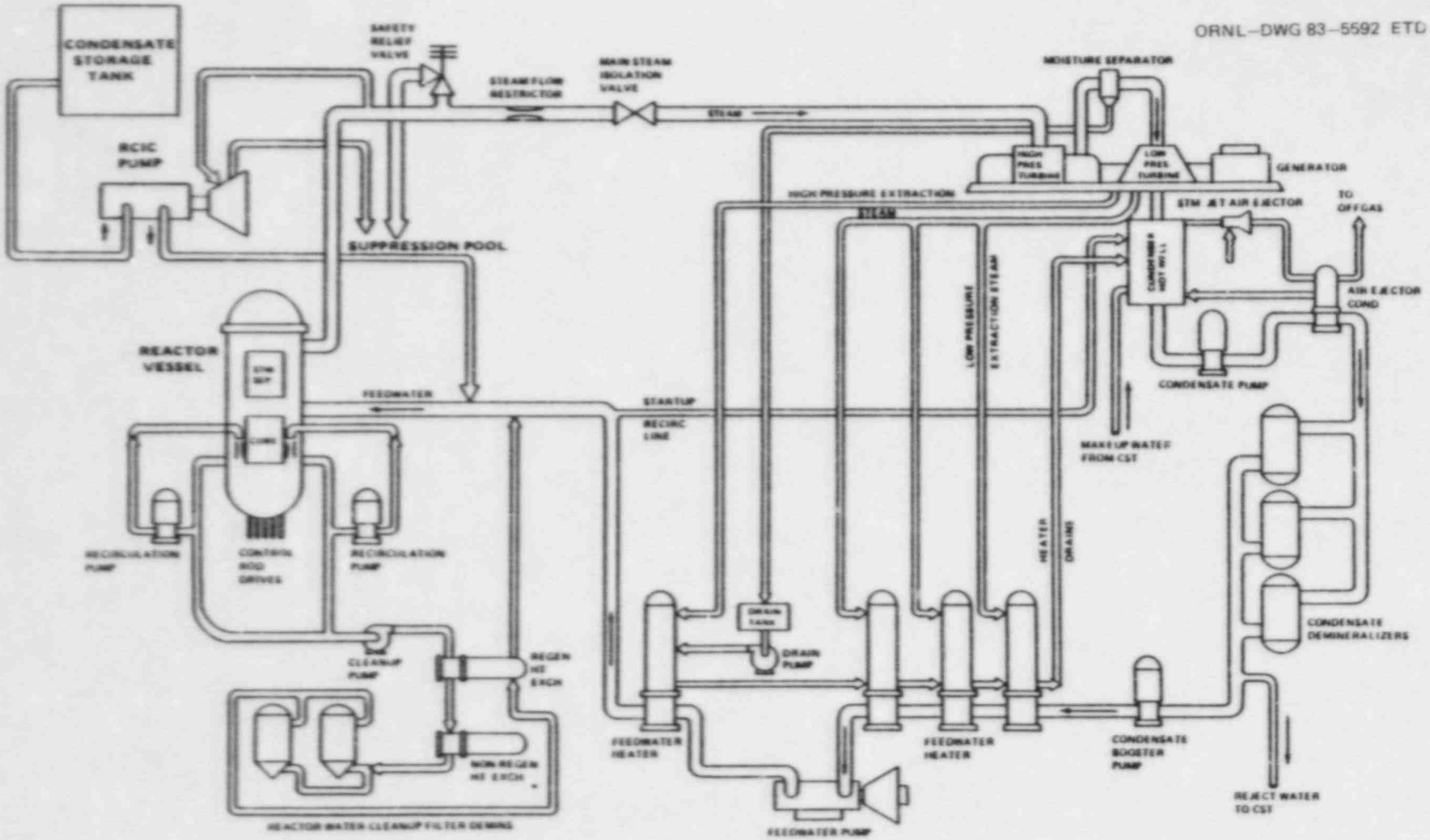


Fig. 2.1 Simplified BWR primary and auxiliary system.

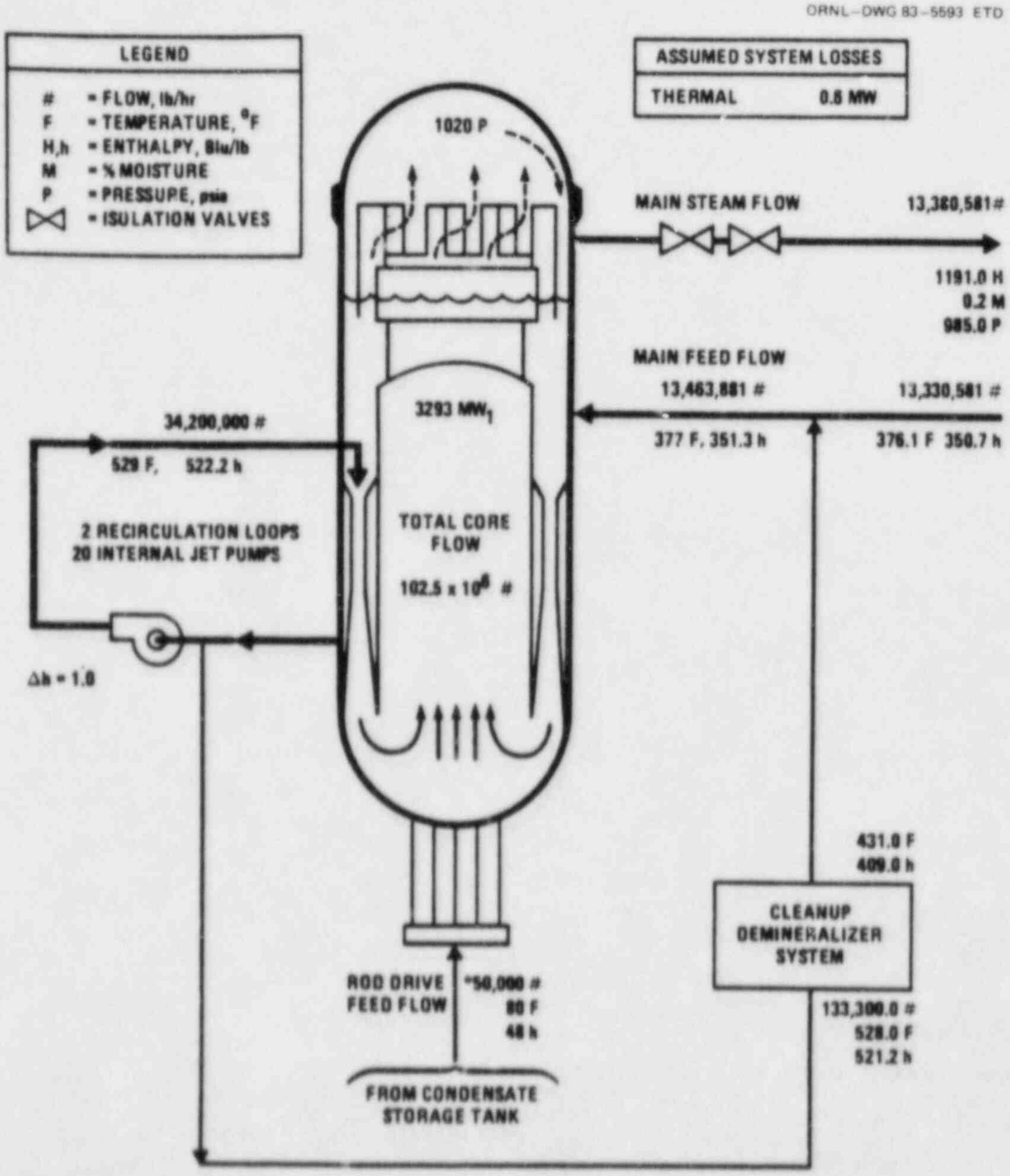


Fig. 2.2. Typical BWR heat balance (at rated power).

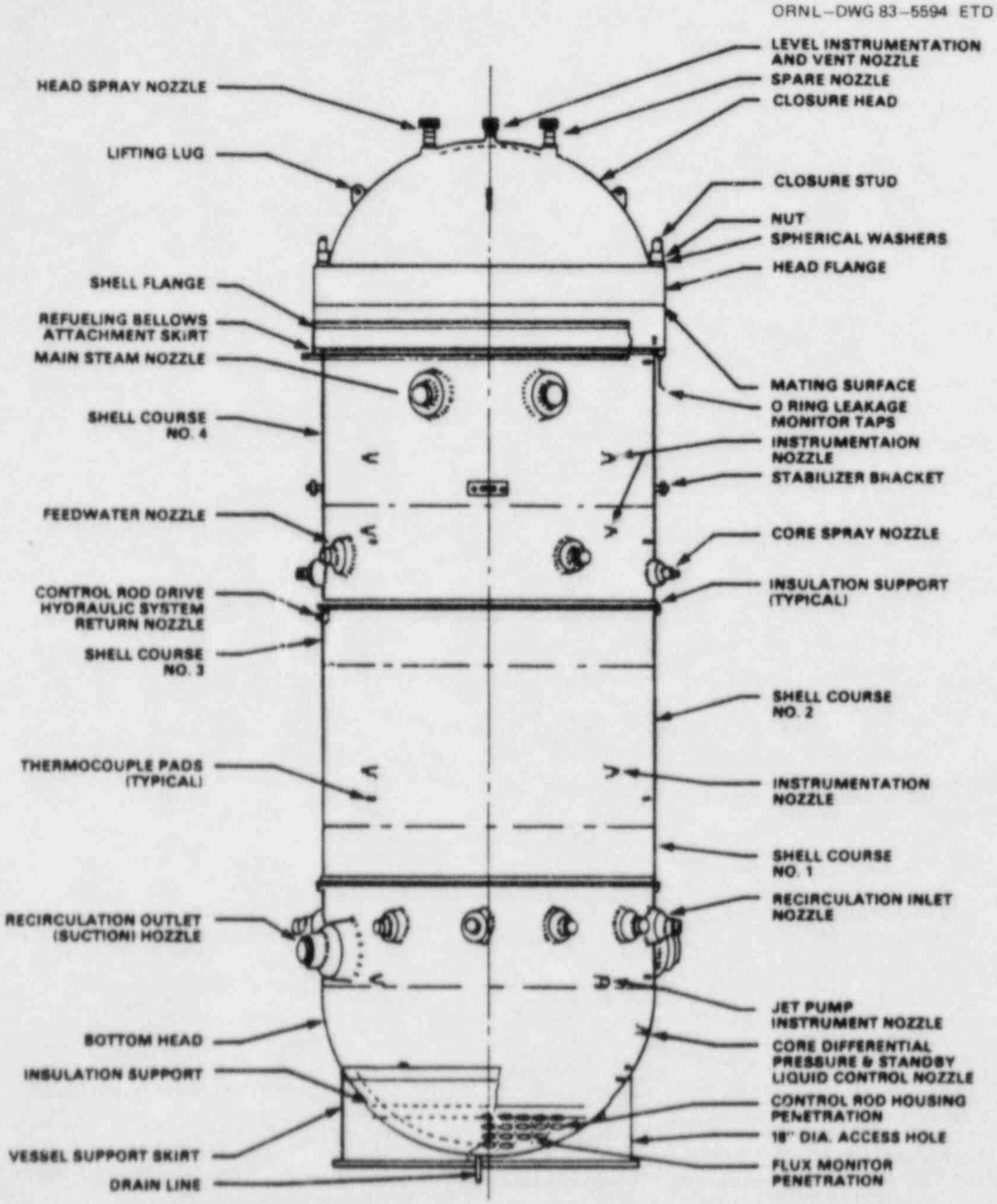


Fig. 2.3. BWR vessel construction.

ORNL-DWG 83-5595 ETD

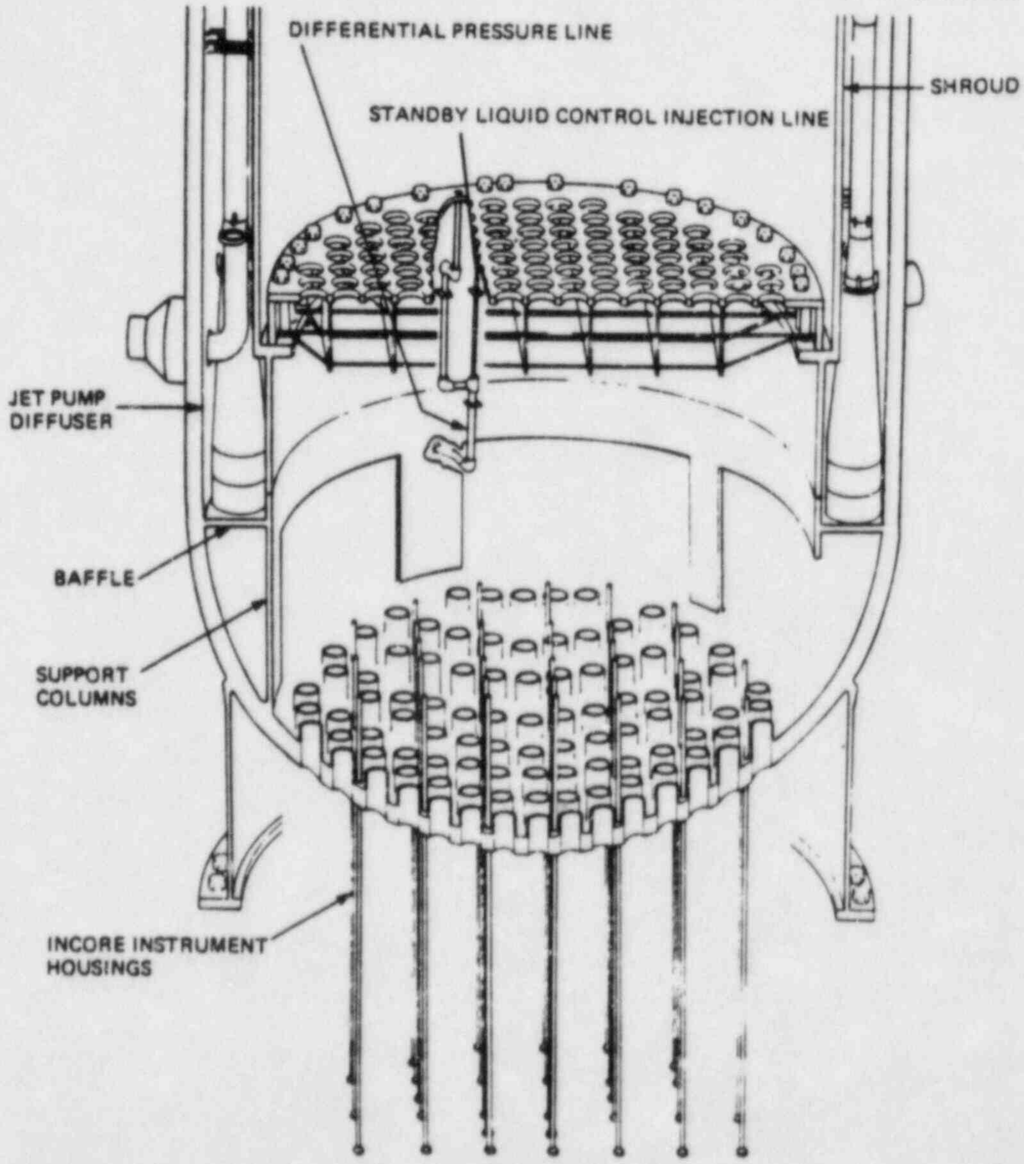


Fig. 2.4. Reactor vessel lower plenum components.

ORNL-DWG 82-5178R ETD

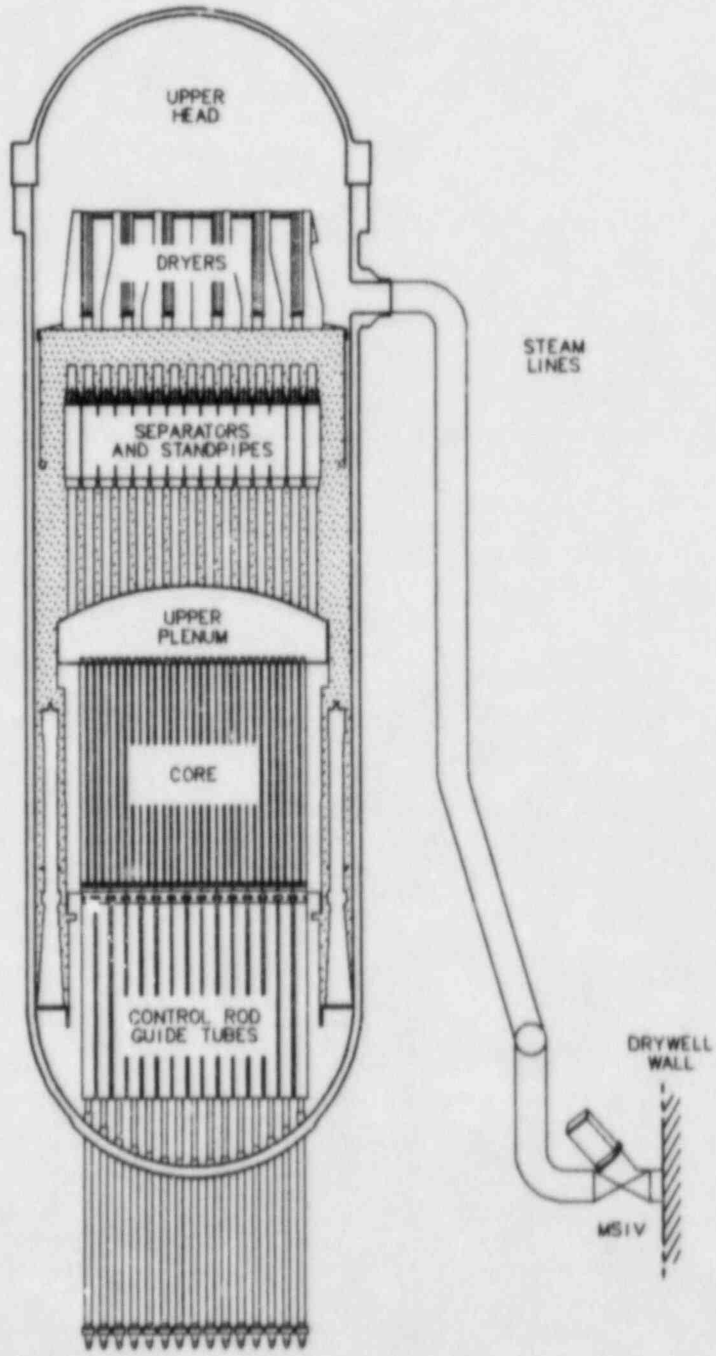


Fig. 2.5. Reactor vessel internal structures.

ORNL-DWG 83-5596 ETD

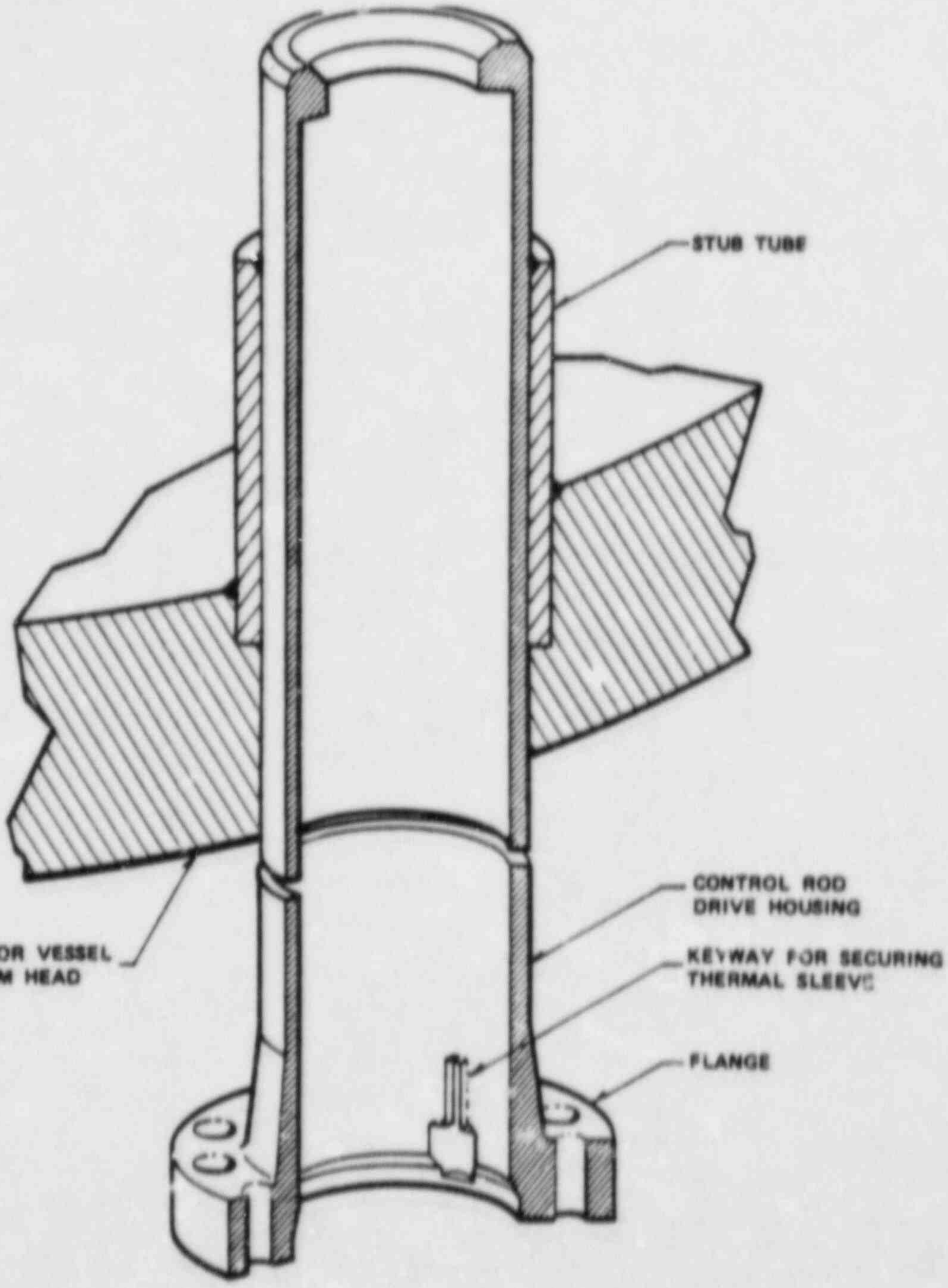


Fig. 2.6. BWR control rod drive housing.

ORNL-DWG 83-5597 ETD

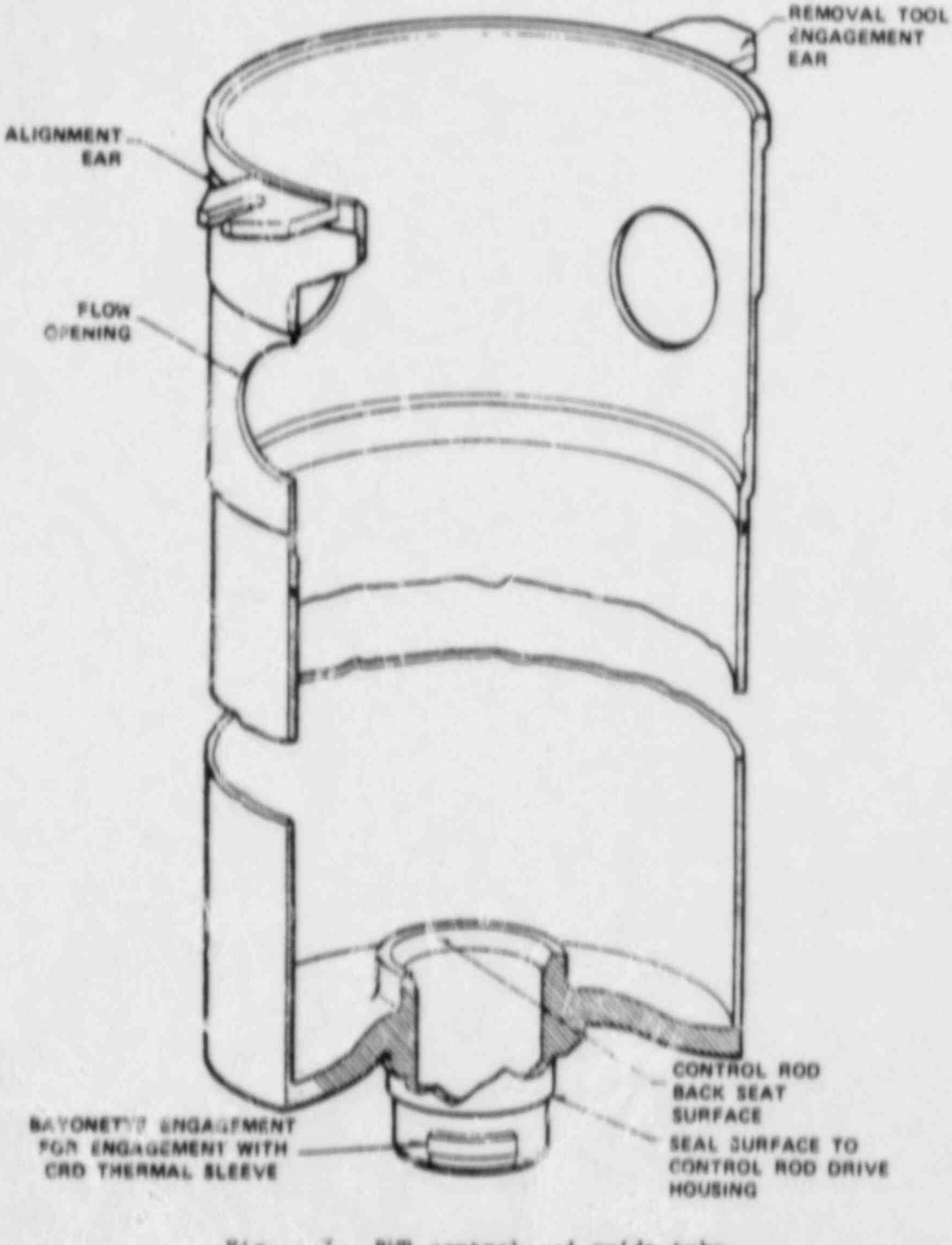


Fig. 2.7. BWR control rod guide tube.

ORNL-DWG 83-5598 ETD

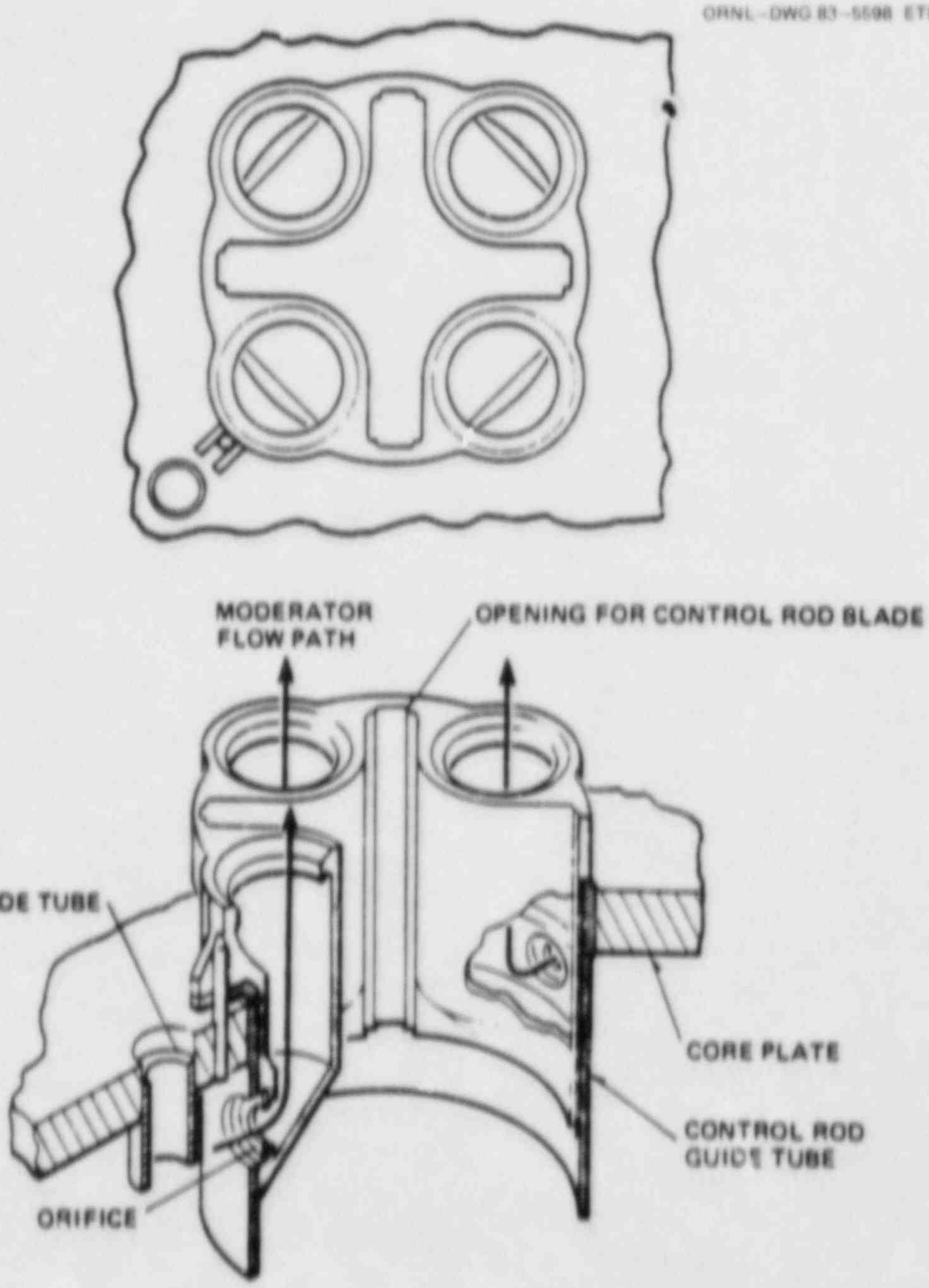


Fig. 2.8. Fuel support piece.

ORNL-DWG 83-5599 ETD

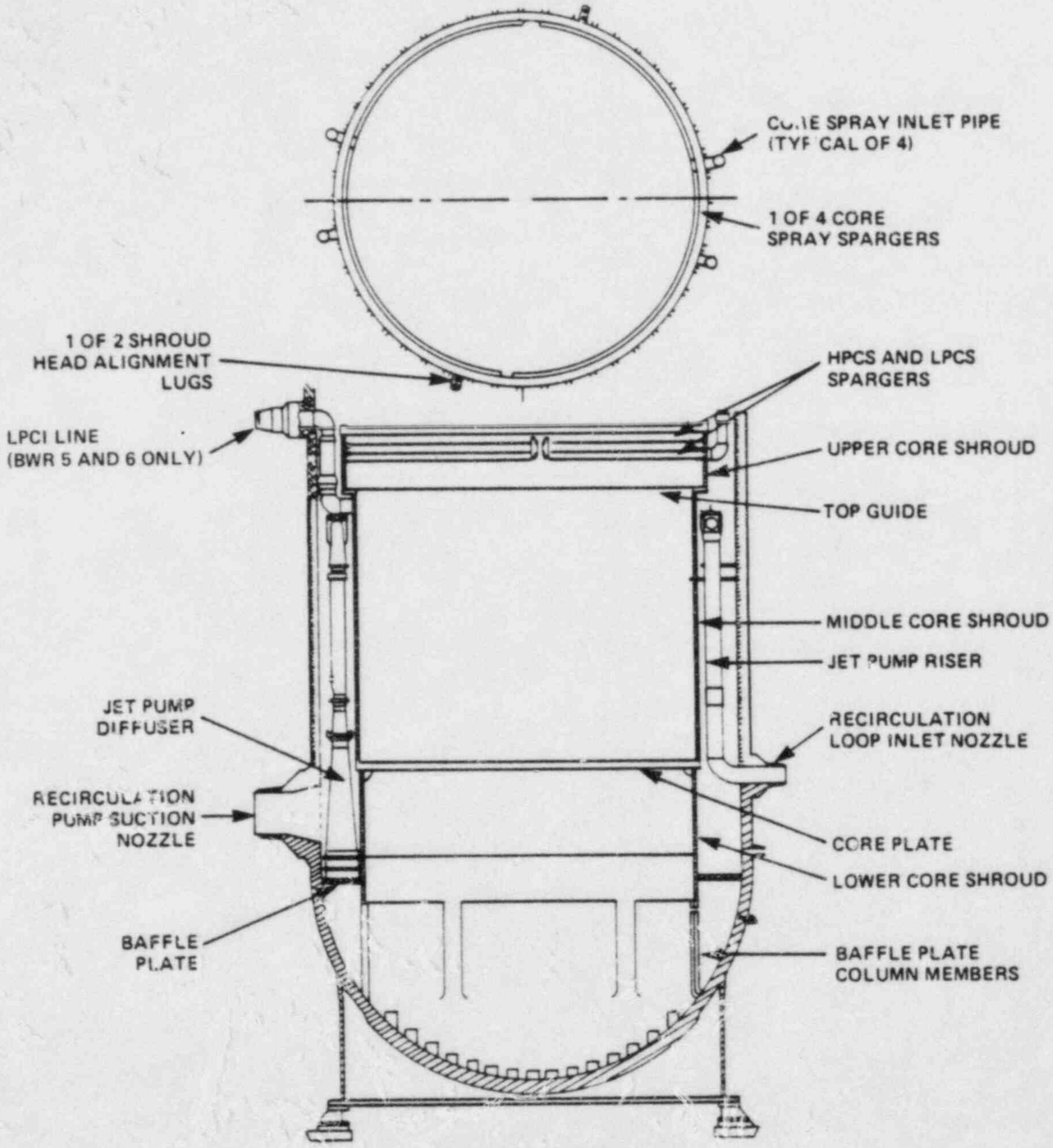


Fig. 2.9. BWR core shroud construction.

ORNL-DWG 82-5179 ETD

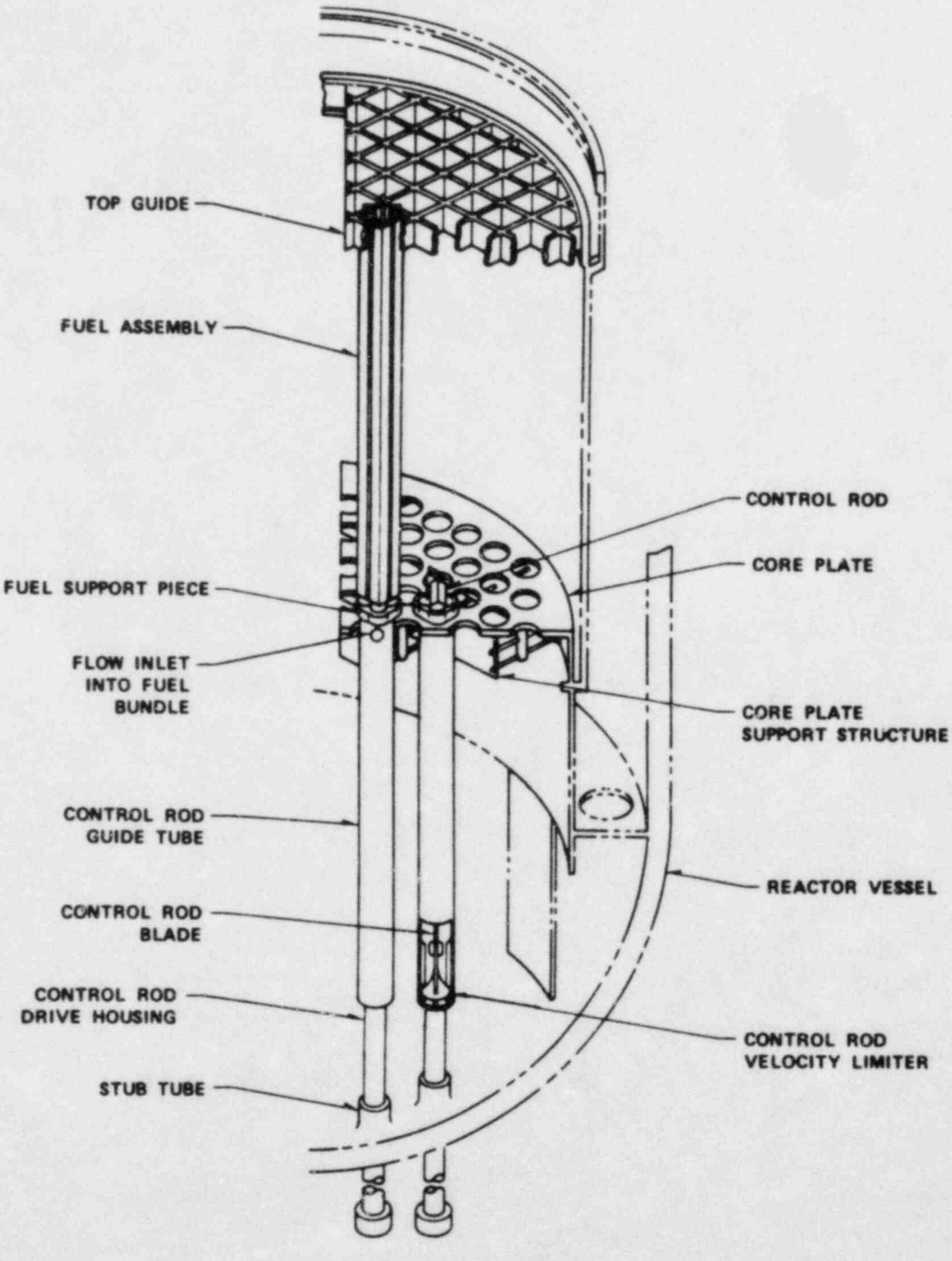


Fig. 2.10. BWR core support structures.

ORNL-DWG 83-5600 ETD

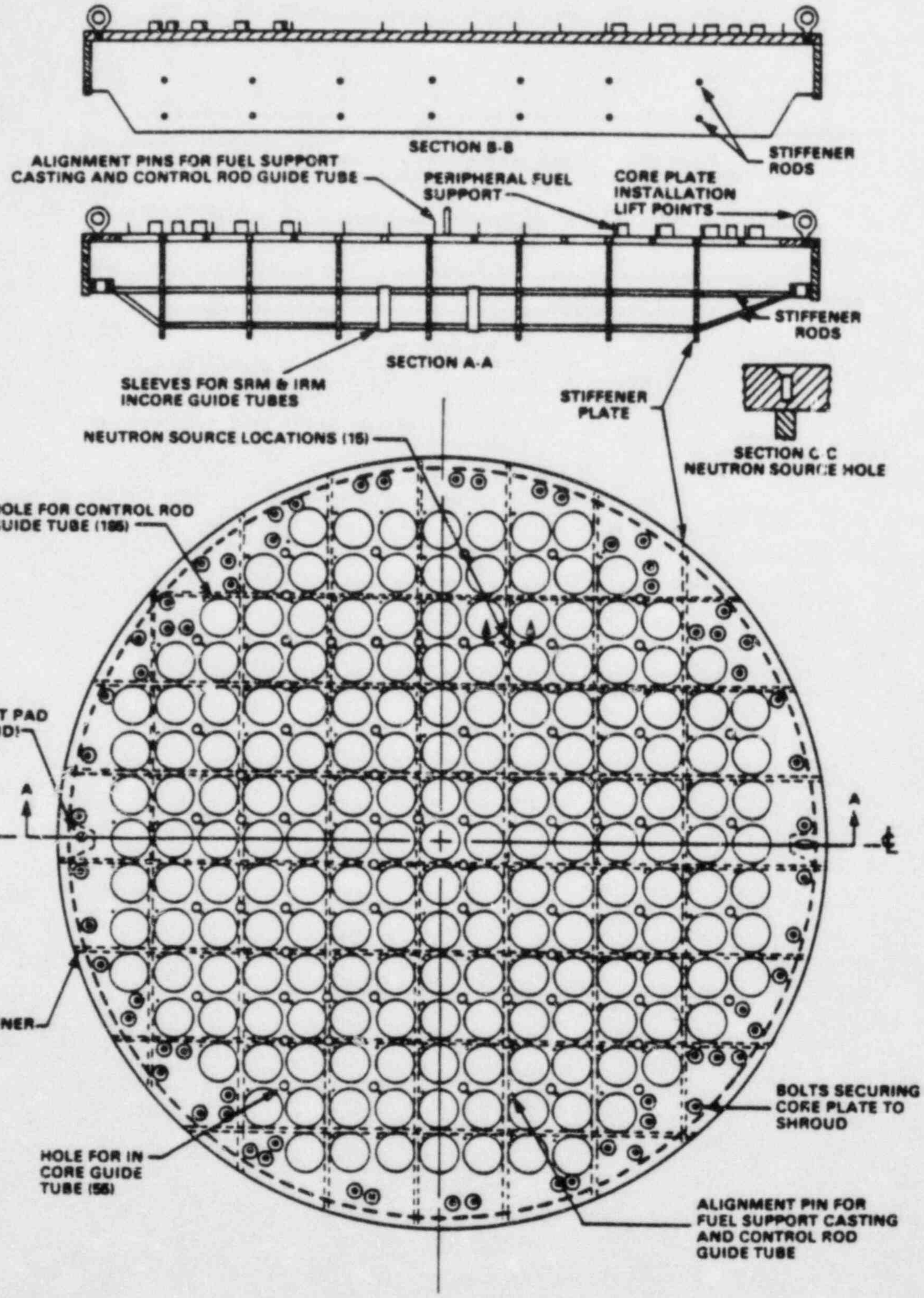


Fig. 2.11. BWR core plate construction.

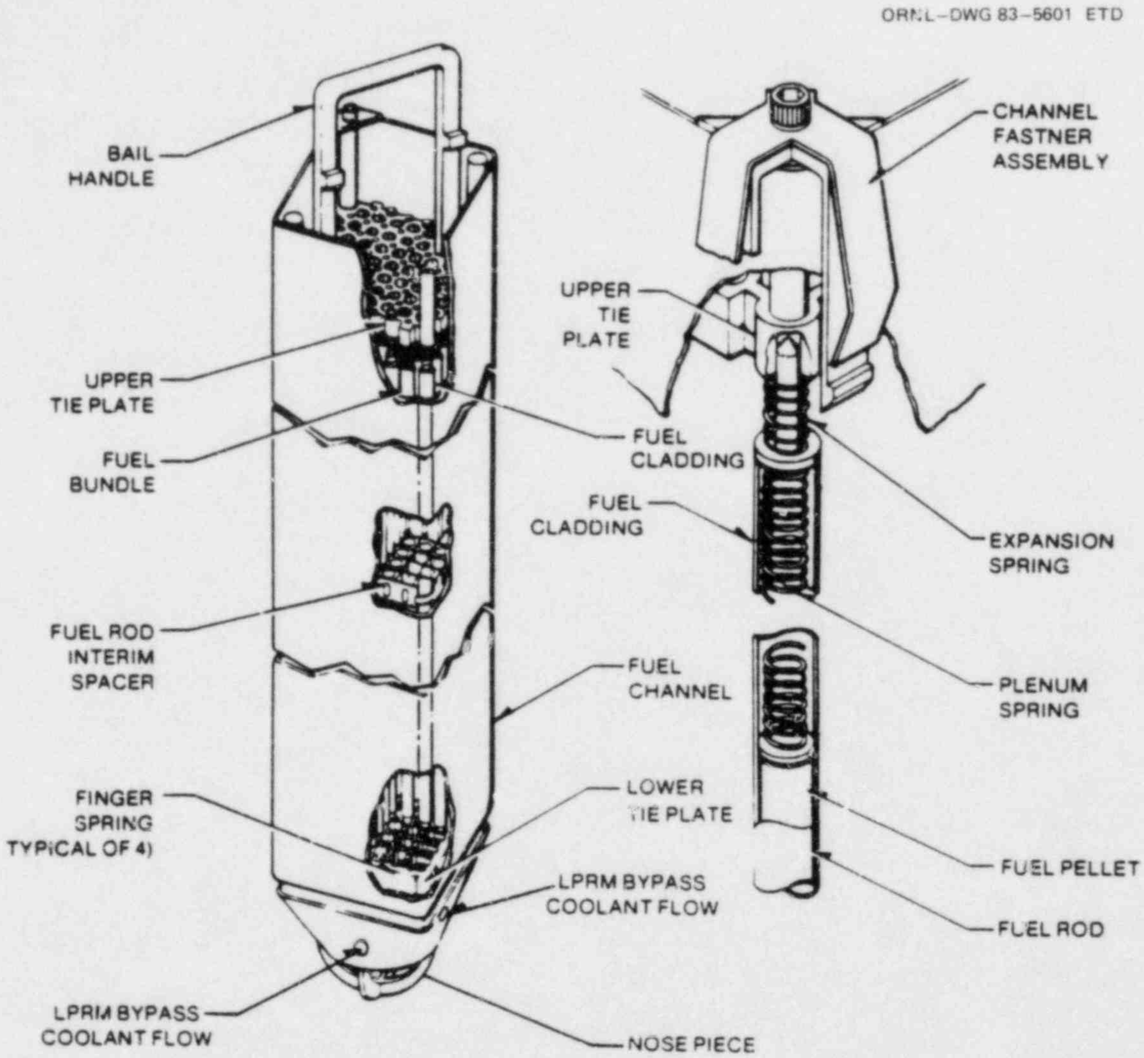
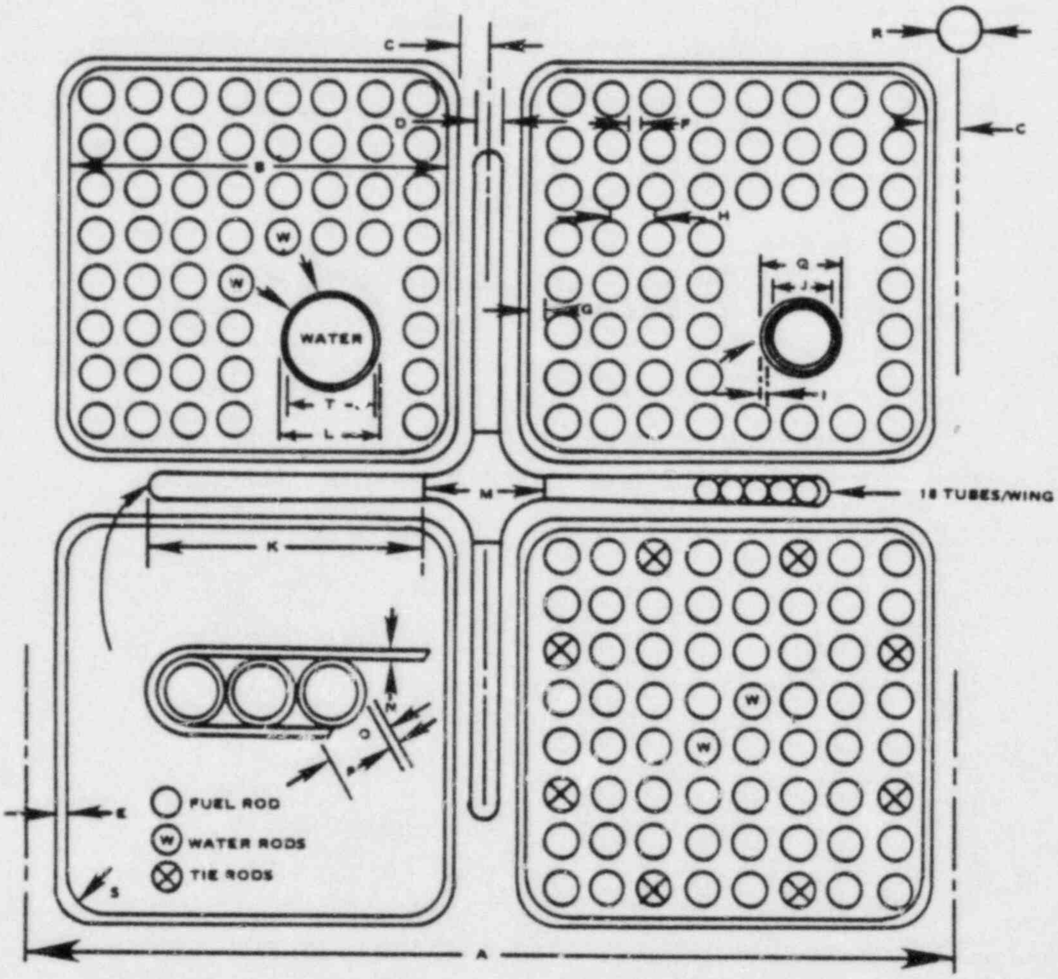


Fig. 2.12. BWR fuel assembly.

ORNL-DWG 83-5602 ETD



DIM. IDENT	A	B	C	D	E	F	G	H	I	J
DIM. INCHES	12.0	5.215	0.2725	0.323	0.120	0.153	0.140	0.636	0.032	0.410

DIM. IDENT	K	L	M	N	O	P	Q	R	S	T
DIM. INCHES	4.094	0.591	1.562	0.046	0.220	0.027	0.483	1.062	0.380	0.531

Fig. 2.13. Fuel cell cross section.

ORNL-DWG 83-5603 ETD

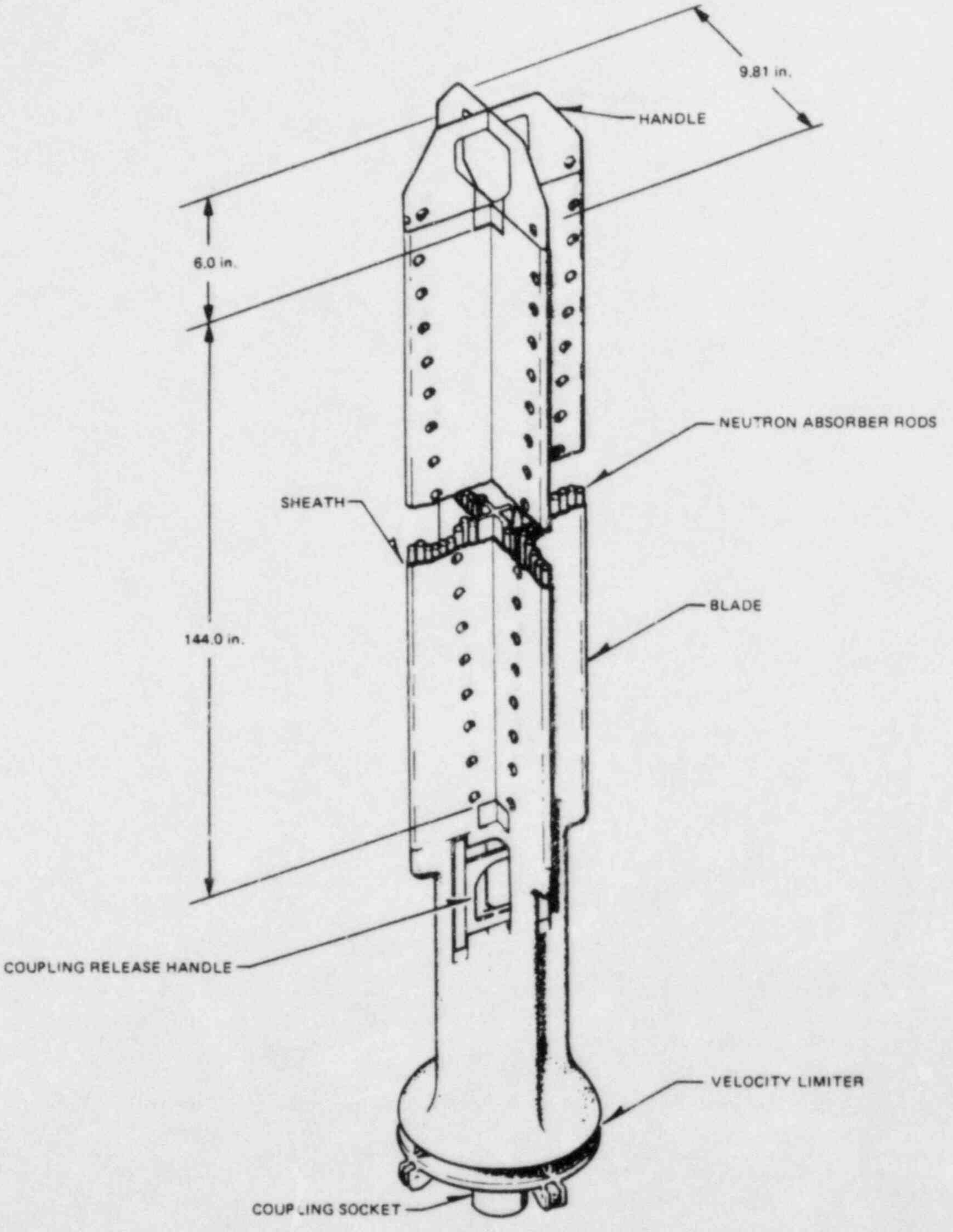


Fig. 2.14. BWR control rod assembly.

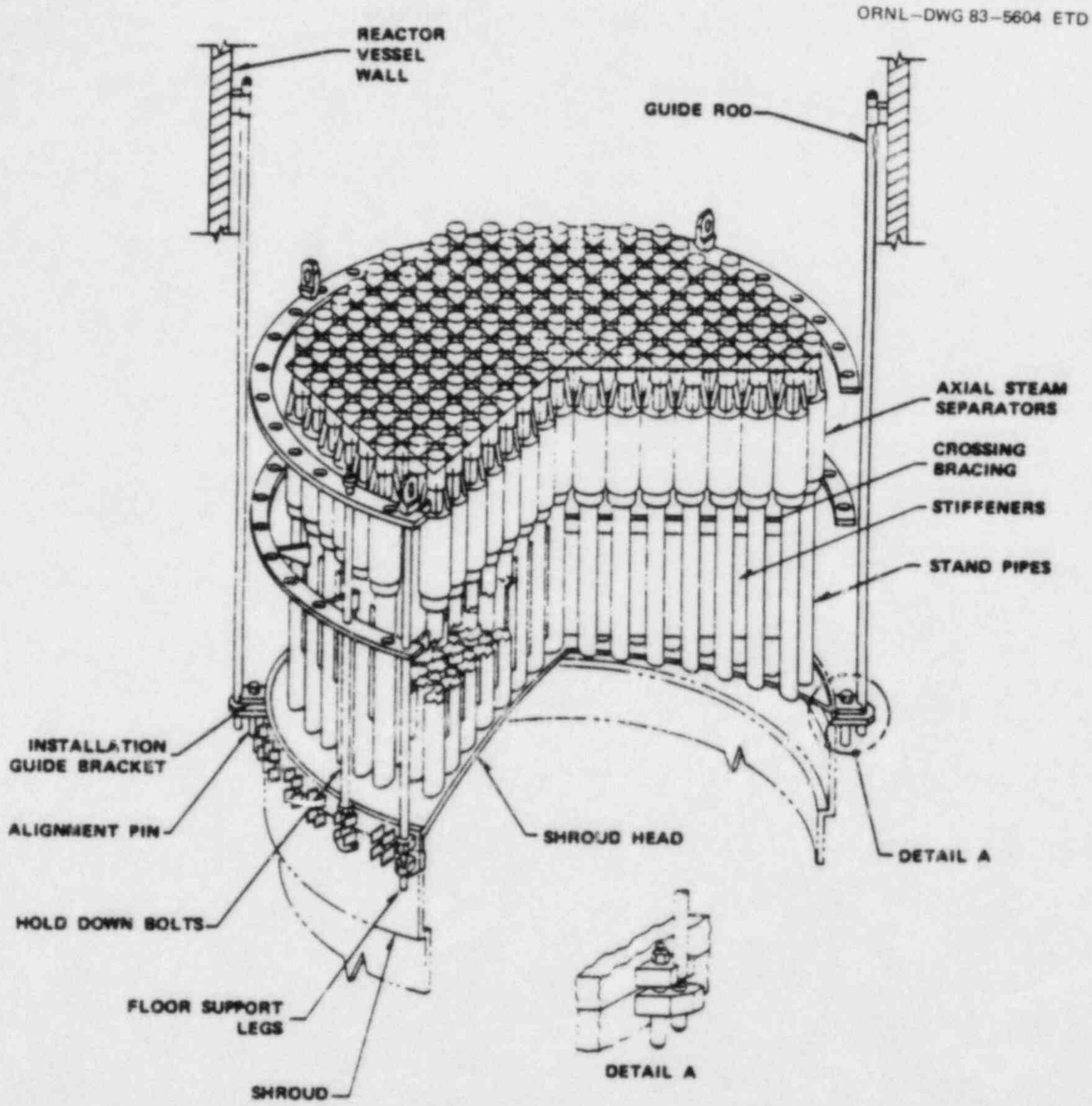


Fig. 2.15. Shroud head and steam separator assembly.

ORNL-DWG 83-5605 ETD

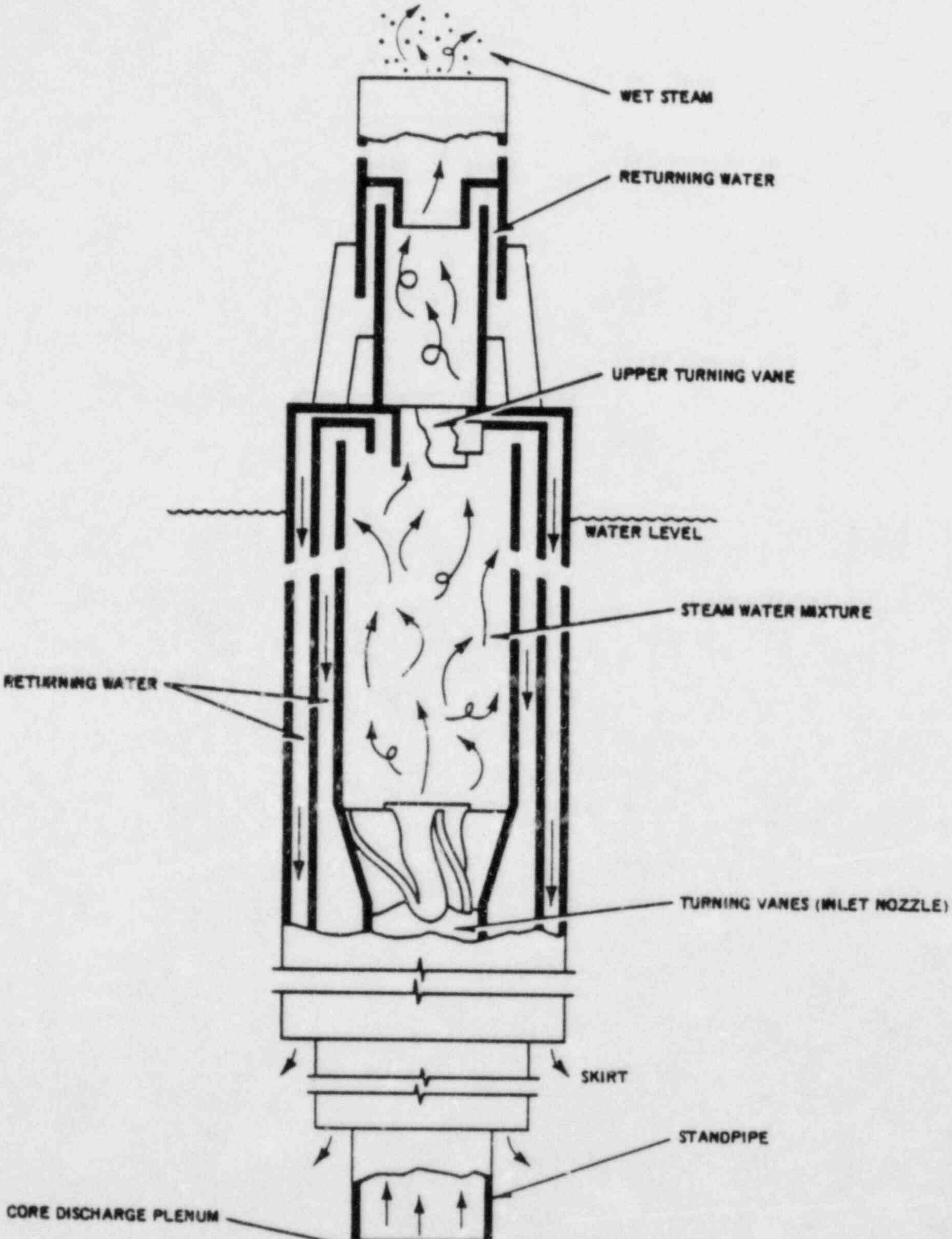


Fig. 2.16. Steam separator flow diagram.

ORNL-DWG 83-5606 ETD

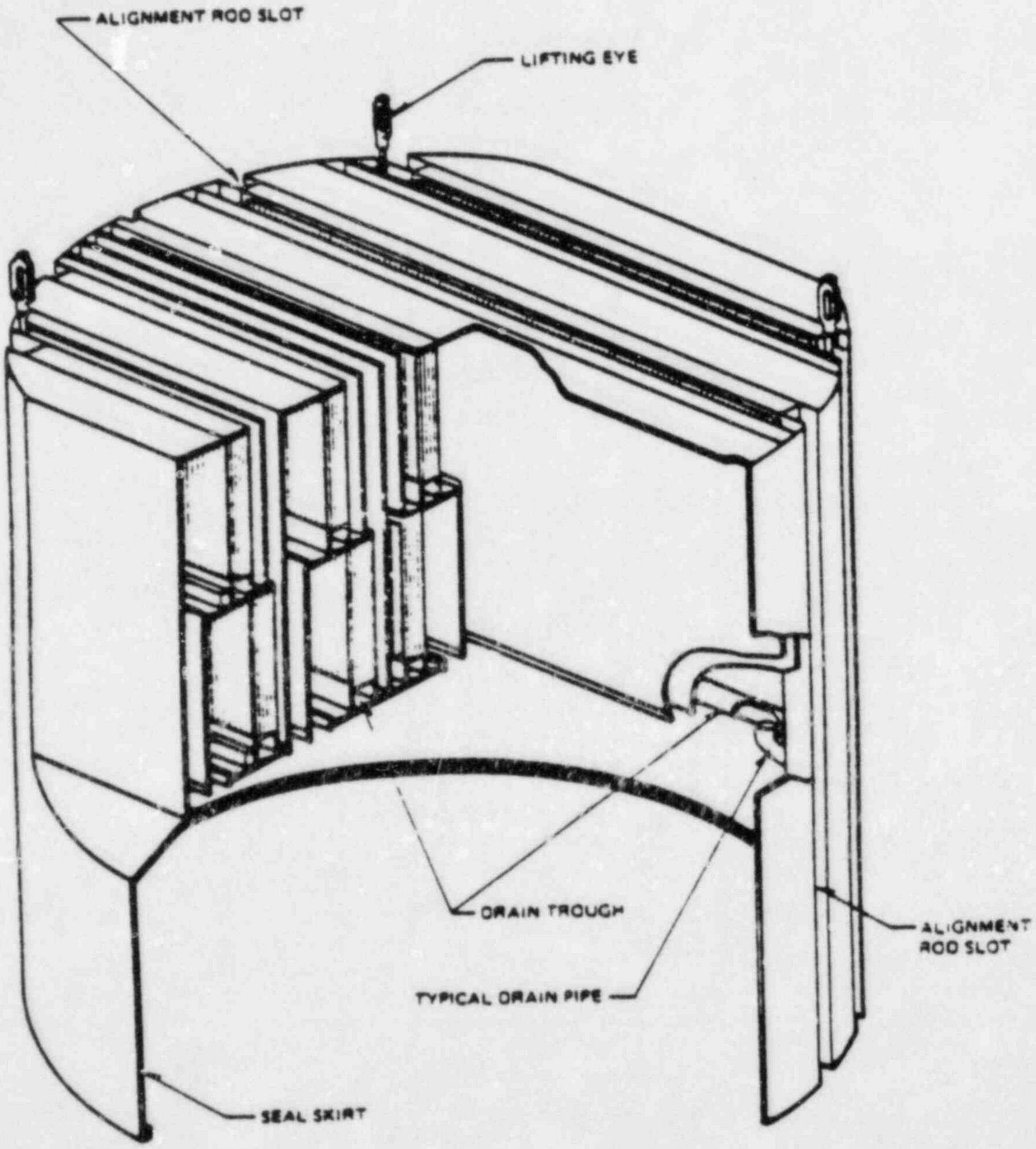


Fig. 2.17. Steam dryer assembly.

ORNL-DWG 83-5607 ETD

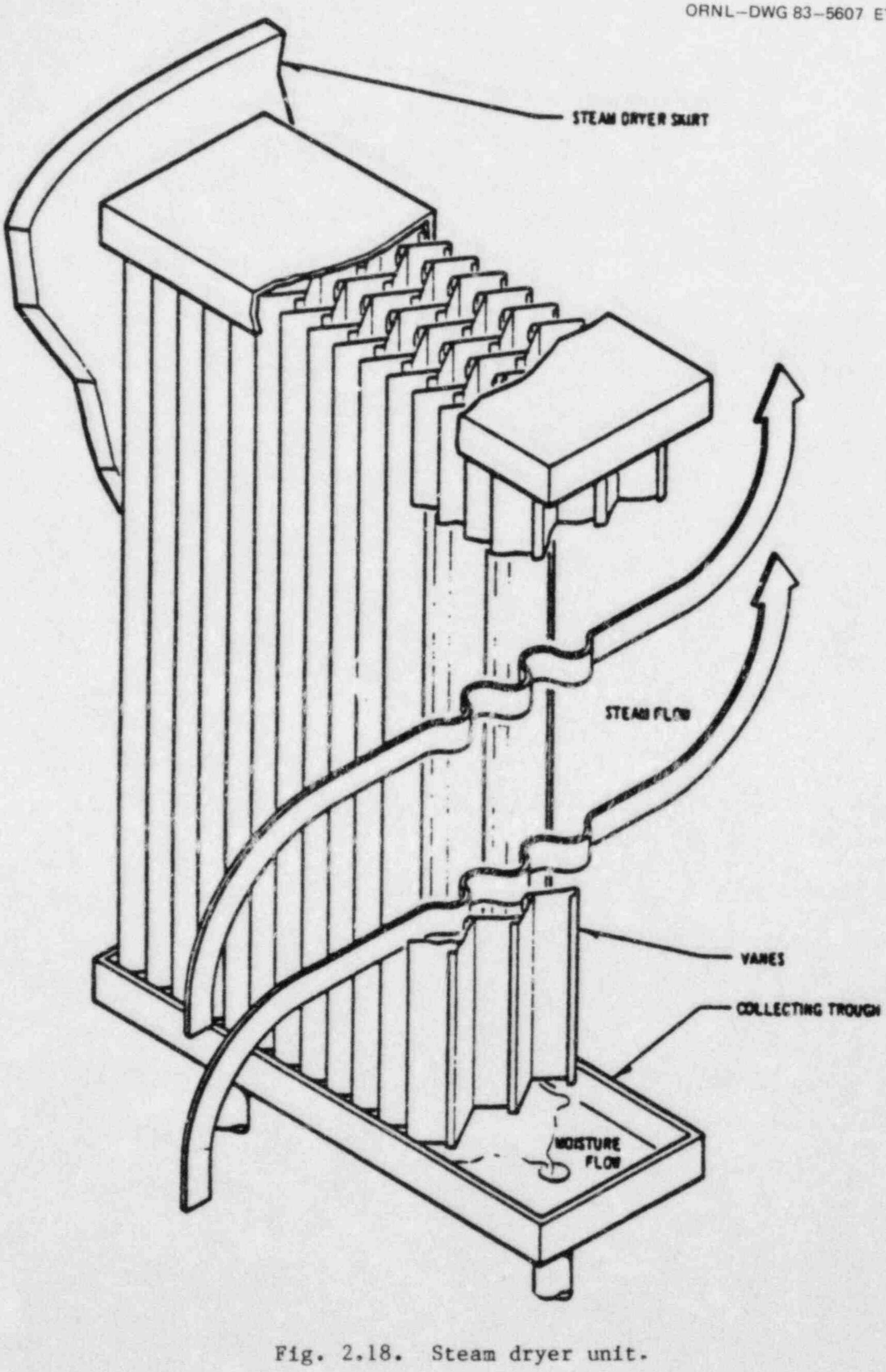


Fig. 2.18. Steam dryer unit.

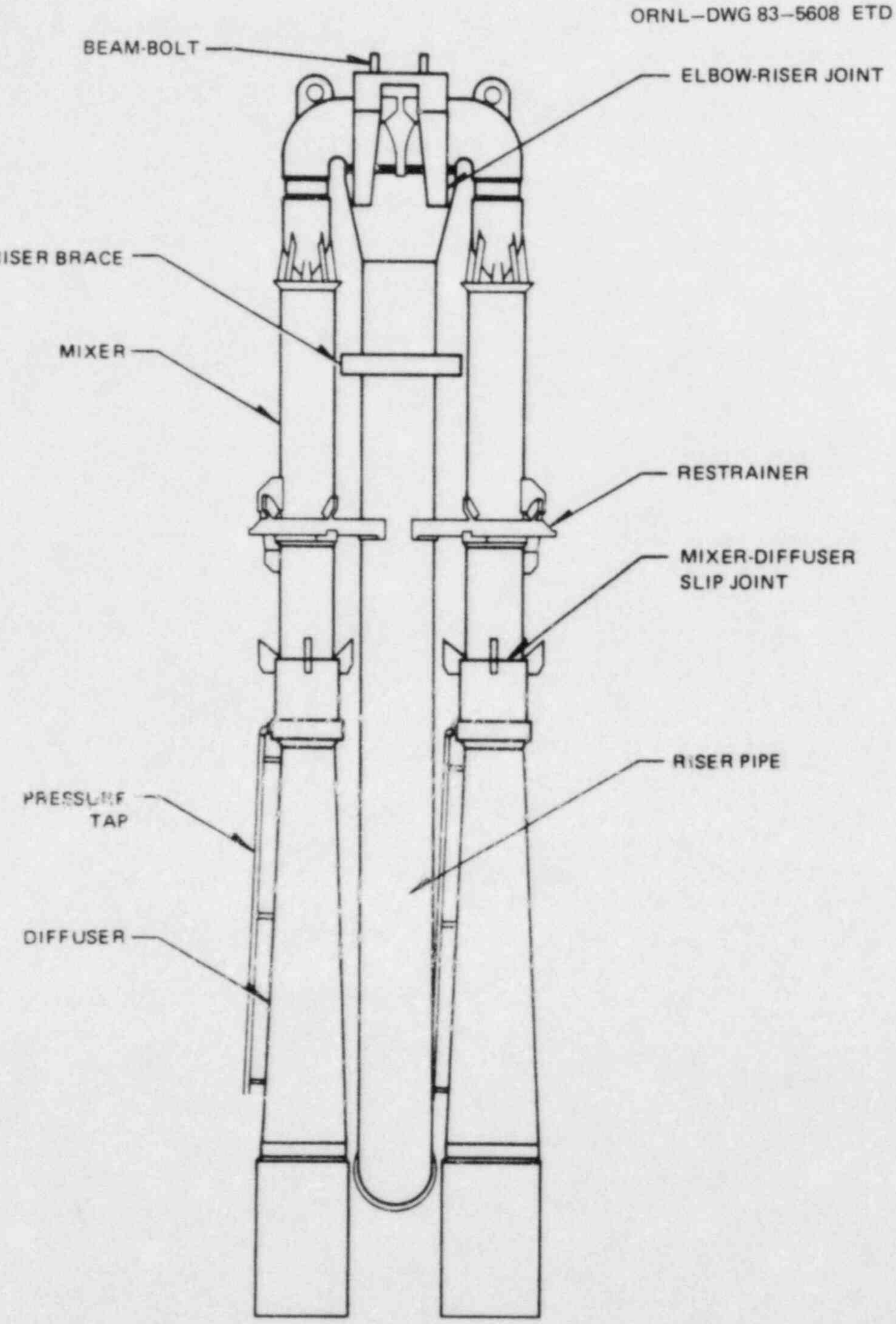


Fig. 2.19. Complete BWR jet pump assembly.

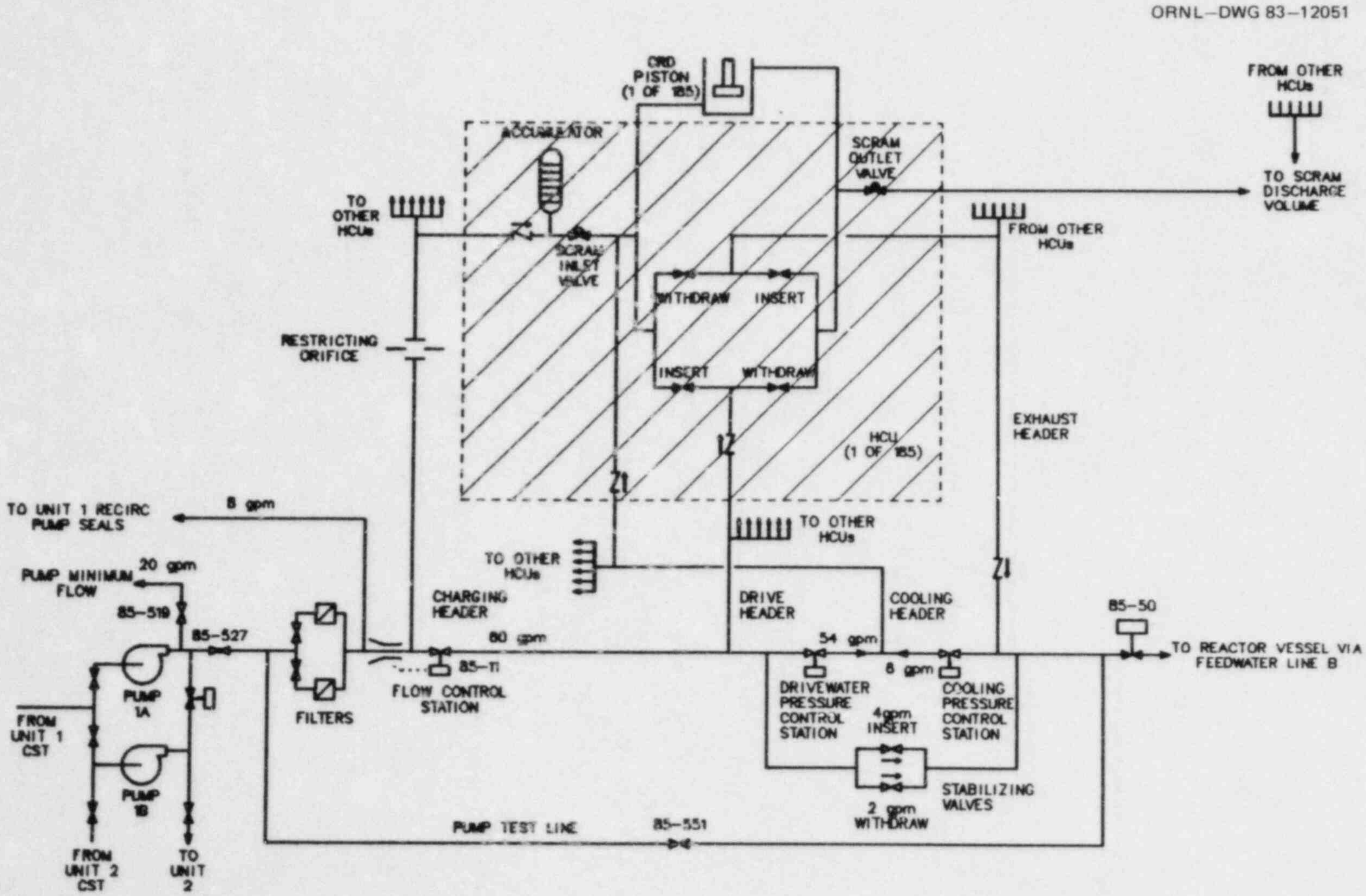


Fig. 2.20. Control rod drive hdraulic system.

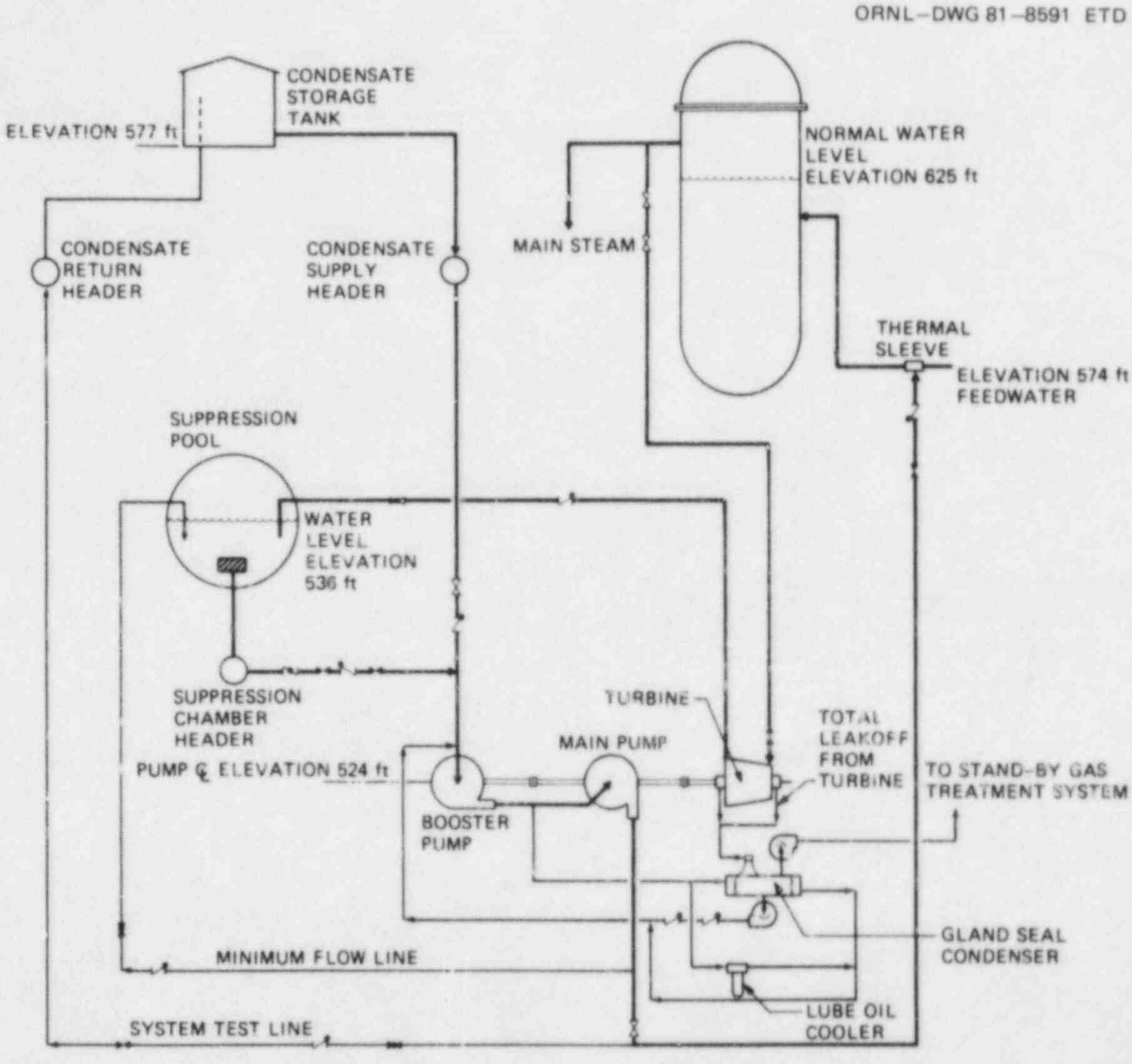


Fig. 2.21. High pressure coolant injection system.

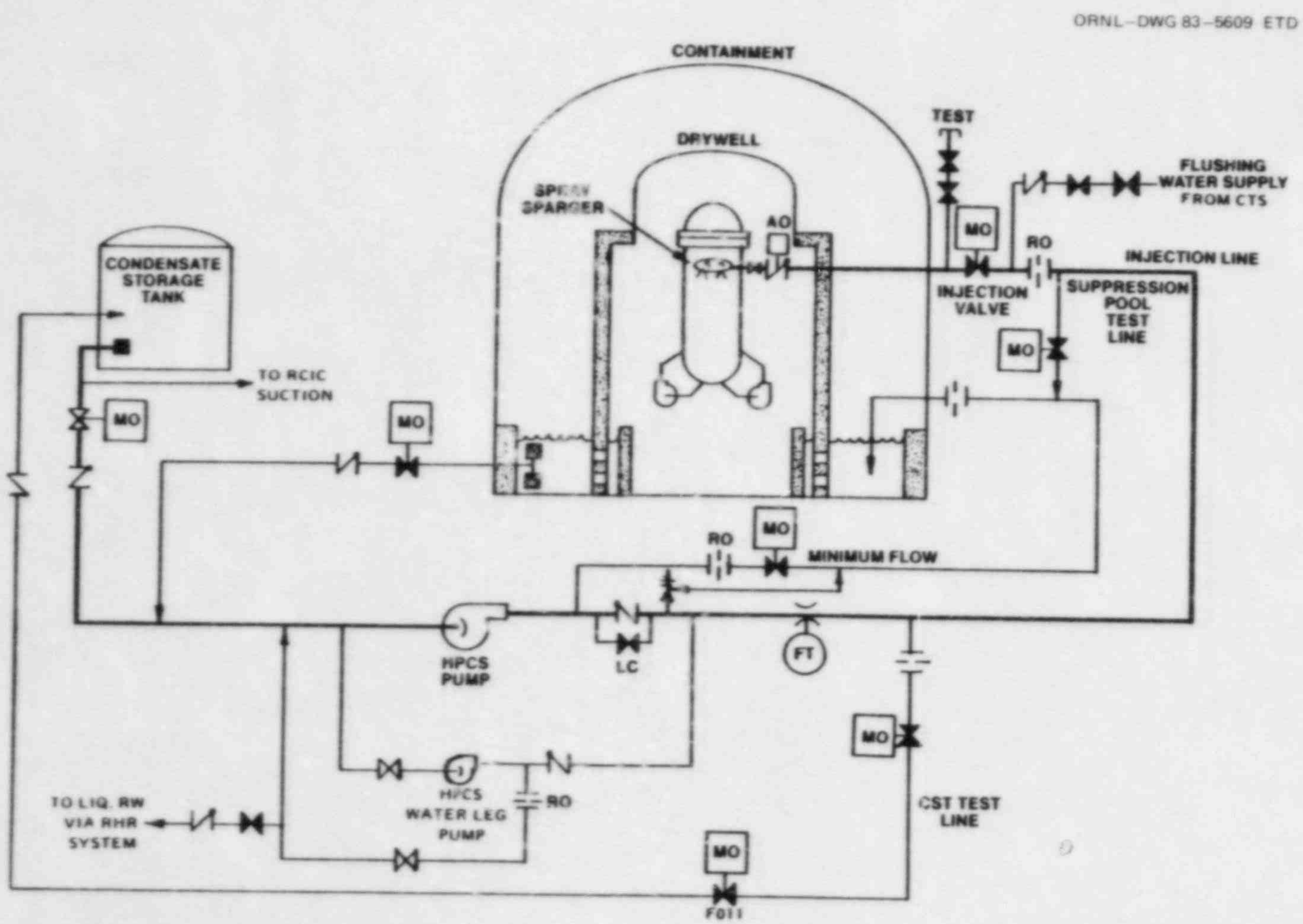


Fig. 2.22. High pressure core spray system.

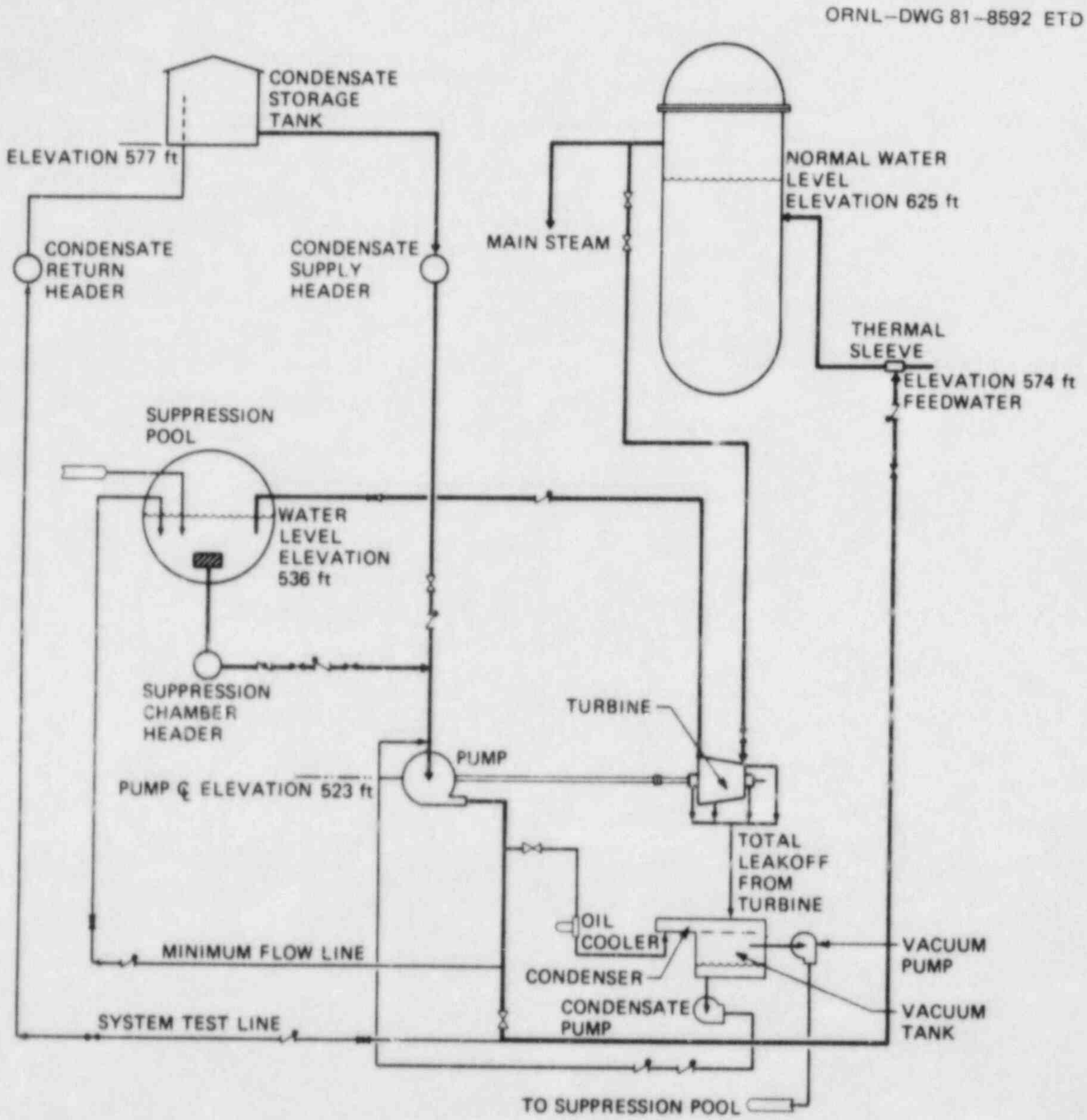


Fig. 2.23. Reactor core isolation cooling system.

ORNL-DWG 83-5630 ETD

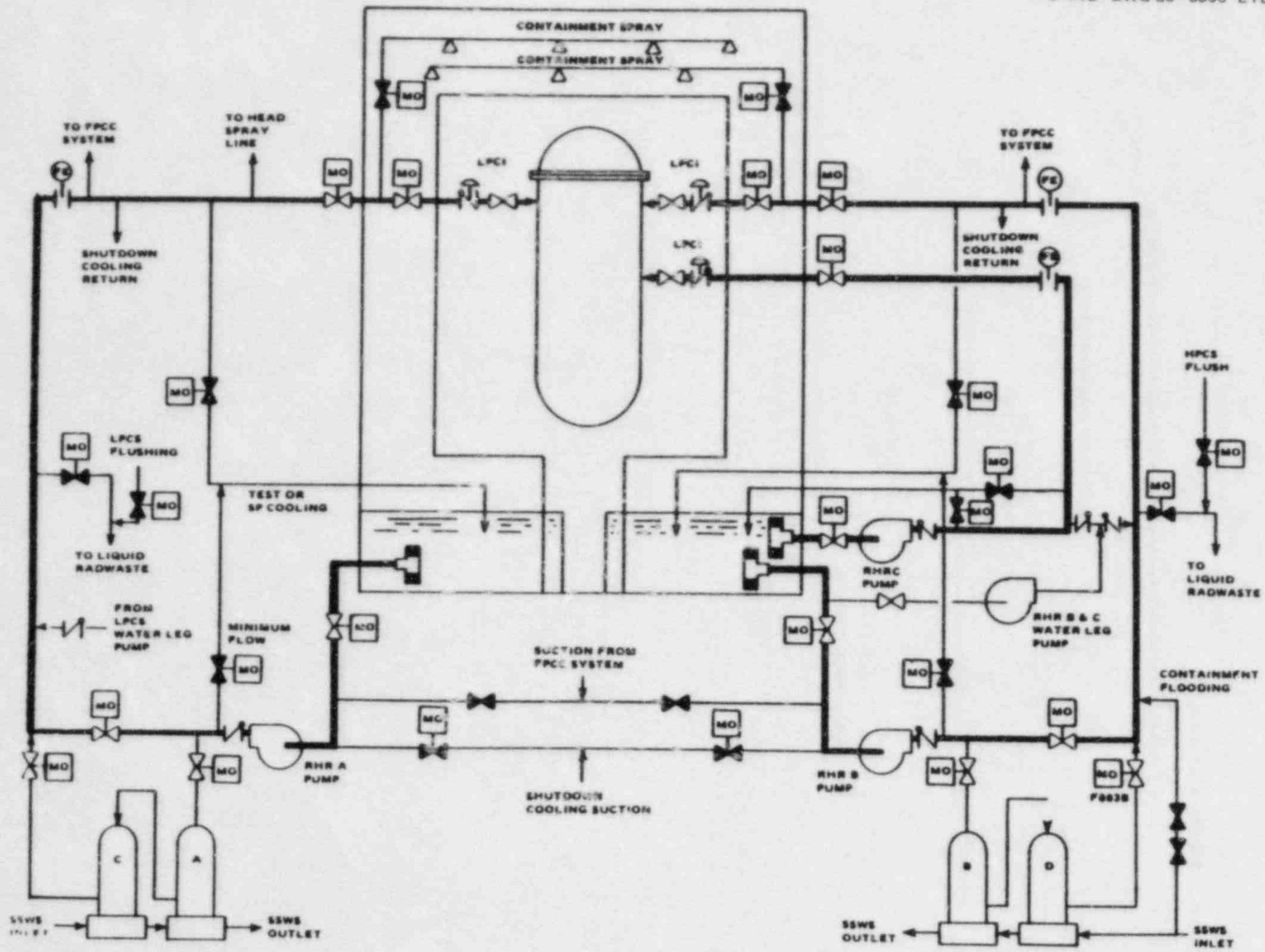


Fig. 2.24. BWR-6 residual heat removal system.

ORNL-DWG 83-5611 ETD

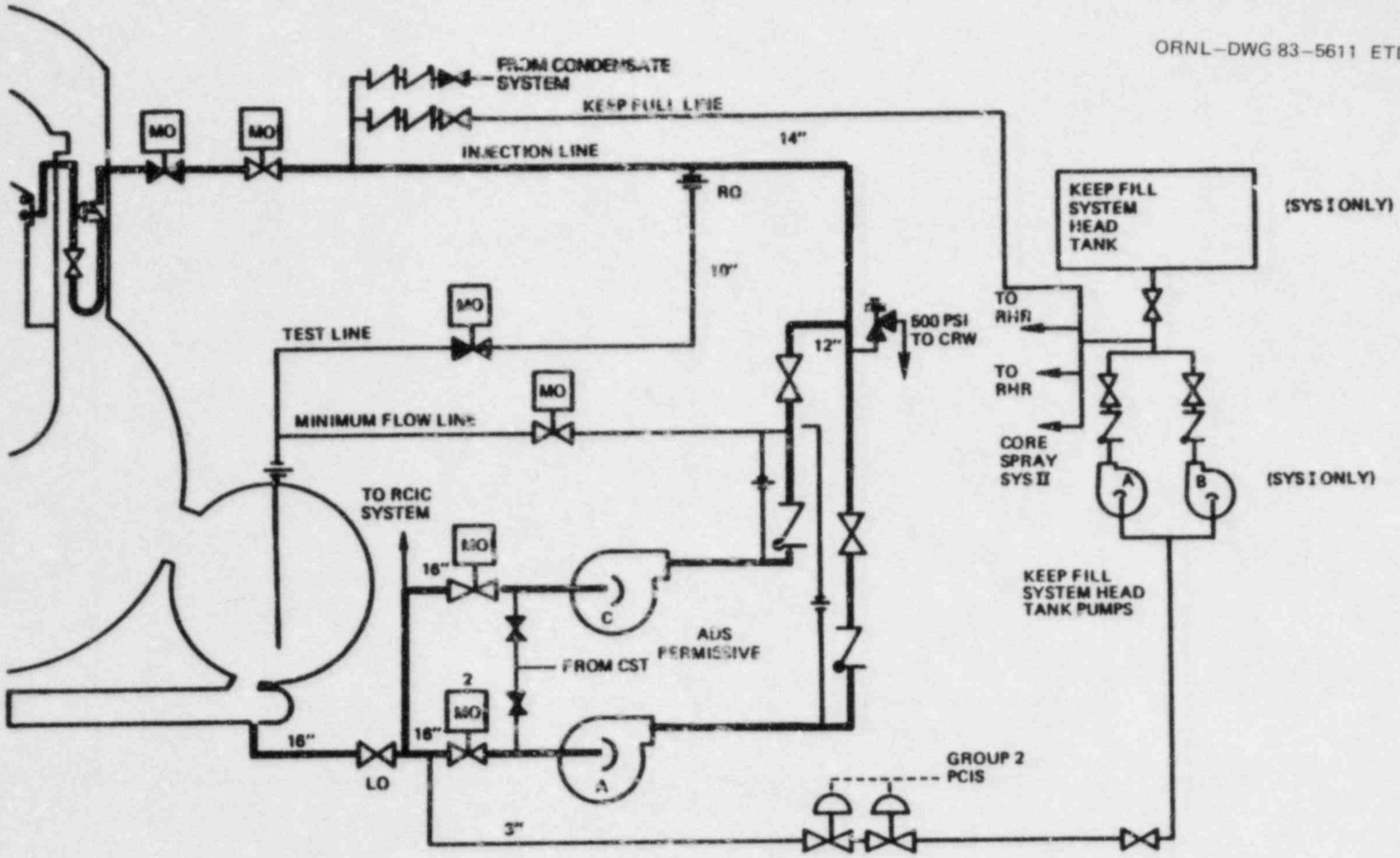


Fig. 2.25. Low pressure core spray system.

ORNL-DWG 83-5612 ETD

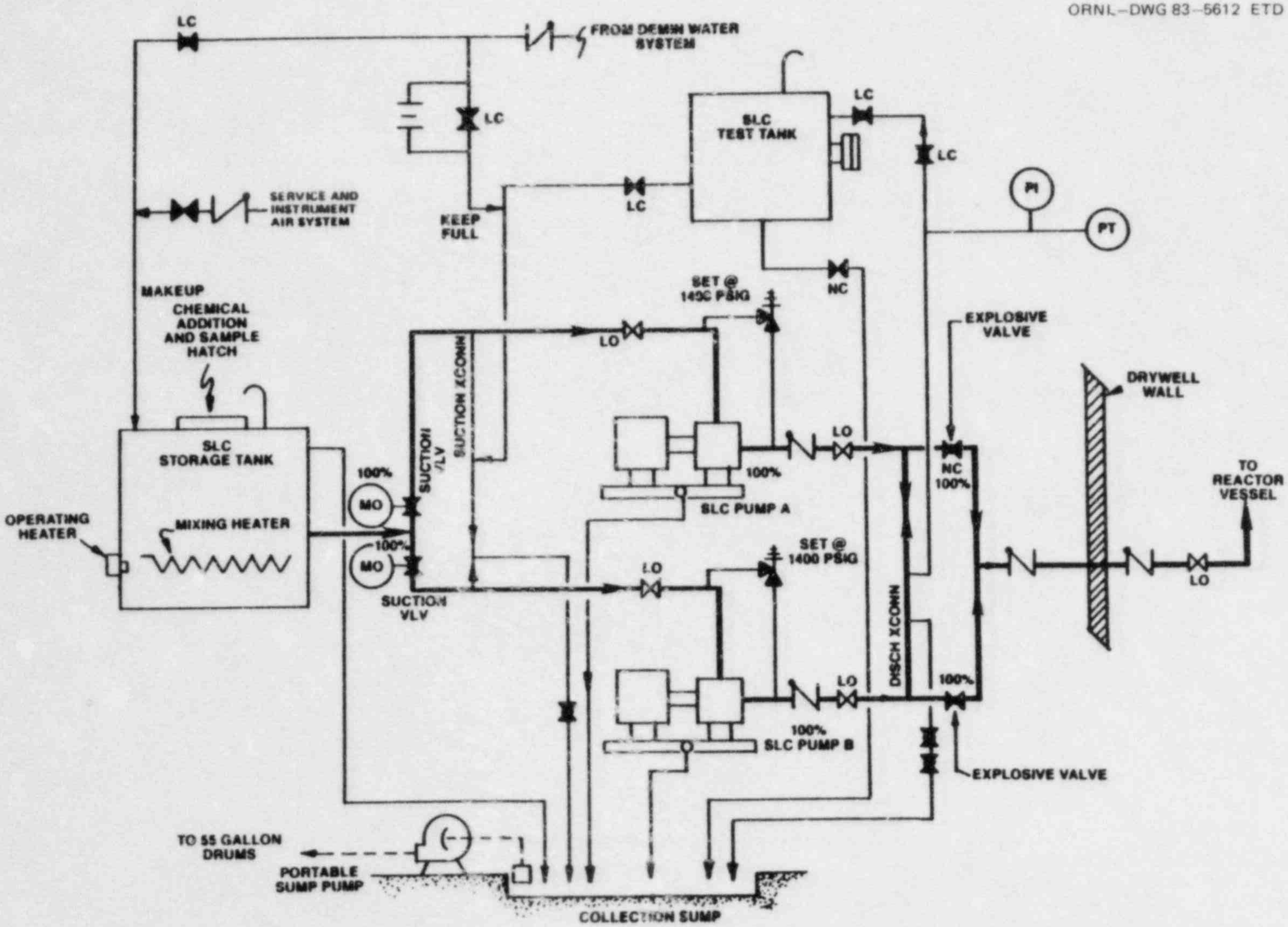


Fig. 2.26. Standby liquid control system.

ORNL-DWG 83-5613 ETD

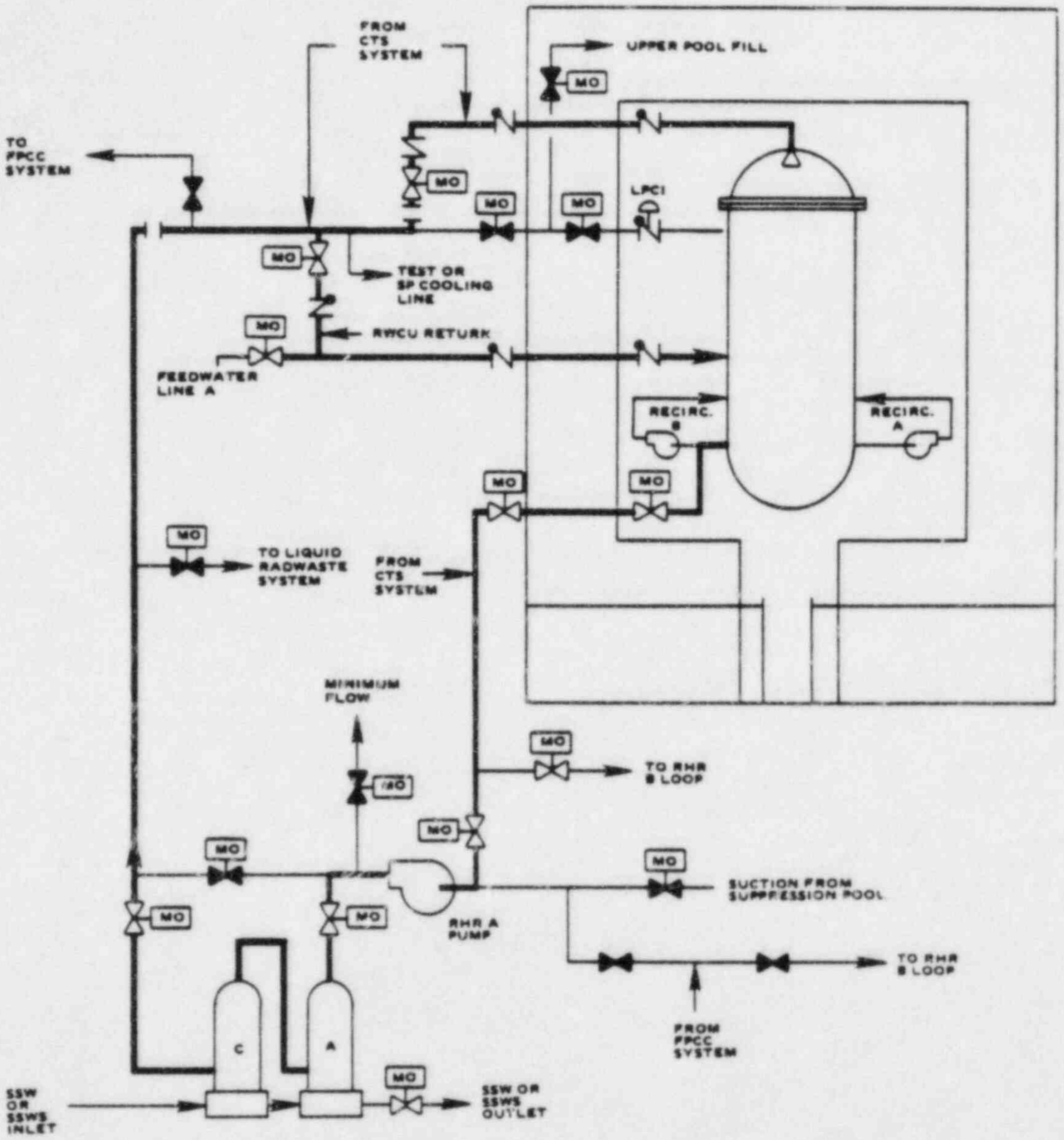


Fig. 2.27. BWR-5, 6 RHR system shutdown cooling mode.

ORNL-DWG 83-5614 ETD

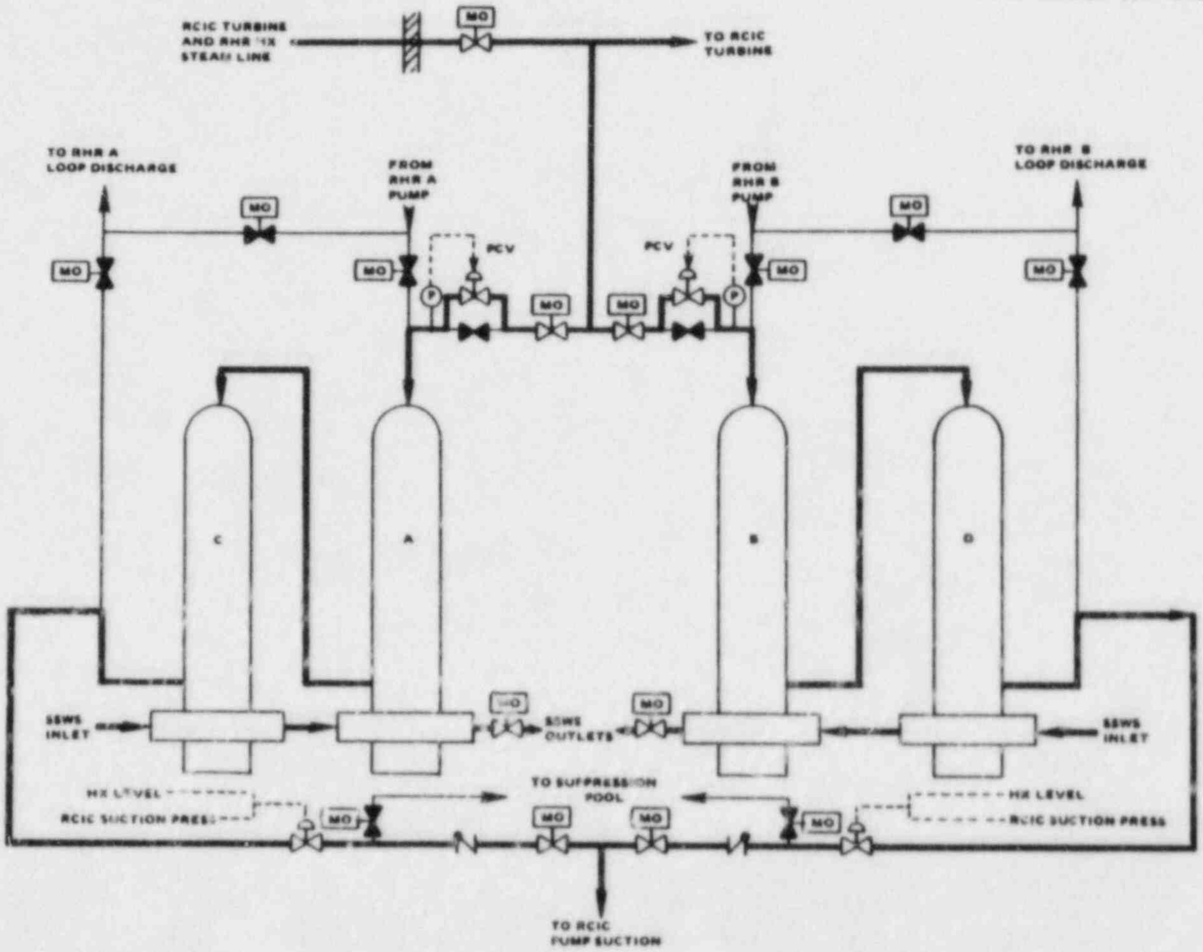


Fig. 2.28. RHR system steam condensing mode.

ORNL-DWG 83-5615 ETD

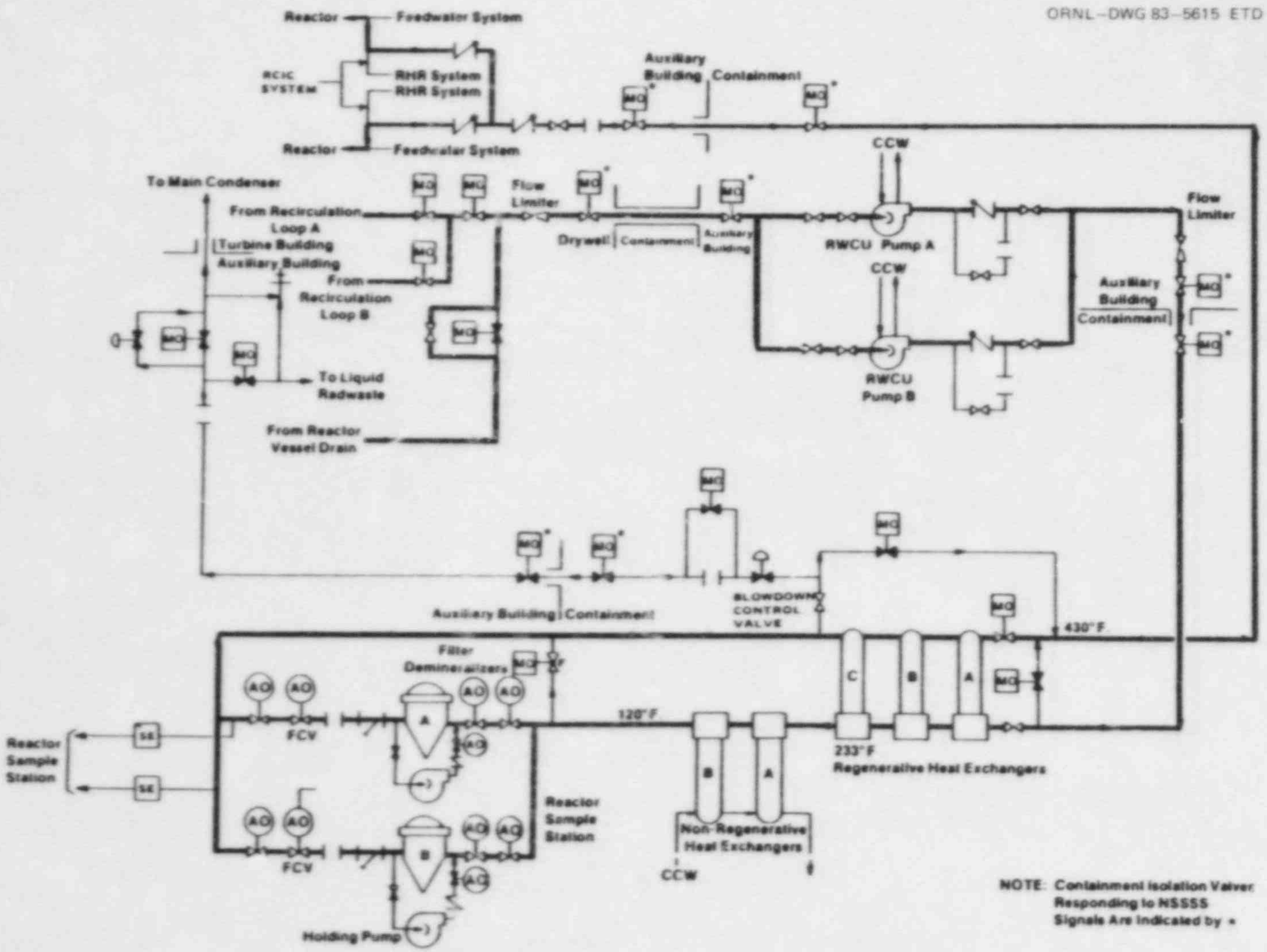


Fig. 2.29. Reactor water cleanup system.

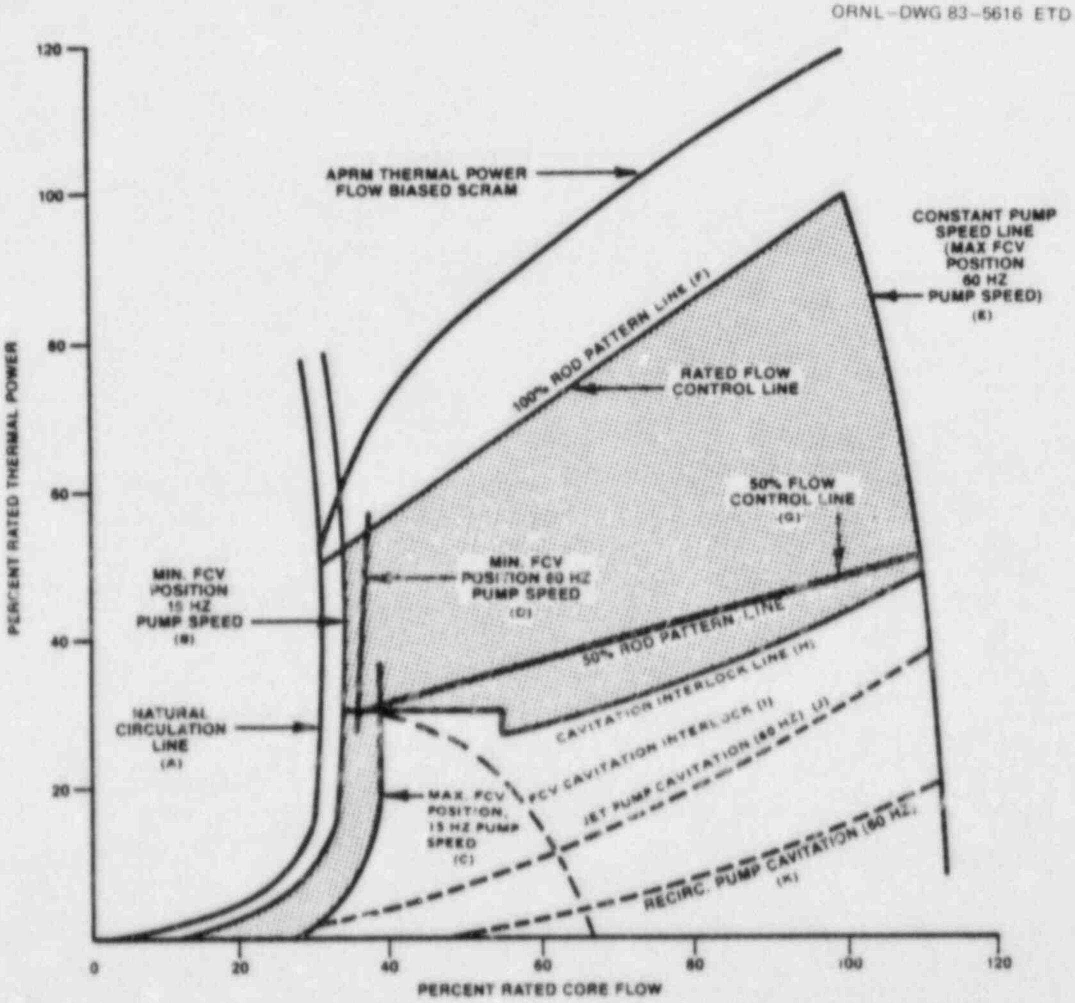


Fig. 2.30. BWR power/flow map.

ORNL-DWG 83-12787

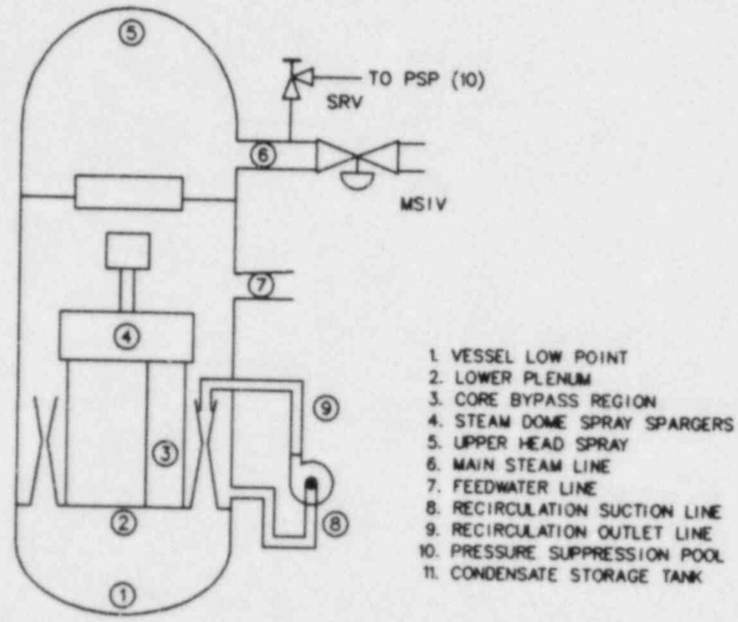


Fig. 2.31. Generic BWR primary systems — I.

ORNL-DWG 83-12742

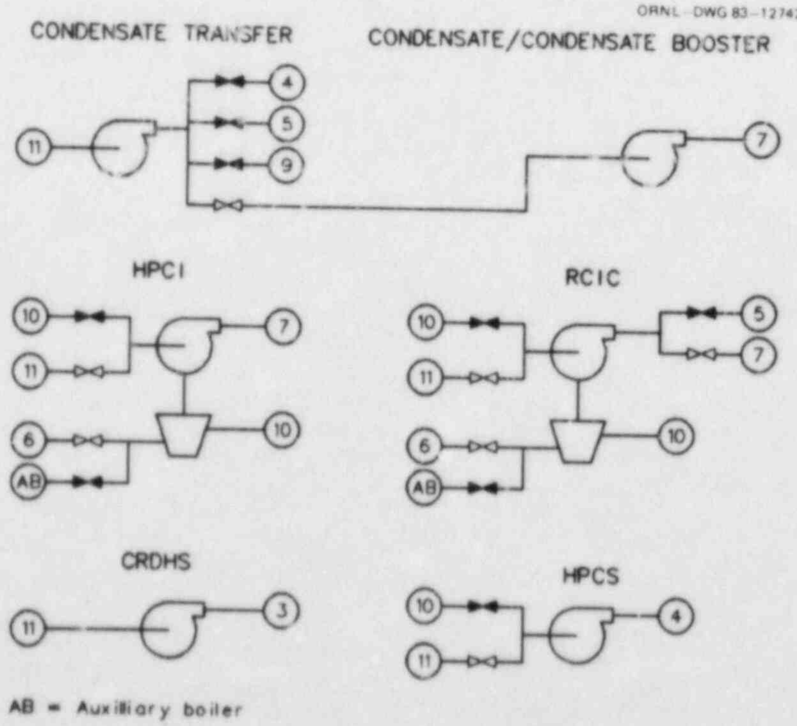


Fig. 2.32. Generic BWR primary systems — II.

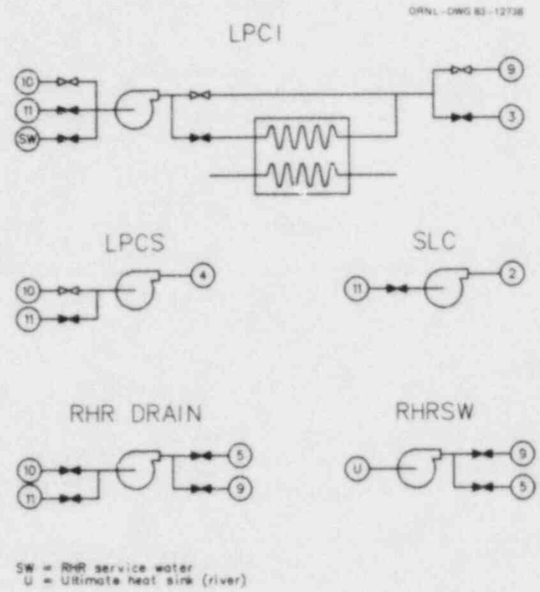


Fig. 2.33. Generic BWR primary systems — III.

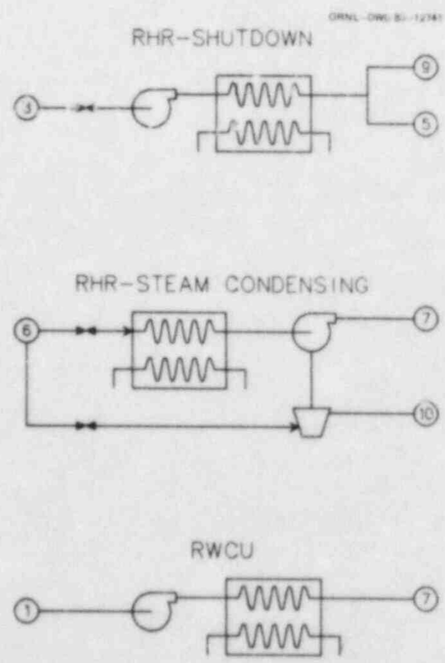


Fig. 2.34. Generic BWR primary systems — IV.

3. PLANT DESCRIPTION — CONTAINMENT

3.1 Introduction

All BWR-4, 5, and 6 plants utilize a multibarrier pressure suppression type of containment. Three distinct containment variations are employed in existing plants. The MARK I (MK I) containment design has been utilized only in conjunction with BWR-4 and earlier reactors. The MARK II (MK II) containment system is utilized with both BWR-4 and BWR-5 plants. The MARK III (MK III) containment design is the current General Electric product line, and is utilized only in conjunction with BWR-6 reactors.

For all three containment designs, the primary containment consists of a drywell, which encloses the reactor vessel and recirculation system, a pressure suppression chamber, which stores a large volume of water, and a connecting vent system between the drywell and the suppression chamber. The suppression chamber completely encloses the drywell in the MK III design.

The secondary containment in the MK I and II designs is the reactor building. The MK III design employs a secondary containment consisting of a shield building, auxiliary building, and fuel building. The alternative (Grand Gulf) MK III secondary containment design incorporates a reactor enclosure and auxiliary building.

The remaining sections of this chapter will describe the structural design of these four BWR containment variations and those interacting systems which would be of importance should a severe accident occur in one of these facilities.

3.2 MARK I Containment Structural Design

Most BWR-4 plants employ a MARK I pressure suppression primary containment system which houses the reactor vessel and coolant recirculation loops. The design consists of a drywell, constructed in the shape of an inverted light bulb, a toroidal pressure suppression chamber, which normally contains ~3785 m^3 (one million gallons) of water, and a connecting vent system between the drywell and the pressure suppression pool (Fig. 3.1).[3.1] Pertinent primary containment design parameters are given in Table 3.1. *Most of the dimensional information cited in this section is specific to the Browns Ferry nuclear plant. The reader is cautioned that many of these design parameters will vary from plant to plant.*

The drywell is a steel pressure vessel with a spherical lower portion 19.8 m (65 ft) in diameter and a cylindrical upper portion 11.7 m (38 ft, 6 in.) in diameter. The overall height of the drywell is ~35 m (115 ft). The drywell is designed for an internal pressure of 0.478 MPa (56 psig) coincident with a temperature of 411.5 K (281°F), plus the dead, live, and seismic loads imposed on the shell. The thickness of the drywell wall varies from a minimum of 1.9 cm (3/4 in.) in the cylindrical section, to a maximum of 5.9 cm (2-5/16 in.) in the toroidal sphere/cylinder knuckle region.

The entire weight of the reactor is supported by a reactor vessel support assembly (Fig. 3.2) which consists of a ring girder, sole plate, and the assorted hardware necessary to position and transfer the weight of the reactor to the support pedestal. The concrete and steel support pedestal is constructed integrally with the reactor building foundation. The reactor pedestal is ~25 ft tall, with a maximum wall thickness of 5 ft and a minimum wall thickness of ~3 ft. The pedestal has one or two major doorway openings on opposite sides which extend down to the drywell floor.

The pressure suppression chamber is a steel pressure vessel of toroidal shape, located below and surrounding the drywell. The centerline diameter of the torus is ~33.8 m (111 ft) and the cross-sectional diameter is 9.5 m (31 ft). The torus contains ~3823 m (135,000 cubic ft) of water at maximum pool level. The thickness of the torus wall varies between 1.9 and 2.9 cm (3/4 and 1-1/8 in.). The suppression chamber is designed to the same material and code requirements as the steel drywell vessel, and all attachments to the torus are by full penetration welds.

The drywell and suppression chamber are connected by a vent system which, under accident conditions, conducts flow from the drywell into the suppression pool and distributes this flow uniformly around the pool. Eight circular vent pipes, each 2.06 m (6.75 ft) in diameter, connect the drywell to the suppression chamber. Jet deflectors are provided in the drywell at the entrance to each vent pipe. These vents are connected to a 1.45-m (4-ft, 9-in.) diameter vent header of toroidal shape, which is contained within the airspace of the suppression chamber. Ninety-six downcomer pipes, each 0.61-m (24-in.) diameter, project downward into the suppression pool, terminating 1.22 m (4 ft) below the surface of the pool. Vacuum breakers discharge from the suppression chamber atmosphere into the vent pipes to prevent the suppression pool pressure from exceeding the drywell pressure by more than 0.5 psi. The suppression chamber, which is located in a separate room in the reactor building basement (Fig. 3.3), is accessible only through two normally closed 1.22-m (4-ft) diameter manhole entrances with double testable seals and bolted covers.

In addition to serving as a heat sink for drywell blowdown following a loss-of-coolant accident, the suppression pool serves as a source of water for the HPCI, HPCS, RCIC, LPCS, and RHR systems, as well as a heat sink for the SRV discharge and the HPCI and RCIC turbine exhausts.

Several types of piping and electrical penetrations, as well as personnel and equipment access hatches penetrate the primary containment. The general design of the piping penetrations incorporates a penetration sleeve which passes from the reactor building, through the shield wall concrete, and projects into the gap region between the shield wall and the drywell liner. Guard pipes and expansion bellows are incorporated where necessary to allow for movement and protection of process lines. Personnel and equipment hatches incorporate double, testable seals to ensure containment integrity.

During normal operations, the Mark I containment atmosphere is nitrogen inerted to less than 4% oxygen. Fan-forced drywell atmospheric cooling units (see Sect. 3.5.5) maintain the atmospheric temperature between 330 and 339 K (135 and 150°F) during normal operations.

Protection of the primary containment from exceeding the design maximum external pressure (2 psi) is provided by a system of self-actuating swing check vacuum relief valves. The valves will completely open within one second after a 3.4 KPa (0.5 psi) differential pressure is applied across the seat.

Two valves in series are used in each of two lines from the reactor building atmosphere to the air space above the suppression pool. The reactor building/suppression chamber valves are intended to bleed air from the reactor building into the suppression chamber and will be completely open within one second after a 0.5 psi differential pressure is applied across the seat.

Drywell/suppression chamber vacuum breakers are remotely testable using air cylinder actuators, while reactor building/suppression chamber vacuum breakers are manually testable using an accessible lever arm.

The secondary containment or reactor building completely encloses the drywell and suppression chamber which make up the primary containment. The purpose of the secondary containment is to minimize the ground-level release of airborne radioactive materials and provide for the controlled and elevated release of the building atmosphere via the Standby Gas Treatment System under accident conditions. When the primary containment is open, such as during refueling and maintenance operations, the secondary containment serves as the primary containment.

In addition to the primary containment, the reactor building houses the refueling and reactor service areas, the new and spent fuel storage facilities and other reactor auxiliary and service equipment, including the Reactor Core Isolation Cooling System, Reactor Water Cleanup System, Standby Liquid Control System, Control Rod Drive Hydraulic System equipment, the emergency core cooling systems and electrical components.

The normal ventilation system provides filtered air to the reactor building and then exhausts it through an elevated release. The ventilation system maintains the reactor building at a 0.25 inch water negative internal pressure, thereby ensuring inleakage.

The reactor building substructure consists of poured-in-place reinforced concrete exterior walls that extend up to the refueling floor. The refueling room floor is also made of reinforced poured-in-place concrete. The superstructure of the reactor building above the refueling floor is structural steel.

The refueling floor walls are covered with insulated metal siding. The reinforced concrete exterior walls and the structural steel for the superstructure are designed for tornado considerations and missile protection.

Excessive reactor building-to-atmosphere pressure differentials due to steamline ruptures and tornadoes are prevented by venting to the atmosphere through relief panels. Three sets of relief panels and a flow limiter prevent overpressurization of the secondary containment system. These consist of the main steam relief panels, the zone relief panels, the exterior siding panels, and the HPCI steam line flow limiter. Main steam ruptures would be vented to the turbine building through main steam relief panels. Zone relief panels vent the reactor building to the refueling floor. The exterior siding panels vent the refueling floor to the atmosphere.

All entrances and exits to and from the reactor building are through double door personnel and equipment air locks. Each pair of access doors is equipped with weather-strip type rubber construction seals and is electrically interlocked so that only one of the pair may be opened at a time.

3.3 MARK II Containment Structural Design

The MARK II Containment utilizes the 'over-under' design in its suppression pool arrangement. This type of containment is used on only a limited number of late model BWR/4 and all BWR/5 reactors. Typical MARK II Containments are illustrated in Figs. 3.4 and 3.5.

The MARK II design provides a more compact arrangement of the pressure suppression system and reactor building than does the MARK I design. The containment is constructed of prestressed or reinforced concrete with the suppression chamber located directly below the drywell in the same structure. The base foundation slab is a reinforced concrete mat 2.1 m (~7 ft) thick. The top of the base foundation slab within the containment is lined with stainless steel plate that serves as the suppression pool floor.

The drywell and suppression pool are steel lined structures constructed of either prestressed or reinforced concrete in the shape of a truncated cone and cylinder, respectively. The drywell head is bolted to a steel ring girder which is attached to the top of the concrete containment wall. The floor of the drywell serves as a pressure barrier between the drywell and suppression chamber and as a support structure for the reactor pedestal and downcomers. The drywell cone and suppression pool cylinder are ~24.4 m and 18.3 m (80 ft and 60 ft) high, respectively. The drywell floor is ~0.9 m (3 ft) thick.

The reactor pedestal wall thickness in the drywell region varies between 1.2 and 1.8 m (4 and 6 ft) thick. The reactor pedestal stands ~25.6 m (84 ft) tall from its base to the vessel support lip. The pedestal may be either solid (Fig. 3.5) or hollow (Fig. 3.4) in the suppression pool region (plant dependent). In plants that have hollow pedestals, the pedestal volume is open to the suppression pool via openings in the pedestal wall, and the region inside the pedestal is therefore filled with water. The hollow pedestal region directly beneath the vessel in the drywell is accessible through two open manways. In some plants, the drywell floor elevation inside the reactor pedestal is several feet lower than that outside the pedestal, forming a concrete cavity directly beneath the reactor vessel (Fig. 3.5). All vent openings are shielded by steel deflector plates to prevent overloading any single vent by direct flow from a pipe break.

Vacuum breakers are provided to equalize the static pressures between the suppression chamber and the drywell and provide a controlled return flow path from the suppression chamber to the drywell to assure design operation of the suppression chamber in the event of a small steam leak.

In contrast to the MK I system, no vacuum relief is provided between the inside of the primary containment and the reactor building atmosphere in the MK II containment. The concrete containment structure has the ability to accommodate subatmospheric (negative) pressures of ~34.5 KPa (5 psi) absolute. Typical MARK II Containment design specifications are listed in Table 3.2.

The reactor building completely encloses the reactor and its primary containment. The structure provides secondary containment when the primary containment is closed and in service, and primary containment when the primary containment is open, as it is during the refueling period. The reactor building houses the refueling and reactor servicing equipment, the new and spent fuel storage facilities, and other reactor auxiliary or service equipment, including the reactor core isolation cooling system, reactor water cleanup demineralizer system, standby liquid control system, control rod drive system equipment, the emergency core cooling systems, and electrical equipment components.

The reactor building exterior walls and superstructure up to the refueling floor are constructed of reinforced concrete. Above the level of the refueling floor, the building structure is fabricated of structural steel members, insulated siding, and a metal roof. Joints in the super-structure paneling are designed to assure leaktightness. Penetrations of the reactor building are designed with leakage characteristics consistent with leakage requirements of the entire building. The reactor building is designed to limit the inleakage to 100% of the reactor building free volume per day at negative 0.25 inch H_2O gauge, while operating the standby gas treatment system. The building structure above the refueling floor is also designed to contain a negative interior pressure of 0.25 inch H_2O gauge.

3.4 MARK III Containment Structural Design

3.4.1 Introduction

MARK III containment systems are employed on all BWR-6 plants. These MX III containments are the only BWR containments which are not inerted. Figures 3.6 and 3.7 are illustrations of two versions of the MK III containment concept. Figure 3.6 is the "standard" MARK III design, while Fig. 3.7 illustrates an alternative MK III configuration utilized at the Grand Gulf nuclear plant. The designs differ in that the Grand Gulf approach utilizes a reactor enclosure building as part of the secondary containment system rather than a shield building. Table 3.3 is a listing of typical Mark III primary containment design specifications.

3.4.2 Standard MK III containment design

The containment vessel is a free standing, vertical, cylindrical steel pressure vessel with an ellipsoidal head and a flat bottom steel liner plate.[3.2] The cylindrical shell has horizontal external stiffeners and is anchored 1.5 m (5 ft) into the concrete mat foundation. The containment is a seismic category I structure. The flat bottom liner plate is approximately 3/4 inch thick and is continuously supported by the concrete mat.

The containment has an inside diameter of 36.6 m (120 ft) and is 55.8 m (183 ft) in overall height with an internal volume of 33,066 m^3 (1,168,000 ft^3). It is designed to withstand an internal differential pressure of 0.1 MPa (15 psig), an external differential pressure of 5.5 KPa (0.8 psig), and an internal temperature of 358 K (185 F). The containment vessel surrounds the drywell and suppression pool and forms the primary leaktight barrier to limit fission product leakage during a LOCA. To avoid exceeding the containment design negative pressure, redundant vacuum breaker systems are provided to connect the containment volume to the annulus volume bounded by the steel containment and the shield building.

The containment vessel is free standing and receives no structural support except at the embedment in the foundation mat. Likewise the containment provides no major structure support. The containment shell has an average thickness of 4.4 cm (1 3/4 in.). Major platforms and floors within the containment are supported by the drywell. However, the containment walls do support an overhead 125 ton capacity polar crane, some attached piping such as the containment spray headers, and miscellaneous electrical connections, personnel locks, fans, ladders, and walkways.

Among the postulated loss of coolant accidents, some accidents may require flooding the containment to remove the fuel from the reactor and affect repairs. Although it is anticipated that for most accidents, defueling of the reactor would be accomplished by the normal procedures and equipment, as a contingency to cover undefined damage resulting from a LOCA, the containment can be flooded to a level 2.08 m (6.8 ft) above the top of the active fuel in the core.

The drywell (Fig. 3.8) is a cylindrical reinforced concrete structure with a removable steel head to allow vertical access to the reactor vessel for refueling or maintenance. The drywell is constructed of 1.5 m (5 ft) thick reinforced concrete walls and roof, has an inner diameter of 22.2 m (73 ft), a height of 27.7 m (91 ft), and has a volume of 7770 m^3 (274,500 cubic ft). The drywell is designed for an internal pressure of 0.21 MPa (30 psi) gauge, an external differential pressure 0.14 MPa (21 psig), and an internal temperature of 439 K (330°F).

Two reinforced concrete walls 1.2 m (4 ft) thick and 7.6 m (25 ft) high are located across the drywell top slab. These comprise the longitudinal walls of the upper containment pools and serve as the supporting structure for the operating floor and structural stiffeners for the drywell top slab.

The suppression pool, both inside and outside the drywell, is an open top, steel lined structure. Up to about 0.3 m (1 ft) above the normal suppression pool level, the carbon steel of the containment vessel is clad with stainless steel. This clad provides a maintenance free, easily decontaminated surface and eliminates the need for a protective paint coating. The water used to fill the pool is either condensate or demineralized water. The water is generally air saturated and stagnant, but retains high purity.

The suppression pool contains 3668 m^3 (129,550 ft^3) of water at the low water level. The normal pool level may vary between a depth of about 6.2 m (20.5 ft) (high level) and about 6.1 m (20 ft) (low level). This condition allows a normal vent submergence of nearly 1.8 m (6 ft) (minimum) and a minimum freeboard height of about 1.7 m (5.5 ft).

A weir wall forms the inner boundary of the suppression pool, and is located inside the drywell. The wall is built of reinforced concrete ~0.6 m (2 ft) thick and lined with steel plate on the suppression pool side. Since the weir wall forms the inside wall of the suppression pool, it confines the pool and channels the steam released by a LOCA into the suppression pool for condensation. The weir wall height is 7.6 m (25 ft).

The MARK III arrangement uses horizontal vents to conduct the steam from the drywell during a LOCA to the suppression pool. In the vertical section, the drywell wall is penetrated by a series of 70 cm (27.5 in.) diameter horizontal pipes. There are three rows of these horizontal vent pipes with their centerlines 2.3, 3.7, and 5.0 m (7.5, 12, and 16.5 ft) below the surface of the suppression pool.

Any buildup of pressure in the drywell forces the water down in the vent annulus. When the water is depressed to the level of the first row of horizontal vents, steam is vented to the suppression pool. If the pressure in the drywell is high enough, the water in the annulus is depressed further, thereby uncovering the second and third row of vents. In addition to the LOCA steam condensing function, the pool provides a heat sink for SRV and RCIC exhaust steam, and an alternative source of water for the emergency core cooling systems.

The reactor vessel support pedestal is located below the reactor vessel and the reactor shield wall. The pedestal, which is supported by a massive concrete base located on the containment base slab, supports both the reactor vessel and reactor shield wall.

The reactor vessel support pedestal is a reinforced concrete circular cylinder about 6.4 m (21 ft) high and with a constant outside diameter of 9.8 m (32 ft) and an inside diameter which varies from about 6.4 m (21 ft) at the lower part to about 5.8 m (19 ft) at the upper part. It has openings for access, control rod drive piping, and neutron monitoring instrumentation. The vessel support skirt is attached to the pedestal. Due to the recessed floor level inside the pedestal, a cavity is formed which would receive any material leaving the reactor vessel in the event of a melt-through of the lower vessel head.

The reactor shield wall, which rests on the reactor pedestal, has a cylindrical shape and surrounds the reactor vessel up to the main steam line penetrations. The shield wall is penetrated by numerous pipes which connect to the reactor vessel. Because of the number of piping penetrations, the reactor shield wall is made of composite structural steel and concrete. Both surfaces of the shield wall are lined with carbon steel plate for strength. High-density concrete is placed between the plate surfaces for shielding. The reactor shield wall effectively reduces radiation levels in the drywell to permit inspection and maintenance when the unit is shut down.

The containment upper pool walls are above the drywell and within the containment volume. The outer walls form a rectangular pool which is subdivided by two interior sections. All of these walls are joined to the drywell roof slab which constitutes the pool base slab. The pool is completely lined with stainless steel plates. The pool consists of five regions: a moisture separator storage area; the reactor well; a steam dryer storage area; a temporary fuel storage area; and a fuel transfer region. The overall pool is ~11 m (36 ft) wide, 29.3 m (96 ft) long, and 7.3 m (24 ft) deep, while the fuel transfer and storage area is 12.8 m (42 ft) deep. The upper pool provides the following functions: radiation shielding when the reactor is in operation; storage space for the dryer, separator, and fuel assemblies during refueling: an area for fuel transfer during refueling; and a large volume of water as a suppression pool makeup water source.

The suppression pool makeup system (SPMS) provides additional water from the upper containment pool to the suppression pool by gravity flow during accident conditions. The SPMS piping consists of two lines which penetrate the drywell end of the upper containment pool through the side-walls. The elevation of the pool penetrations limits the volume of water which can be dumped to a 4.16-m (13-ft 7.75-in.) thick slice across the entire upper pool surface area.

The upper pool is dumped by gravity flow after opening two normally closed motor operated valves in series on each dump line. The upper pool dumps on receipt of a suppression pool low-low level signal [0.46 m (18 in.) below low water level] or 30 min after receipt of concurrent low reactor vessel level and high drywell pressure signals (i.e., a LOCA signal). The 30-min delay in the LOCA-induced pool dump is implemented by means of a timer which is tripped on receipt of the LOCA signal.

The secondary containment is the physical boundary which encloses the primary containment boundary, those systems external to the primary containment which would contain reactor coolant after a LOCA, and the areas in which spent fuel is stored and handled.

The purpose of the secondary containment is to prevent the uncontrolled ground level release of fission products to the environment in the event of a LOCA or a fuel handling accident. It serves as a dilution and holdup volume for fission products which may leak from the primary containment following an accident. Structurally, this is accomplished by the leak tight design of the secondary containment buildings, which are designed to leak no more than 100% of their contained volume in a 24-hour period at design negative pressure. These buildings are constructed to maintain this leak tight functional integrity in the event of an earthquake.

The external walls of the secondary containment also provide tornado missile protection for enclosed safety related components. The double doors which connect portions of the secondary containment to other areas of the auxiliary building are designed so that one door can always remain closed.

During normal operation the secondary containment areas are maintained at a pressure slightly less than atmospheric by the heating, ventilating and air conditioning (HVAC) systems serving these areas. The fuel building and the auxiliary building are maintained at a minimum of 0.825 cm (0.325 in.) of water below ambient pressure.

The normal exhaust air flow from the secondary containment is to the plant vent exhaust. This exhaust air flow is diverted to the Standby Gas Treatment System (SGTS) during abnormal or emergency conditions. There are, however, potential LOCA fission product leakage paths through the primary containment boundary that could bypass the secondary containment and thus be released to the environment without filtration by the SGTS. These consist of process piping containment penetrations that are routed through or terminated outside the secondary containment areas.

Several containment design features are provided in order to eliminate the potential for secondary containment bypass leakage. The Main Steam Isolation Valve-Leakage Control System (MSIV-LCS) is provided to collect any leakage past the MSIVs. The leakage is routed to the shield building annulus lower distribution duct header for mixing within the annulus and processing by the SGTS. The feedwater lines are provided with a positive water seal from the Residual Heat Removal (RHR) System to preclude leakage past the feedwater containment isolation valves. The HVAC supply and exhaust ductwork penetrating the containment is provided with three containment isolation valves (two outside and one inside) in which the duct between the 2 isolation valves outside containment is vented to the annulus.

The secondary containment structural boundaries encompass the shield building to containment annulus (hereafter referred to simply as "the annulus"), all of the fuel building except the stairwells and elevator vestibules, and the portions of the auxiliary building housing the emergency core cooling system (ECCS) pumps, the Residual Heat Removal (RHR) System heat exchangers, and the Reactor Water Cleanup (RWCU) System pumps. It encloses the primary containment boundary (except for the reactor building foundation mat, portions of the main steam and feedwater guard pipes and the main steam isolation valves in the steam tunnel), those systems external to the primary containment which would contain reactor coolant after a LOCA, and areas in which spent fuel is stored and handled.

The shield building is a 39.6 m (130 ft) diam cylindrical shaped, conventionally reinforced concrete structure with a shallow domed roof, 0.9-m (3-ft) thick wall, and an overall height of 60 m (197 ft).

The radial annulus, the space between the containment vessel and shield building, is 1.5 m (5 ft) wide with a minimum dome clearance of 2.3 m (7.5 ft) and a volume of 12,260 m^3 (433,000 ft^3). The walls of the shield building, which encompass the containment vessel, function as a secondary containment barrier, form the annular space for the collection and filtration of fission product leakage from the steel containment vessel, and provide biological shielding for plant personnel and the public. During normal and emergency operations, this annulus space is maintained at a slightly negative pressure relative to atmospheric pressure [approximately minus 12.7 cm (5 in.) of water] so that any leakage through the shield building or containment vessel will be into this space.

The auxiliary building is located adjacent to the reactor building and opposite the fuel building. It is supported by a reinforced concrete mat. Concrete walls and structural steel members carry vertical loads, provide lateral stability, and afford missile protection. Steel framing and grating platforms provide support to interior equipment compartments. The principal structural requirement of the auxiliary building is the support and protection of the safety and operating systems, equipment, and piping it encloses.

The emergency core cooling system (ECCS) pumps and equipment and Reactor Core Isolation Cooling (RCIC) system equipment are supported at the foundation level of the auxiliary building in watertight compartments fitted with bulkhead doors. The exhaust duct penetrations from these rooms are constructed to prevent flooding of an adjacent compartment if one of the compartments is flooded due to a pipe break. The RHR System heat exchangers are situated in vertical compartments on either side of the steam tunnel (the compartment through which the main steam lines are routed to the turbines).

The auxiliary building steam tunnel and RHR system rooms are designed to handle the consequences of high energy pipe breaks. The RHR system rooms are designed for a differential pressure of 13.8 KPa (2 psi) the associated temperature changes and jet forces. Blowout panels in the steam tunnel walls are provided to relieve pressure following a steam line rupture within the RHR system compartments.

The auxiliary building is divided into zones for ventilation purposes. The zones are necessary because of the possibility of radioactive releases or extreme environments in the secondary containment portions of the building. The ductwork routes air flow from areas of low radioactive levels to areas of potentially higher contamination. Backdraft dampers are provided in the ducts serving areas of high radioactivity levels. A pressure gradient is maintained between areas of low and potentially high radioactivity levels by exhausting more air from areas of potentially high radiation than is supplied. This prevents migration of the radioactive contaminants from the rooms.

The fuel building is the structure located adjacent to the reactor building and opposite to the auxiliary building. The fuel building houses equipment and facilities for receiving, storing, shielding, shipping, and handling fuel. A continuous reinforced concrete foundation mat supports the fuel building. The fuel building is enclosed by concrete walls and a concrete roof which are designed for tornado and missile protection. The central part of the building is occupied by the fuel pool and equipment compartments formed by concrete walls and slabs. Stainless steel liner plates seal the interior pool surfaces. The fuel building personnel and equipment entrances are provided with airtight doors to maintain the leak tightness of the building. The access doors are provided with an electrical system indicating when a door is open. A transfer tube passes fuel from the transfer compartment to the reactor building. The fuel building exhaust fans, the SGTS equipment, and the annulus recirculation/exhaust fan equipment are located in separate compartments within the building.

3.4.3 Alternative MK III containment design differences

The Grand Gulf nuclear plant incorporates an alternative containment design consisting of an auxiliary building which completely surrounds the lower portion of the concrete containment and an enclosure building which completely surrounds the containment above the auxiliary building roofline.[3.3]

The containment is a reinforced concrete structure consisting of a flat circular foundation mat, a right circular cylinder, and a hemispherical dome. Its internal surface is completely lined with welded steel plate which forms a leaktight barrier.

The containment wall is a right circular cylinder, 0.9-1.8 m (3-6 ft) thick, with an internal diameter of 37.8 m (124 ft) and a height of about 44.2 m (145 ft). The containment dome is a hemispherical shell, 0.6-1.8 m (2-6 ft) thick, with an internal diameter of 37.8 m (124 ft).

The containment provides vertical support for a number of intermediate platforms and directly supports a 125-ton polar bridge crane. Two personnel access locks with double, interlocked doors, and one equipment hatch are provided for access into the containment.

There are three vacuum relief systems associated with the MARK III Containment alternate design. The Normal Drywell Vacuum Relief System, consisting of a valved penetration from the containment to the drywell, is provided to relieve a vacuum in the drywell which may occur due to normal temperature and humidity changes in the drywell that cannot be accommodated by the Drywell Cooling System. This is not a safety system and is not connected with the other vacuum relief systems. An interlock is provided through the Drywell Purge System to keep the normal vacuum relief line closed during a LOCA.

The second vacuum relief system is part of the Drywell Purge System. Each drywell purge air compressor discharge line has a vacuum relief line tied into it. This vacuum relief function is provided only after a LOCA. Each of these two vacuum relief lines would draw air from the containment volume and discharge into the drywell to relieve the vacuum in the drywell due to steam condensation following LOCA blowdown. The vacuum breakers in the drywell purge compressor discharge lines open automatically when drywell pressure falls to within 6.9 KPa (1 psi) above containment pressure.

The third vacuum relief system is the Post-LOCA Vacuum Relief System. This system consists of two separate vacuum relief lines which share a common penetration to the drywell. This vacuum relief function is also provided only after a LOCA and serves to back up the vacuum relief provided by the vacuum relief lines associated with the Drywell Purge System. The post-LOCA vacuum relief lines open to draw air from the containment to the drywell when drywell pressure falls 3.4 KPa (0.5 psi) below that of the containment.

The auxiliary building is a reinforced concrete structure with walls several feet thick. The building, which is a multilevel structure, houses both normal and emergency auxiliary systems, the nuclear steam supply system and fuel handling facilities. The normal auxiliary systems include the Residual Heat Removal (RHR) System, Reactor Core Isolation Cooling (RCIC) System, part of the Control Rod Drive (CRD) System, and the Fuel Pool Cooling and Cleanup (FPCC) System. The emergency auxiliary systems include the Residual Heat Removal (RHR) System, High Pressure Core Spray (HPCS) System, Low Pressure Core Spray (LPCS) System, and Standby Gas Treatment System (SGTS). The building also houses electrical and instrumentation piping penetration rooms; ventilation equipment for the auxiliary building, containment, fuel handling area, and SGTS; electrical equipment such as load centers, motor control centers, and emergency cable trays; and normal and emergency process piping.

The enclosure building is a limited leakage, steel-framed, seismic Category I structure with uninsulated metal siding and insulated roof deck. It completely encloses the portions of the containment above the auxiliary building roof levels and is designed and constructed to limit leakage of radioactive materials into the environment following a loss-of-coolant accident. The structural steel frame is supported solely by struts attached on the containment shell. To maintain the required leakage limits, a flexible seal is provided around the entire periphery of the enclosure/auxiliary building interface. This seal is designed to absorb all anticipated differential seismic movements between the containment and the auxiliary building without loss of the seal's leak-tight integrity.

The annulus area between the containment and the enclosure building is maintained at a slightly negative pressure (0.25 inch w.g.) during accident conditions by the Standby Gas Treatment System.

Typical MK III alternative design parameters are listed in Table 3.4.

3.5 Containment Systems

3.5.1 Introduction

BWR containment designs incorporate several types of safety systems for the conditioning and treatment of containment atmospheres and isolation of various systems that have the capability to compromise containment integrity during accident situations. In general, each of these systems can be grouped into one of three categories: (1) mass addition or removal systems, (2) energy addition or removal systems (systems that transfer energy without exchanging mass), and (3) containment reconfiguration systems (systems that change the containment system boundaries). The purpose of this section is to describe the design and operation of these systems.

3.5.2 Residual heat removal system

The Residual Heat Removal (RHR) System, shown in Fig. 3.9, is a multipurpose system which has six or seven operational modes (plant specific), each with a specific purpose. The low pressure coolant injection mode (see Sect. 2.6.7) restores and maintains reactor vessel water level following a LOCA. The containment spray mode condenses steam and reduces airborne activity in the containment following a LOCA. The suppression pool cooling mode removes unwanted heat from the suppression pool. The shutdown cooling mode removes decay heat from the core following reactor shutdown (see Sect. 2.7.1). The steam condensing mode (BWR 5 and 6 only) condenses reactor steam and returns the resultant condensate to the reactor vessel via the Reactor Core Isolation (RCIC) System (see Sect. 2.6.6). The fuel pool cooling mode augments the Fuel Pool Cooling and Cleanup (FPCC) System if additional cooling capacity is required. The containment flooding mode allows flooding of the containment if required for post-LOCA recovery operations. Under accident conditions, the LPCI mode and later (in MK III plants) the containment spray mode are automatically initiated. All other modes require manual system alignment for proper operation.

BWR-5 and BWR-6 plants utilize 3 loop/2 heat exchanger RHR systems, while BWR-4 plants utilize 4 loop/4 heat exchanger RHR systems. Each loop of the BWR-4 system incorporates an RHR heat exchanger, while one of the RHR loops in the BWR-5 and BWR-6 systems does not incorporate a heat exchanger. In all cases, the secondary side of the heat exchangers is fed by the RHR (MK I) or station (MK II and III) service water systems or emergency standby water supply systems.

The containment cooling subsystem is an integral part of the RHR system and is placed in operation to limit the temperature of the water in the suppression pool. With the RHR in the containment cooling mode of operation, the RHR main system pumps are aligned to pump water from the suppression pool through the RHR heat exchangers where cooling takes place by transferring heat to the RHR service water.

The water pumped through the RHR heat exchanger can be diverted to spray headers in the drywell, containment building (MK III design), or above the suppression pool (plant dependant). The spray headers in the drywell condense any steam that may exist in the drywell, thereby lowering containment pressure. The spray collects in the bottom of the drywell until the water level rises to the level of the pressure suppression vent lines (MK I and II), or the suppression pool weir wall (MK III) where it overflows and drains back to the suppression pool. In some plants, part of this flow can be directed to the suppression chamber spray ring to cool any noncondensable gases collected in the free volume above the suppression pool, but the spray headers cannot be placed in operation unless the core cooling requirements of the Low Pressure Coolant Injection subsystem have been satisfied. These requirements can be manually bypassed under certain conditions.

The containment spray mode is automatically initiated in MK III plants on receipt of a high drywell pressure signal plus a 10 min time delay. Simultaneous initiation of multiple containment spray loops is prohibited by a timer which inhibits activation of a second loop until 90 seconds after initiation of the first loop. The purpose of this delay is to prohibit abrupt containment steam condensation transients that could result in containment failure due to subatmospheric containment pressures.

The flow path for the containment flooding mode (MK III systems) is from the ultimate heat sink (pond, lake, river, or ocean as appropriate for the plant site), through the service water pumps and piping into the RHR B loop downstream of the heat exchangers, through the RHR B loop discharge piping and to the suppression pool via the full flow test line. The flow can also be directed to the reactor vessel via the regular LPCI injection lines. This mode requires manual valve alignment outside the control room.

3.5.3 Standby gas treatment system

The purpose of the plant Standby Gas Treatment System (SGTS) is to process exhaust air from the Main Steam Isolation Valve Leakage Control System and the secondary containment boundary under design basis accident conditions. The SGTS can also be used to purge air from the reactor drywell under certain conditions. Each of the two or three trains of the SGTS (Figs. 3.10–3.13) consists of a moisture separator, a heating element to reduce relative humidity, a prefilter, a high efficiency particulate absolute (HEPA) filter, a charcoal filter, a second HEPA filter, and a blower. Table 3.5 is a summary listing of the locations which can be aligned to feed the SGTS system.

All SGTS trains are automatically initiated by receipt of any of the signals listed in Table 3.6. All normal containment ventilation systems automatically shut down upon receipt of a SGTS initiation signal, and all air flow is processed through the filter trains. Present SGTS designs vary widely in their rated capacities (i.e., 8,000 SCFM to 25,000 SCFM). In general, the smaller capacity systems incorporate some recirculation of filtered containment air as part of the treatment process.

The Grand Gulf MK III design (Fig. 3.13) utilizes two redundant SGTS loops, each consisting of a 17,000 cfm enclosure building recirculation fan and a charcoal/HEPA filter train with its own 4,000 cfm centrifugal blower. The recirculation fans draw air from the auxiliary building and the enclosure building, mix this air by turbulent flow in the ductwork, and return most of the mixed air to the enclosure building. The minimum mixing ratio of enclosure building air to auxiliary building air is 8:1. A portion of the recirculation fan discharge is drawn into the charcoal filter train and exhausted to the atmosphere.[3.4]

The Limerick BWR 4-MK II plant utilizes a SGTS in conjunction with a Reactor Enclosure Recirculation System (RERS) (Fig. 3.11). The RERS is activated by the same signals that initiate SGTS operation. The RERS consists of two 60,000 scfm capacity recirculation fans and charcoal/HEPA filter trains. The system takes suction from either the reactor building or the refueling floor, passes through the filter trains, and is exhausted back to the desired compartment (i.e., reactor building or refueling floor). A small portion of the return recirculation flow (~3000 scfm) is diverted to the SGTS where it is passes through the SGTS filter trains before being exhausted to the atmosphere.

3.5.4 Combustible gas control systems

Several methods are employed in BWRs to reduce the probability of gas combustion within the primary containment. In general, these systems can be classified as hydrogen or oxygen reduction systems, atmospheric mixing systems, and containment venting systems. All domestic MK Is and MK IIs employ a primary containment inerting system which maintains the drywell and wetwell atmospheric oxygen fractions at very low values (typically ~4%) during normal plant operations. This is accomplished by injecting nitrogen into the primary containment atmosphere while purging the resulting containment gas mixture via the SGTS or containment ventilation system. The inerting system continues to supply nitrogen to the containment during operation, to account for oxygen mass concentration changes resulting from containment atmospheric temperature changes, leakage, etc.

MK Is and some MK IIs also employ a containment atmospheric dilution (CAD) system which is designed to maintain the post LOCA containment atmospheric oxygen fraction below 5%. This is accomplished by feeding nitrogen to the drywell and/or the wetwell atmospheres, while purging the selected compartment atmosphere via the Standby Gas Treatment System. The operator manually controls the CAD system nitrogen feed and containment venting flow rates and frequencies.

Many plants utilize thermal recombiners for combustible gas control. The exact type and configuration of the recombination system varies significantly from plant to plant. One system employed in a MK II facility utilizes a recombiner located outside the primary containment. The hydrogen-oxygen recombination process takes place within the recombiner as a result of an exothermic reaction. The steam is cooled, and the resulting water and remaining gases are returned to the primary containment. The cooling water used to cool the return gases is taken from the RHR system. Recombiner suction is taken from the drywell and the discharge is returned to the suppression chamber air space. The system, which requires a 1 to 2 hour warmup, is manually controlled by the operator. The recombiners in MK III plants are typically natural convection units, located within the primary containments. The waste heat from these systems (~50 kW each for two recombiners) is dumped directly into the containment atmosphere.

In addition to recombiners, MK III plants (which are not inerted) utilize two other systems for combustible gas control. A hydrogen mixing system (Fig. 3.14) is utilized to draw air from the containment and discharge it into the drywell. The resulting drywell pressurization depresses the pressure suppression pool level within the drywell, uncovering some of the suppression pool vents. This allows the drywell and containment atmosphere to mix. The compressors for this system are located within the containment. The second system utilized in standard MK III facilities is a containment purge system. This system is utilized in conjunction with the hydrogen mixing system described above for cases in which the mixing system cannot adequately control the hydrogen concentrations within the drywell and containment. The purge system employs a 2 in. drywell vent line which terminates in the annulus between the containment and the shield building. This line is opened as necessary to vent the containment to the annulus compartment where the gas is subsequently treated and released by the SGTS. A similar venting system is available in MK I plants, where the drywell can be vented to the SGTS.

Finally, hydrogen igniter systems are currently being installed in some MK III plants to assist the operators in controlling containment combustible gas fractions in the event of a severe accident. These systems are designed to initiate "burning" of combustible mixtures before "explosive" gas concentrations are reached.

3.5.5 MK I and MK II containment HVAC systems

As previously described (Sects. 3.1-3.3), MK I and MK II designs incorporate a primary containment consisting of a drywell and suppression chamber, and a secondary containment consisting of a reactor building which completely encloses the drywell/wetwell system. Figure 3.15 is a simplified schematic representation of the MK I (also typical of MK IIs) containment ventilation system (drywell coolers not shown).

All MK I and MK II containment designs utilize drywell atmospheric cooling systems for the maintenance of appropriate drywell conditions during normal operation. The temperature of the drywell is maintained by multiple fan forced cooling units which incorporate heat exchangers to transfer energy to the reactor building closed cooling water system. This heat removal capability is necessary to balance drywell atmospheric heat inputs from sources such as motors and the reactor vessel and steam line surfaces during both reactor operation and after reactor scram. The units can be powered by emergency power supplies and would be available during many severe accident situations. Drywell cooler systems typically have rated heat removal capacities of 5×10^6 Btu/h (Browns Ferry).

The reactor building is heated, cooled, and ventilated during normal and shutdown operation by a circulating air system. The reactor building heating and ventilating system is shut down and isolated when the secondary containment is isolated and connected to the Standby Gas Treatment System. While the reactor building heating and ventilating system is not an engineered safeguard, certain components do perform engineered safeguard functions. The double isolation valves, the vacuum relief valves, and the equipment area cooling units serve engineered safeguard systems and are designed to engineered safeguard standards and criteria.

The ventilation system provides 100% makeup air. Outside air is filtered and then passes across hot water coils for winter heating and through evaporative coolers for summer cooling, and hence to the supply fans. The filters, coils, coolers, and supply fans are located outside the reactor building. The ventilation system supplies 23.5 m^3/s (50,000 ft^3/min) of air per unit to the refueling zone [11.8 m^3/s (25,000 ft^3/min) during heating season]. The reactor zone ventilation system supplies 47.2 m^3/s (100,000 ft^3/min) of air per unit [23.5 m^3/s (50,000 ft^3/min) during the heating season].

The ventilation of air from the reactor building is ducted to exhaust fans located on the reactor building roof. The air from each zone is monitored before release. High activity will isolate the secondary containment. Normal ventilation air exhaust is not filtered.

The RHR pumps and the core spray pumps are located in the basement rooms of the reactor building. The heat loss from the motors, pumps, and piping is removed with air-cooling units.

The reactor building ventilation system can also supply 2.8 m^3/s (6000 ft^3/min) to the drywell or pressure suppression chamber. This air is used for purge and ventilation of the primary containment system. The purge and ventilation exhaust from the primary containment is first processed by a filter train assembly and then channeled through the reactor building exhaust system. The primary containment purge and ventilation system is isolated from the primary containment by two isolation valves in series, during power operation. These valves are part of the primary containment isolation system.

3.5.6 MK III containment HVAC systems

3.5.6.1 Primary containment systems. The primary containment HVAC system (consisting of the Drywell Recirculation System, Drywell Purge Ventilation System, Containment Normal Ventilation System, Containment High Flow Purge System, and the Containment Recirculation System) provides an environment with controlled temperatures, humidities, and air flow patterns to ensure the comfort and safety of personnel and the operability of equipment located in the containment. It is used to remove potentially radioactive air from the containment during normal operation and to provide outside air for the purge of the drywell during refueling operations. The primary containment HVAC system is shown schematically in Figs. 3.16 and 3.17.

Rooms or areas which might contain relatively high airborne radioactivity levels are exhausted so as to maintain them at a negative pressure with respect to the general containment volume. Exhaust from the containment exhaust fans is normally directed to the plant exhaust vent, but can be manually diverted to the Standby Gas Treatment System. All Primary Containment HVAC System equipment is supplied power by the Standby AC Power System.

The Drywell Recirculation System conditions the air in the drywell to maintain it within acceptable environmental conditions. Each drywell recirculating air handling unit (AHU) houses two cooling coils and a motor driven centrifugal fan which delivers 18.9 m^3/s (40,000 cfm).

The drywell recirculating AHUs circulate the existing air through the drywell since there are no normal sources of supply or exhaust air to the drywell. The drywell recirculation units are divided into two groups, each consisting of three air handling units and one electric heating coil with a common header. Two units on each group are normally operating with the third unit in standby.

The Drywell Purge Ventilation System is used to purge the drywell during shutdown or refueling prior to personnel entry into the drywell. During purging operations, fresh air is supplied to the containment by the containment high flow purge AHU and the normal supply AHU. The normal containment exhaust fan and the high flow purge containment exhaust fan take a suction on the drywell and containment areas and exhaust this air via the normal exhaust.

The Containment Normal Ventilation System, shown in Fig. 3.17 consists of a normal supply AHU and a normal exhaust fan. When radiation monitors detect a high radiation level in the containment exhaust, all ventilation primary and secondary containment penetrations are automatically isolated.

The containment high flow purge system supply AHU is used in conjunction with the containment high flow purge exhaust fan during refueling to provide additional air flow through the containment. This removes heat, vapor, and radioactive particles from above the upper containment pools for personnel comfort and to minimize radiation exposure. The high flow purge supply AHU and purge exhaust fan are energized during drywell purging prior to personnel entry to provide additional air flow to reduce temperature and radiation levels within the containment. Containment exhaust, instead of being directed to the plant exhaust vent (normal flow path), can be directed to the Standby Gas Treatment System (SGTS) filter train.

The containment recirculation system AHUs, maintain the containment at the proper temperature and humidity during normal operating conditions. Air to the containment is introduced near the intakes of the recirculation AHUs to ensure a uniform distribution of makeup air throughout the containment. Each containment recirculation AHU houses a prefilter, a cooling coil, and a motor driven centrifugal fan. The units are divided into two groups each consisting of three AHUs and an electric heater common to all three AHUs. Two of each group of three AHUs are normally running with the other AHU in each group in standby. There are also two containment dome fans which are located in the top of the containment area. Both fans are normally operating to prevent hot air pocketing in the containment dome.

3.5.6.2 Secondary containment systems. The annulus HVAC system provides means of monitoring, controlling, and treating effluents from the annulus prior to release to the environment. The annulus HVAC system, shown in Fig. 3.18, consists of an upper and lower duct ring header, two recirculation/exhaust fans, motor operated control dampers, and motor operated isolation dampers. During normal operations, one of the two redundant recirculation/exhaust fans is operating, with the other fan in standby. These recirculation/exhaust fans take suction from the top of the annulus through the duct ring header and discharge flow through the motor operated exhaust damper and recirculation damper. The exhaust damper for the operating fan is positioned to exhaust as much air as necessary to maintain an annulus pressure of negative 5 inches w.g. The portion of the flow not required for maintaining the annulus negative pressure is exhausted through the lower duct header. This recirculation flow prevents hot spots and heat buildup in the upper portions of the annulus. Also, the recirculation allows time for short-lived isotopes to decay prior to discharge.

The exhaust from the annulus normally is through the isolation dampers to the plant vent, via the operating containment exhaust fan. An additional air flow connection with the annulus is from the containment vacuum relief valves. These valves are provided to prevent exceeding the containment external design pressure. Each vacuum relief valve consists of a pipe, a check valve, and an air operated valve which connect the containment to the shield building annulus. The shield building annulus functions as a collection point for post LOCA MSIV-LCS flow, post-LOCA drywell purge flow, post LOCA containment duct isolation valve leakage flow, and drywell pressure bleed-off vent flow during reactor heatup.

The auxiliary building is divided into zones for ventilation purposes. The zones are necessary because of the possibility of radioactive releases or extreme environments occurring in the secondary containment portions of the building. The ductwork routes air flow from areas of low radioactive levels to areas of potentially higher contamination. Backdraft dampers are provided in the ducts serving areas of high radioactivity levels. A pressure gradient is maintained between areas of low and potentially high radioactivity levels by exhausting more air from areas of potentially high radiation than is supplied.

The Auxiliary Building HVAC System, shown in Fig. 3.18, consists of the following major components: two full capacity pressure control supply AHUs; two full capacity exhaust fans; and eight fan coil units (FCUs). During normal operation filtered and tempered outside air is supplied to the general areas and corridors of the auxiliary building, to the fuel building stairwell vestibule, and to the HVAC equipment area by one of the pressure control supply air handling units and will be drawn through the ECCS, RCIC System, and RWCU System equipment rooms (within the auxiliary building) and exhausted from the building by one of the exhaust fans. Cooling water for the AHU cooling coils is supplied by the Chilled Water System.

The ECCS, RCIC System, and RWCU System equipment rooms and corridors surrounding these rooms are maintained at a negative pressure (~0.325 inches w.g.) with respect to the outdoors and the surrounding areas of the auxiliary building not included in the secondary containment. This slightly negative pressure is desirable because it limits any possible spread of airborne radioactive particles to the atmosphere. Since the secondary containment portion of the auxiliary building is at a negative pressure with respect to surrounding areas, any leaks are into the auxiliary building secondary containment. The reactor auxiliary equipment and RWCU system equipment rooms in the auxiliary building secondary containment are cooled by individual fan coil units (FCUs). Each FCU circulates the equipment room air through cooling coils of each FCU under normal, loss of preferred power, and accident conditions except for the RWCU System fan coil units during a LOCA. The secondary containment portion of the Auxiliary Building HVAC System can be isolated so that these areas can be ventilated by the Standby Gas Treatment System.

The Fuel Building HVAC System, shown in Fig. 3.18, consists of the following components: pressure control supply AHUs; full capacity exhaust fans; fan coil units (FCUs); unit heaters; and associated ducts, dampers, and controls. Differential pressure controllers modulate inlet vanes at the suction of the exhaust fans to control fuel building exhaust flow and maintain the fuel building negative pressure (-0.325'' w.g.). The supply AHUs supply tempered, filtered air for all portions of the fuel building except for stairwells and elevator vestibules served by other systems. Inlet air to the supply AHUs is normally one-third outside air and two-thirds return air from the general areas of the fuel building. Normally, both supply AHUs are operating and one exhaust fan is operating. The second exhaust fan remains in standby. This standby fan will automatically start when the discharge air flow from the operating exhaust fan is low.

The Fuel Building HVAC System ductwork is designed to prevent the spread of radioactive contaminants within the fuel building. The ductwork routes air flow from areas of low radioactivity levels to areas of potentially higher contamination. Exhaust from areas with potentially high radiation levels is not returned to the supply AHUs, but is routed to the exhaust fan suction via the exhaust duct, where it is exhausted to the plant vent.

The individual equipment room and area environments are automatically maintained by fan coil units and electric unit heaters. With the exception of the fan coil units serving the SGTS equipment rooms, the shield annulus recirculation/exhaust fan rooms, and the fuel pool cooling and cleanup pump rooms (all safety related), the fan coil units utilize chilled water supplied by the Chilled Water System as the cooling medium. The FCUs serving the safety related equipment rooms are supplied cooling water by the Standby Service Water System.

3.5.7 Alternative MK III containment HVAC systems

The maintenance of desirable environmental conditions within the containment of the alternative MK III design is performed by the Drywell Cooling, Containment Cooling, Containment Ventilation, Containment Filtration, and Auxiliary Building Ventilation Systems. All of these systems except the Auxiliary Building Ventilation System are shown schematically in Fig. 3.19.

The Drywell Cooling System consists of recirculating fan coil units and the associated dampers, ducting, and controls. Each FCU consists of two full capacity fans in parallel and two full capacity cooling coils in series. Six fan coil units are provided to distribute cooling air effectively and with minimum duct work. Normally, one fan and one coil of each fan coil unit operate with the other fan and coil in standby. Each unit represents 25% of total capacity.

The Containment Cooling System recirculates the containment atmosphere to maintain design conditions of 300 K (80°F) and 60% relative humidity during normal plant operation. The Containment Cooling System consists of recirculation coolers and the associated dampers, ducting, and controls required to maintain the design containment temperature and relative humidity. Each containment cooler consists of a cooling coil and fan. Normally, two fan coil units are operating with the third in standby.

The Containment Ventilation System consists of two 100% capacity containment ventilation supply fans, two 100% capacity containment ventilation exhaust fans, one 100% capacity containment exhaust charcoal filter train, and the associated ducting, dampers, and controls required to provide a reliable source of fresh air for the comfort and safety of personnel. The containment exhaust charcoal filter train consists of the following components arranged in series with respect to air flow: demister, heating coil, prefilter, high efficiency particulate air (HEPA) filter bank, charcoal filter bank, and HEPA filter bank. A small amount of the containment atmosphere is continuously exhausted during normal operation via the containment exhaust charcoal filter train and one of the containment ventilation exhaust fans.

The Containment Filtration System consists of two 100% capacity containment cooling charcoal filter trains that continuously recirculate a portion of the containment atmosphere to limit the concentration of airborne radioiodines to an acceptable level during normal operation. Each filtration train consists of the following components arranged in series with respect to air flow: demister, heating coil, prefilter, HEPA filter bank, charcoal filter bank, HEPA filter bank, and centrifugal fan. The heating coil in each filtration train provides humidity control.

The Containment Cooling, Ventilation, and Filtration System has several modes of operation. The modes are as follows:

1. Normal operating mode
2. Containment purge mode
3. Containment cleanup mode
4. Drywell purge mode

During the normal operating mode, the containment is maintained at 300 K (80°F) and 60% humidity by recirculating the air through the containment coolers. Makeup air as required for personnel access is supplied to the containment by the ventilation supply fans in a quantity approximately equal to that exhausted by the containment ventilation exhaust fans and associated exhaust filter train.

During the containment purge mode, the entire volume of air routed to one or both charcoal filter trains is discharged to the atmosphere with no recirculation to the containment. The containment ventilation supply and exhaust fans are idle during this mode. The drywell/containment purge fans supply makeup air during this mode. Both charcoal filter trains and purge fans can be used to provide additional purge capacity.

During the containment cleanup mode, the containment atmosphere is routed through one or both charcoal filter trains and the recirculation supply and exhaust fans and the drywell/containment purge fans are idle during this mode.

The Auxiliary Building Ventilation System is designed to provide an environment with controlled temperature and humidity to ensure comfort and safety of personnel and the integrity of auxiliary building equipment. The auxiliary building is divided into six ventilation zones as follows: zones 1-4 for the first through fourth floors, respectively; zone 5 for the fuel handling area; and zone 6 for the pipe tunnel outside the containment.

All zones except the fuel handling area zone are provided with fan coil units with heating and cooling coils. During normal plant operation, each fan coil unit supplies conditioned air to its respective zone. When a given unit is started, its outside air damper opens to its preset normal position to regulate the amount of makeup air to be supplied to the zone.

Since the auxiliary building comprises part of the boundary area for the Standby Gas Treatment System, any SGTS initiation causes isolation of the Auxiliary Building Ventilation System. During normal operation, fan coil units in zones 1-4 provide cooling to rooms occupied by ECCS equipment which is normally idle. When ECCS equipment is in operation, cooling is provided by safety grade equipment area cooling units.

The fuel handling area zone has the following ventilation equipment:

1. Two 100% capacity fuel handling area supply fans which provide ventilation and makeup air to the space during normal operation and during pool sweep equipment operation.

2. Two 100% capacity fuel handling area exhaust fans which exhaust air from the fuel handling area and other areas within the auxiliary building during normal operation.
3. One 100% capacity fuel handling area fan coil unit which recirculates the fuel handling area atmosphere and maintains space design conditions.
4. Two 100% capacity fuel pool sweep supply fans and two 100% capacity fuel pool sweep exhaust fans to provide a controlled circulation of air across the surface of the spent fuel pool, the fuel cask storage pool, and the transfer canal during fuel handling operations.

During normal operation, the fuel handling area is ventilated and maintained at a slightly negative pressure with respect to its surrounding areas by the fuel handling area supply and exhaust fans. During fuel handling operations, the fuel pool sweep supply and exhaust fans are run to supply and remove air across the surface of the pools.

Radiation elements are installed in the suctions of the fuel handling area exhaust fans and the fuel pool sweep exhaust fans. Upon high sensed radiation by either of these elements, the SGTS is started which then causes the Auxiliary Building Ventilation System to isolate.

3.5.8 Secondary containment fire protection sprays

All domestic BWRs incorporate some form of spray system to provide automatic fire protection for areas inside the secondary containment. These systems typically utilize wet pipe sprinkler systems whose operation is initiated in the event of a rise in ambient temperature to the melting point of the fusible links on the sprinkler heads. The flow of water through an alarm check valve energizes a flow switch which starts the system pumps. In addition to this automatic mode of operation the system can also be initiated by local smoke detectors or manually. Typical system spray rates are $7 \times 10^{-5} m^3/s$ (0.15 gpm per square foot) of floor area.[3.5]

3.6 BWR Containment Structures and Systems - Summary

This chapter has presented a summary description of the BWR containment structures and systems which might influence the outcome of a severe accident. These structures and systems are summarized for each of the major containment types in Tables 3.7 through 3.10. The systems and structures in these tables are classified as either heat transfer (Q) or mass transfer (M) systems.

Figures 3.20 through 3.24 are schematic representations of generic MK I, MK II, standard MK III, and alternative MK III containments which incorporate all of the features listed in Tables 3.7 through 3.10. It should be emphasized that no single BWR plant will contain all of the features shown in Figs. 3.21 through 3.24.

An illustration of the usefulness of Figs. 3.21 through 3.24 can be seen by examining Fig. 3.21. The figure illustrates that there are four major compartments within the MK I containment. The drywell contains the reactor vessel, recombiners, fan coolers, sprays, and heat conducting slabs, and is connected to the standby gas treatment system and nitrogen injection systems. The suppression pool is connected with the drywell by vent pipes which extend below the pool surface. The suppression pool serves as a destination for HPCI and RCIC turbine and SRV exhaust steam and is connected to a cooling system. The pool also serves as a source of water for ECC systems. The air space above the pool (in the pressure suppression chamber) is connected to the drywell vents and the reactor building atmosphere via vacuum breakers and contains a spray system. The reactor building houses the drywell and pressure suppression chamber and is connected to the refueling floor via blowout panels. Additionally, the reactor building serves as a heat sink for waste heat from various reactor systems and is cooled by fan coil cooling units. The atmosphere of the reactor building is connected to the suppression chamber (via vacuum breakers), the standby gas treatment system, and spray systems. Finally, the refueling floor is connected to the reactor building and the outside atmosphere via blowout panels and the standby gas treatment system.

Chapter 3 References

3.1 *Browns Ferry Nuclear Plant Finc' Safety Analysis Report*, Tennessee Valley Authority.

3.2 *BWR-6 Boiling Water Reactor Systems Manual*, U.S. Nuclear Regulatory Commission.

3.3 *Grand Gulf Nuclear Station Final Safety Analysis Report*, Mississippi Power and Light Company and Middle South Energy Inc.

3.4. S. A. Hodge and R. A. Lorenz, *Standby Gas Treatment System Operation and Effectiveness under Severe Accident Conditions*, Letter Report, Oak Ridge National Laboratory (May 1983).

3.5. *General Electric Standard Safety Analysis Report GESSAR II*, Chap. 9.5, General Electric Incorporated (1982).

Table 3.1. Typical MK I primary containment design characteristics

	Drywell	Suppression chamber
Internal design pressure, psig	56	15
External design pressure, psig	2	0.8
Design temperature, °F	281	185
Free volume, ft^3	159,000	119,000 (minimum)
Suppression pool water volume, ft^3	135,000 (max)	

Table 3.2. Typical MK II containment design characteristics

	Drywell	Suppression chamber
Internal design pressure, psig	45-55	15
External design pressure, psig	5	0.8
Design temperature, °F	340	185
Free volume, ft^3	160,000-240,000	93,000-170,000
Suppression pool water volume, ft^3	74,000-160,000	

Table 3.3. Typical Standard MK III containment design characteristics

	Drywell	Suppression chamber
Internal design pressure, psig	30	15
External design pressure, psig	21	0.8
Design temperature, °F	330	185
Free volume, ft^3	275,000	1,168,000
Suppression pool water volume, ft^3 (max)	12,000	120,000
Shield building volume, ft	400,000	

Table 3.4. Typical Alternative MK III containment design characteristics

	Drywell	Suppression chamber
Internal design pressure, psig	30	15
External design pressure, psig	21	3
Design temperature, °F	330	185
Free volume, ft^3	275,000	1,400,000
Suppression pool water volume, ft	13,000	125,000
Auxiliary building volume, ft^3	3,000,000	
Enclosure building volume, ft^3	600,000	

Table 3.5. SGTS suction locations

Compartment	Containment type (MK)
Drywell	All
Suppression chamber	I, II
HPCI gland seal exhaust blower	I
Reactor building	I, II
Refueling zone	I, II
Fuel building	III
Auxiliary building	III, IIIA[a]
Shield building annulus	III
Enclosure building	IIIA[a]
Main steam isolation valve leakage control system	II, III

[a]Alternative MK III design.

Table 3.6. SGTS initiation signals

Reactor zone high radiation
Refueling zone high radiation
Low reactor water level
High drywell pressure
Containment exhaust high radiation
Fuel building high radiation
Fuel pool airspace high radiation
Loss of preferred power

Table 3.7. MK I systems and structures[a]

	Q_{in}	Q_{out}	M_{in}	M_{out}
Drywell	Recombiners RV	DC DS	DV Sprays VB Break flow CAD	DV SGTS
Wetwell		RHR	HPCI/RCIC Turbine exhaust SRVs Sprays VB DV	VB ECCS suction
RB/RXZ	ECCE	RC	Sprays Break	BP VB SGTS
RB/RFZ			Sprays BP	SGTS BP

[a]

RV	Reactor Vessel
DS	Drywell Structures
DV	Drywell Vents
DC	Drywell Coolers
VB	Vacuum Breakers
CAD	Containment Air Dilution System
ECCE	Emergency Core Cooling Equipment
RC	Emergency Core Cooling Equipment Room Coolers
BP	Blowout Panels
RB/RXZ	Reactor Zone of Reactor Building
RB/RFZ	Refueling Zone of Reactor Building

Table 3.8. MK II systems and structures[a]

	Q_{in}	Q_{out}	M_{in}	M_{out}
Drywell	Recombiners RV	DC DS	Sprays DV Break flow CAD VB	DV SGTS
Wetwell		RHR	HPCI/RCIC Turbine exhaust SRV Sprays DV	ECCS suction
RB/RXZ	RERS ECCE	RERS RC	RERS SGTS Sprays Break	RERS SGTS BP
RB/RFZ	RERS	RERS	RERS SGTS	RERS SGTS

[a] RV Reactor Vessel
DS Drywell Structures
DV Drywell Vents
DC Drywell Coolers
VB Vacuum Breakers
CAD Containment Air Dilution System
ECCE Emergency Core Cooling Equipment
RC Emergency Core Cooling Equipment Room Coolers
BP Blowout Panels
RB/RXZ Reactor Zone of Reactor Building
RB/RFZ Refueling Zone of Reactor Building
RERS Reactor Enclosure Recirculation System

Table 3.9. Standard MK III systems and structures[a]

	Q_{in}	Q_{out}	M_{in}	M_{out}
Drywell	RV Recombiners	DRS DS	RHR/sprays Break flow VB H_2 mix sys. DV	DV SGTS
Containment		RHR	VB DV RHR/sprays RHR/flood RCIC turbine exhaust SRV	SGTS H_2 mix sys. Purge sys. ECCS suct'
Annulus	Conduction through cont. walls		Drywell purge	SGTS VB
Fuel Bldg			Fire sprays	SGTS
Auxiliary Bldg	ECCE	RC	Fire sprays	SGTS

[a]
RV Reactor Vessel
DS Drywell Structures
DV Drywell Vents
DC Drywell Coolers
VB Vacuum Breakers
DRS Drywell Recirculation System
CAD Containment Air Dilution System
ECCE Emergency Core Cooling Equipment
RC Emergency Core Cooling Equipment Room Coolers
BP Blowout Panels

Table 3.10. Alternative MK III systems and structures[a]

	Q_{in}	Q_{out}	M_{in}	M_{out}
Drywell	RV Recombiners	DC DS	RHR/sprays VB DV Break flow	DV SGTS DP
Containment		CC	RHR/sprays DV RCIC turbine exhaust	SGTS ECCS suction
Auxiliary Bldg		RC	Fire sprays	SGTS
Enclosure Bldg			SGTS	SGTS

[a] RV Reactor Vessel
DS Drywell Structures
DV Drywell Vents
DC Drywell Coolers
VB Vacuum Breakers
DRS Drywell Recirculation System
CAD Containment Air Dilution System
ECCE Emergency Core Cooling Equipment
RC Emergency Core Cooling Equipment Room Coolers
BP Blowout Panels
DP Drywell Purge
CC Containment Cooling

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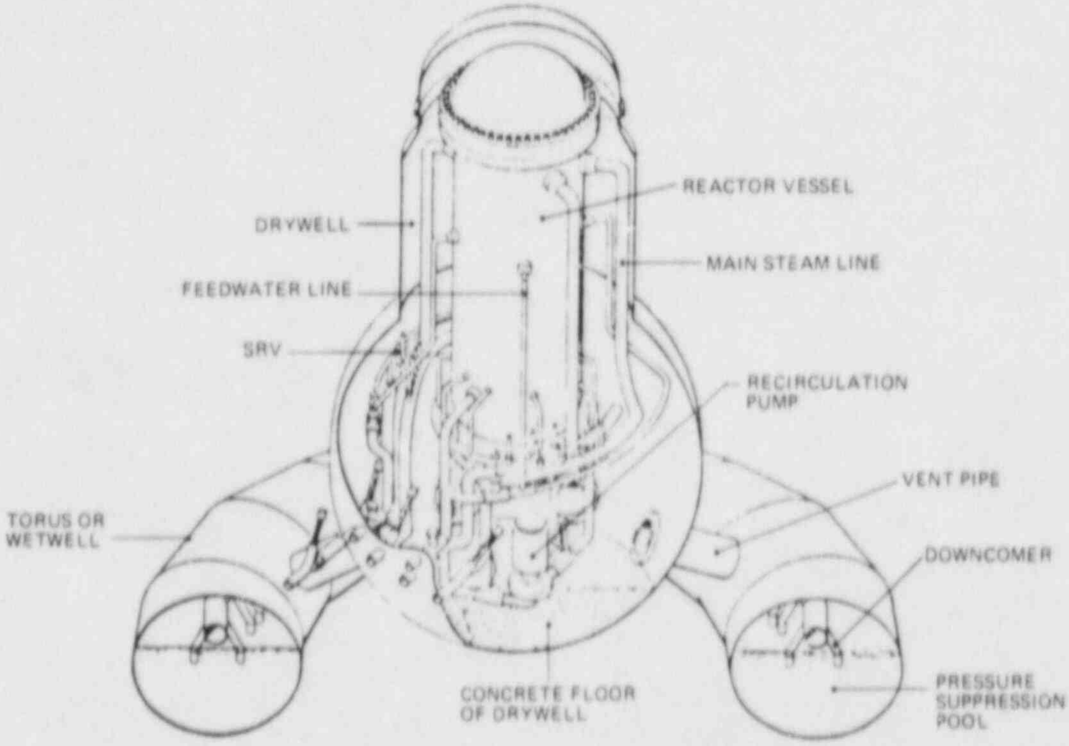

Fig. 3.1. MK I drywell/torus arrangement.

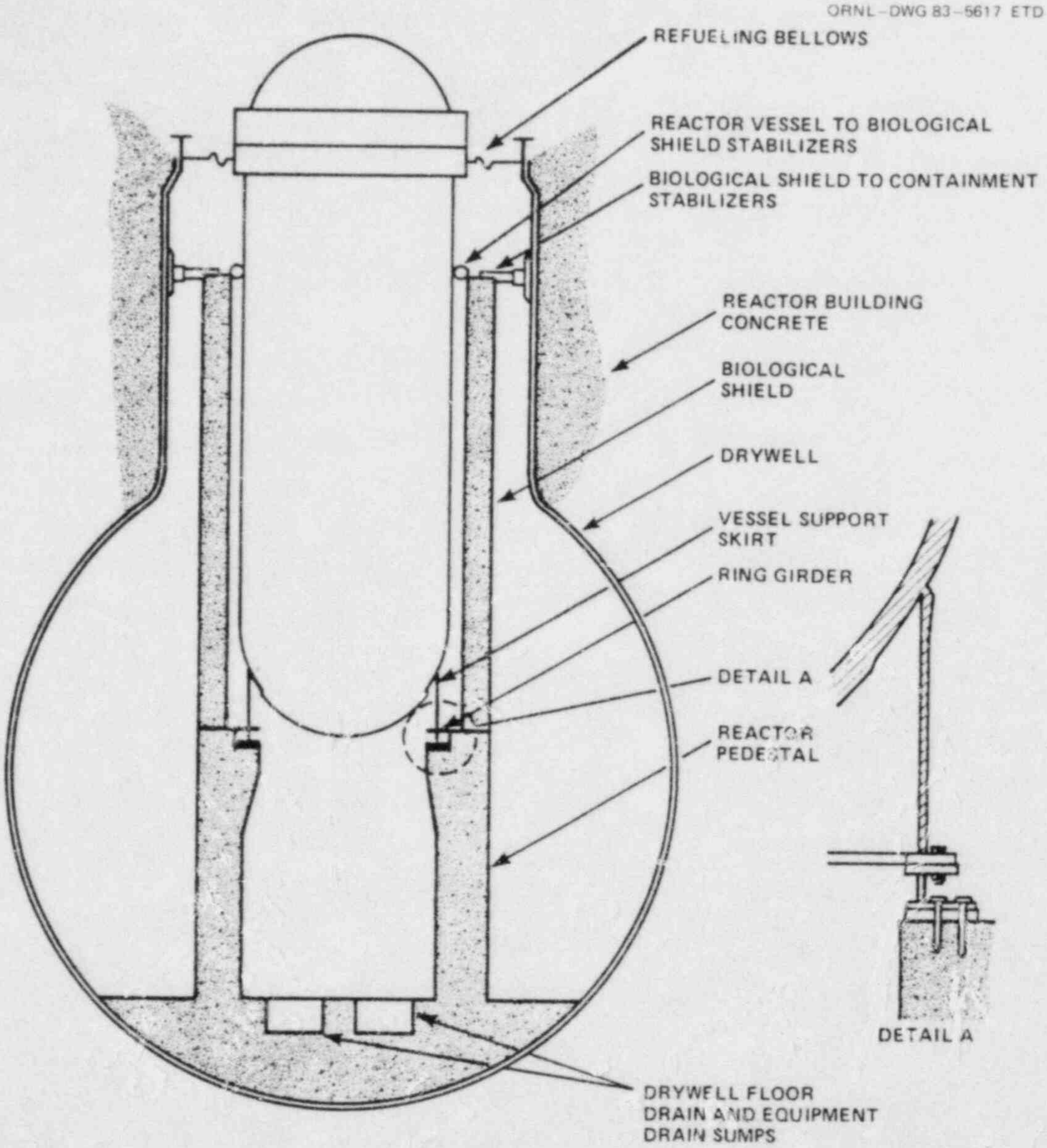

Fig. 3.2. Vessel support and biological shield wall.

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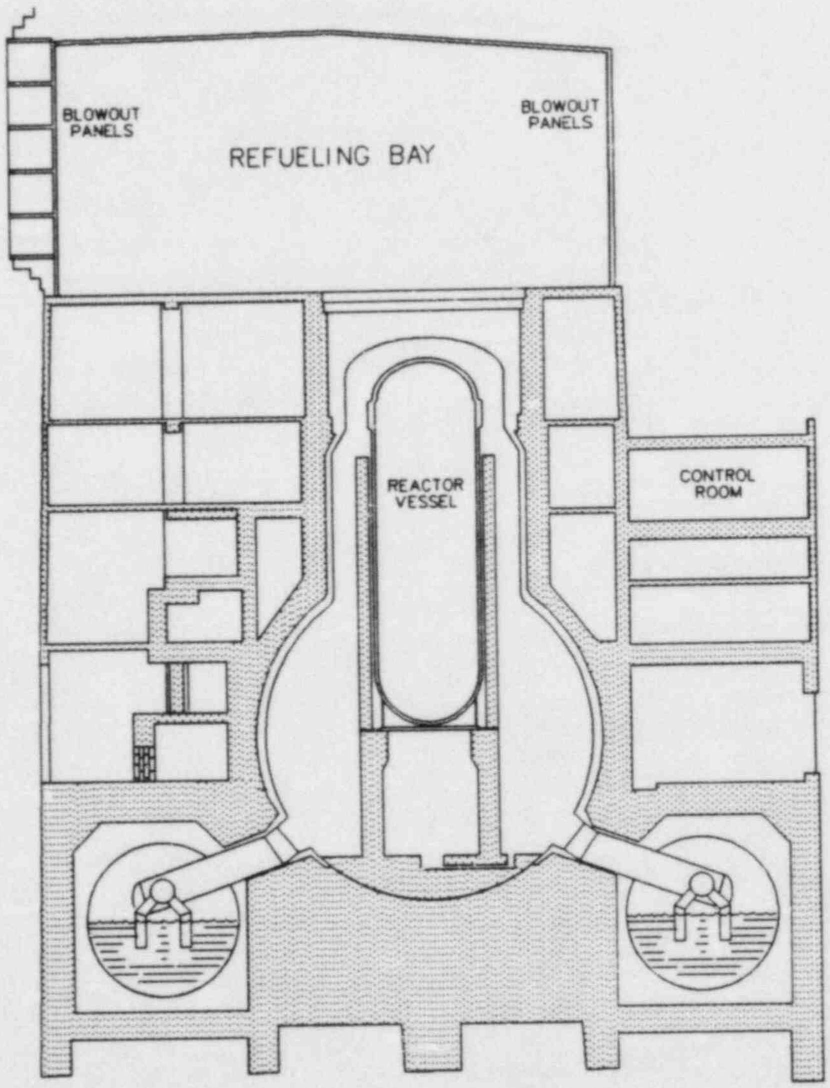

Fig. 3.3. MK I containment.

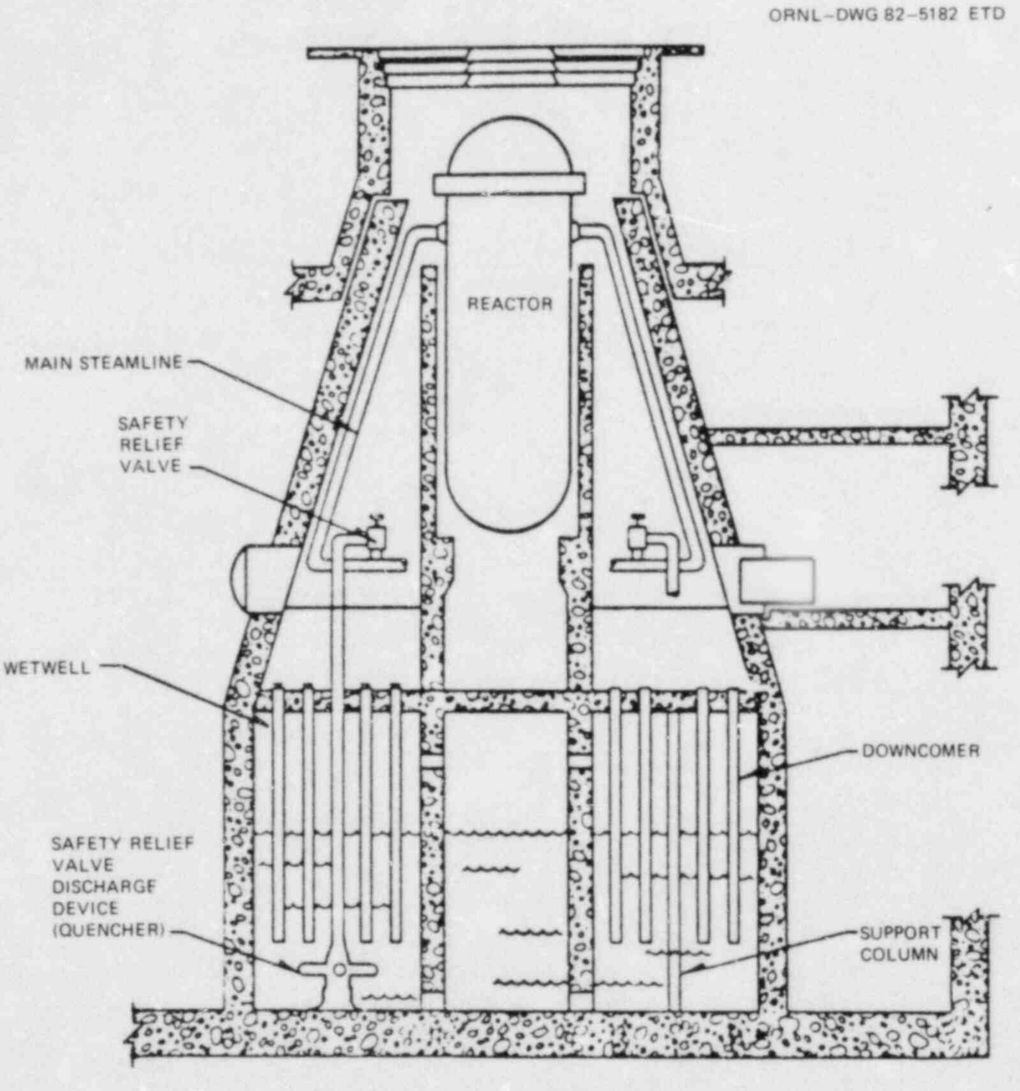

Fig. 3.4. Limerick MK II containment.

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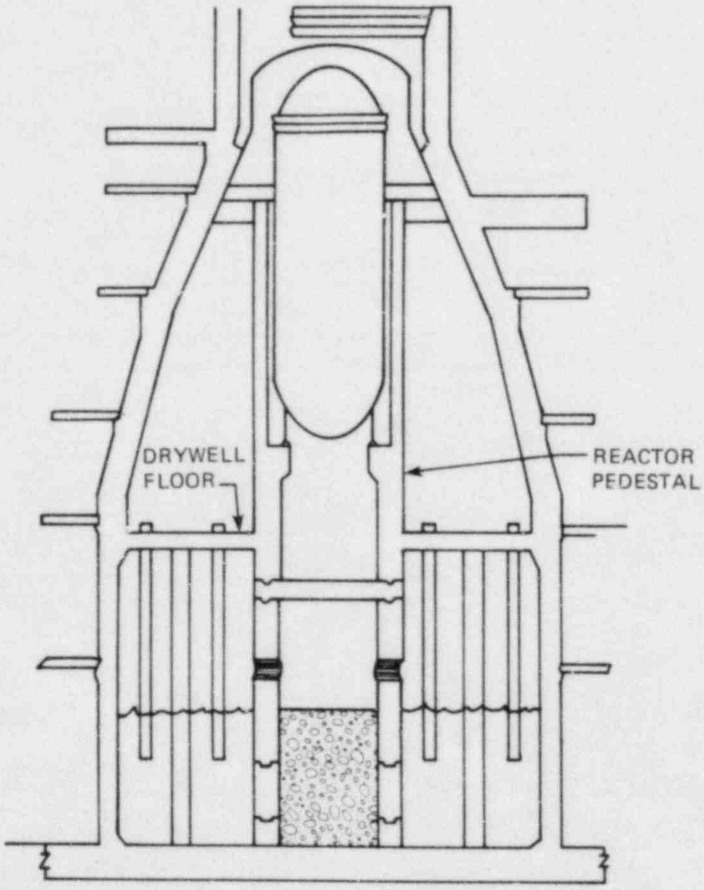

Fig. 3.5. LaSalle MK II containment.

ORNL-DWG 83-5619 ETD

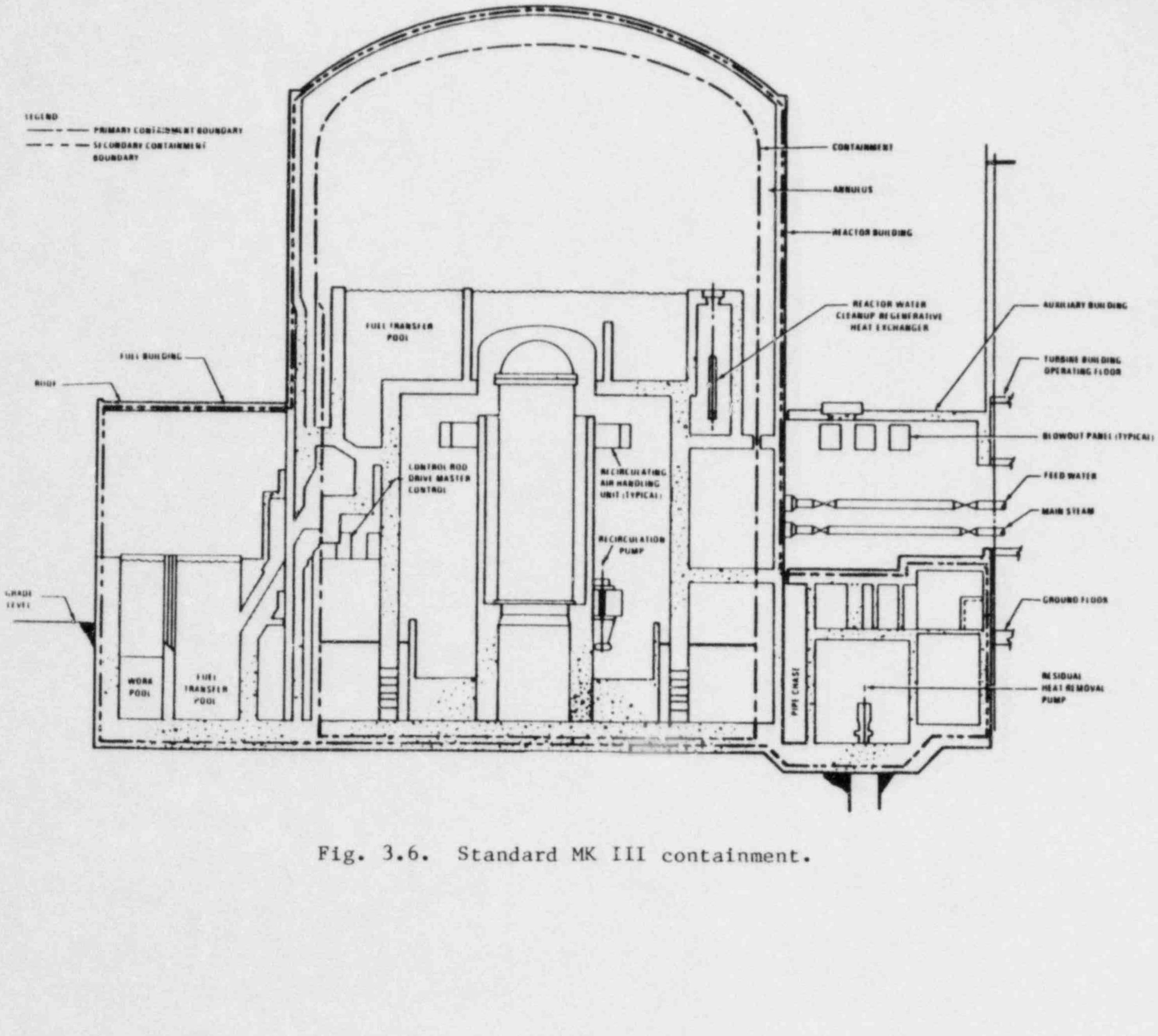

Fig. 3.6. Standard MK III containment.

ORNL-DWG 83-5620 ETD

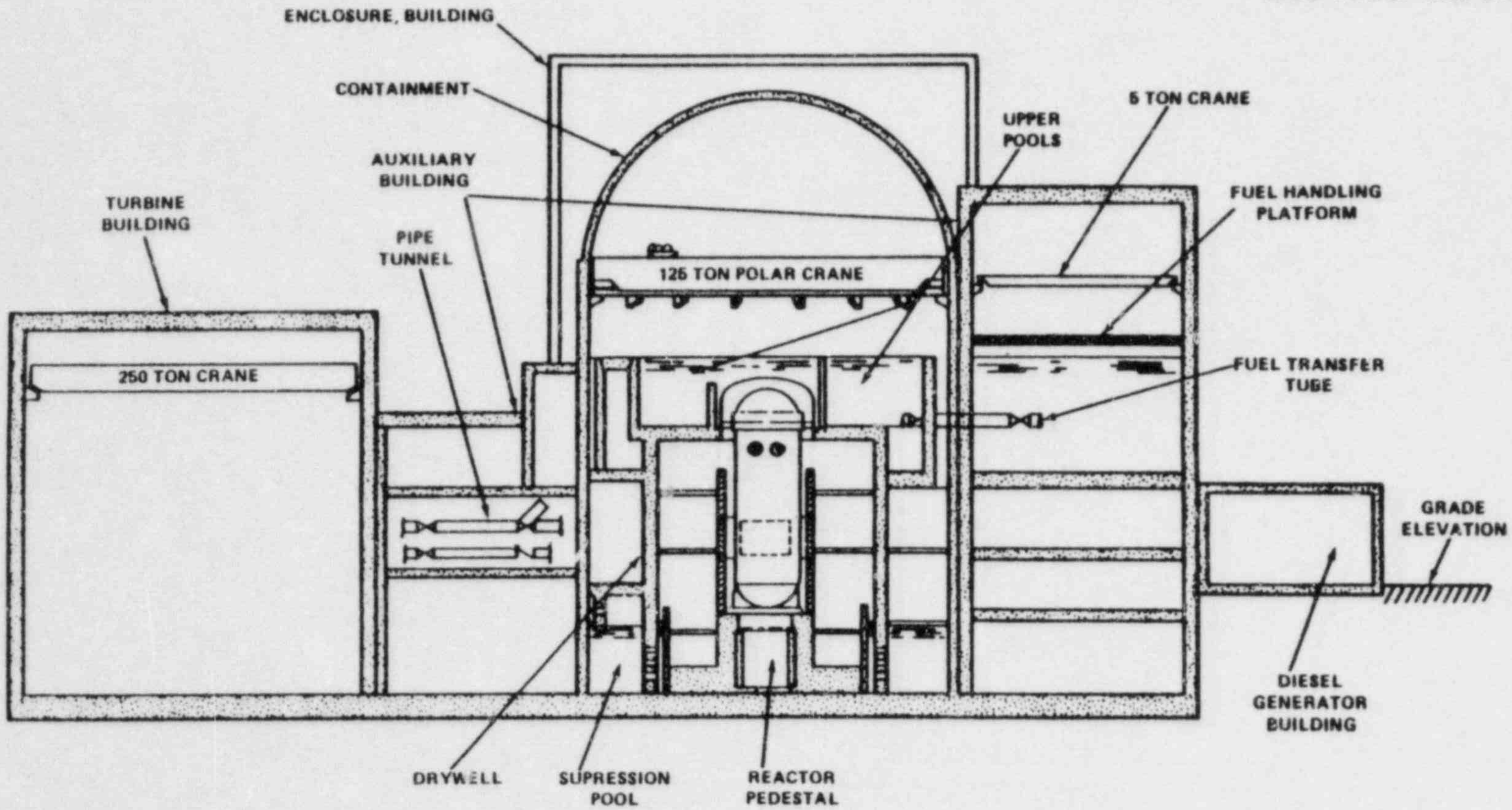

Fig. 3.7. Alternative (Grand Gulf) MK III containment.

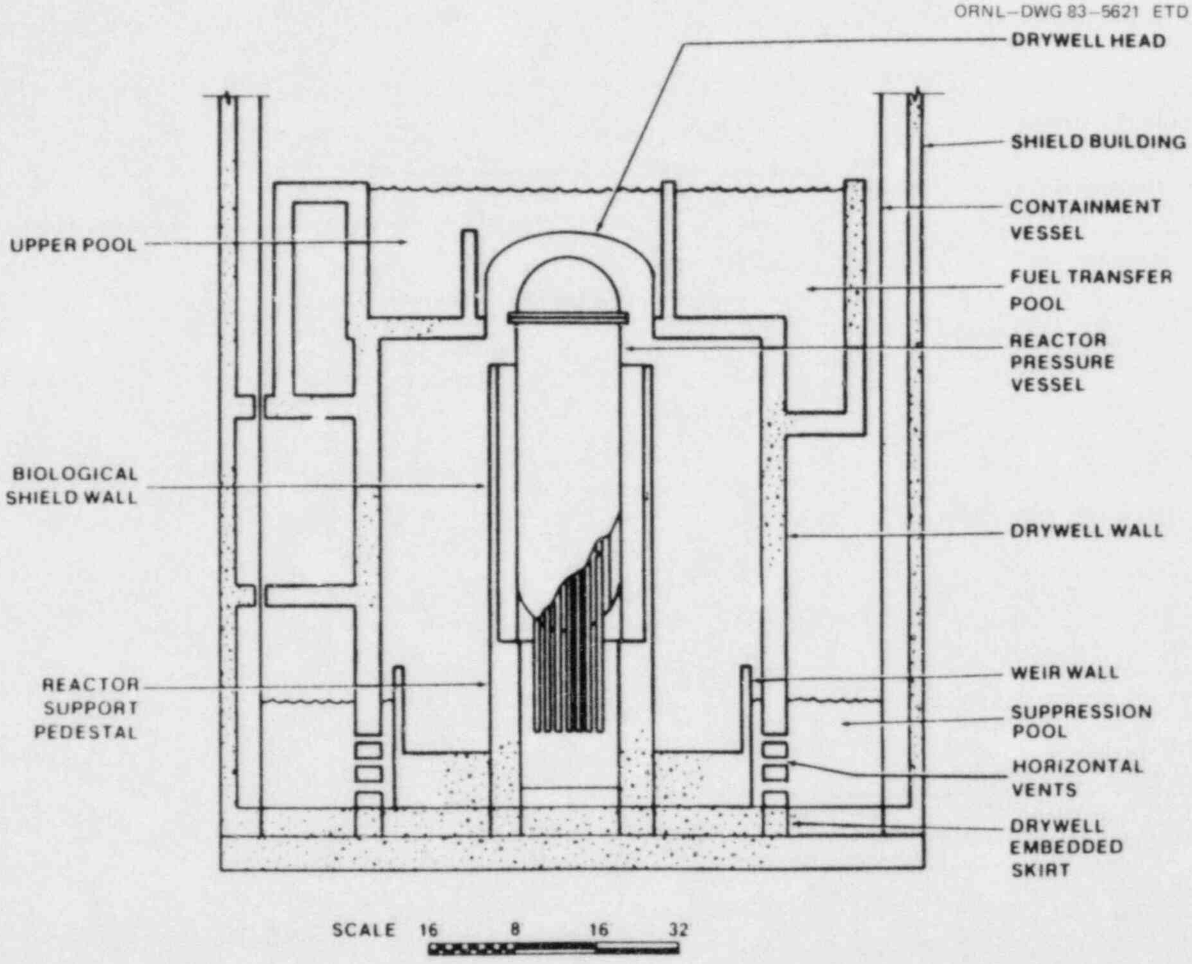

Fig. 3.8. Standard MK III primary containment structures.

ORNL-DWG 83-5610 ETD

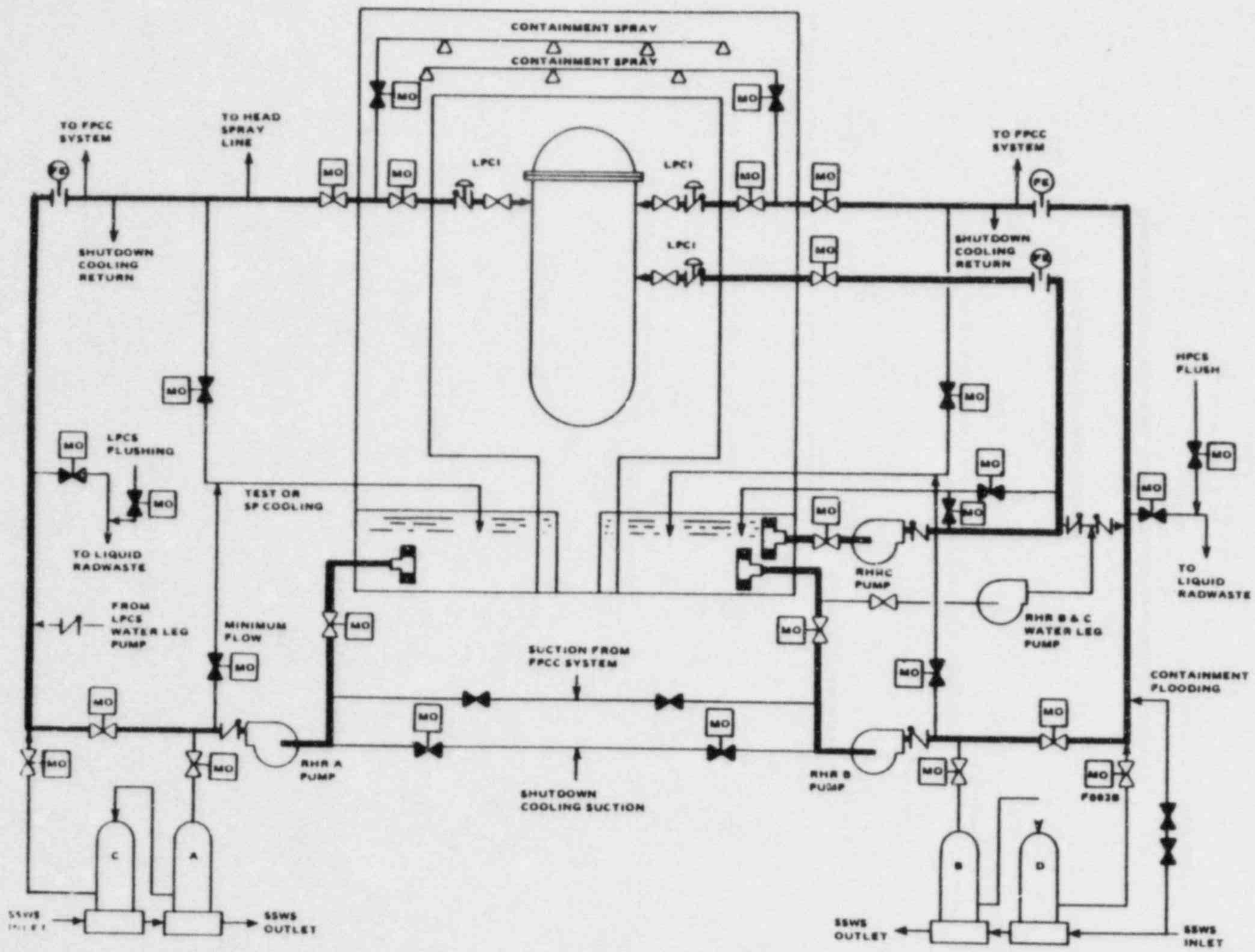

Fig. 3.9. BWR-6 residual heat removal system.

ORNL–DWG 83–5623 ETD

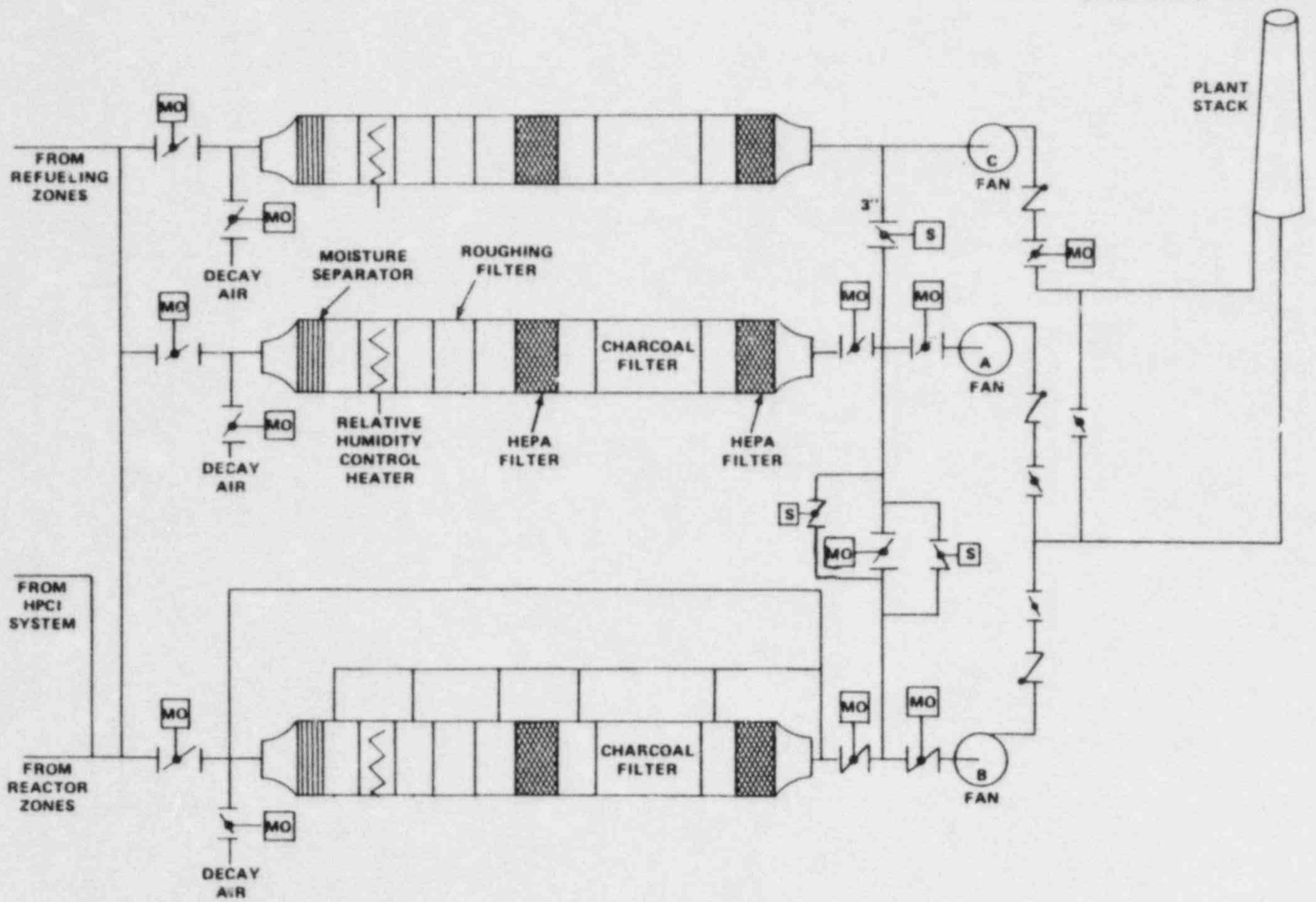

Fig. 3.10. MK I standby gas treatment system.

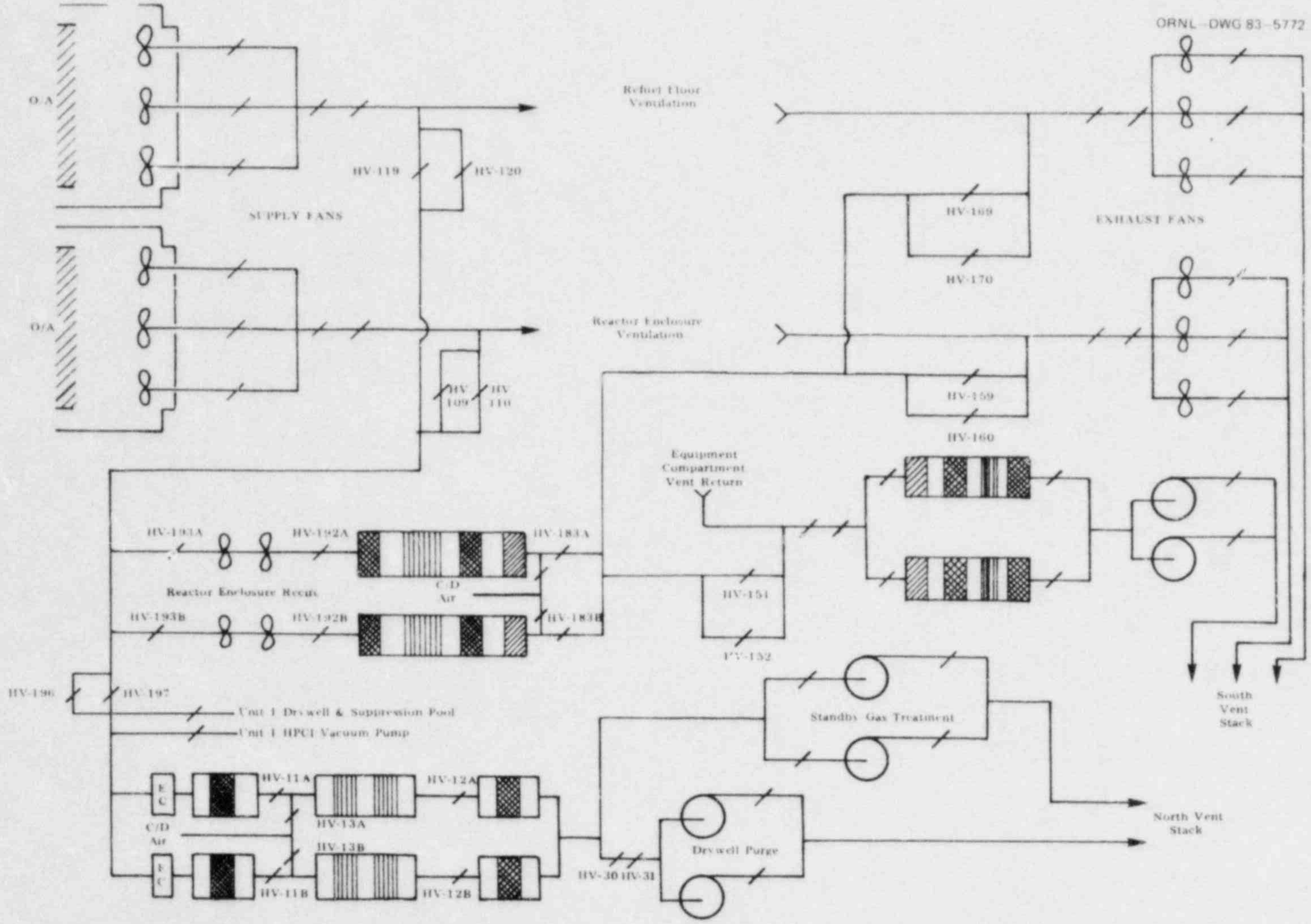

Fig. 3.11. Limerick standby gas treatment system.

ORNL-DWG 83-5622 ETD

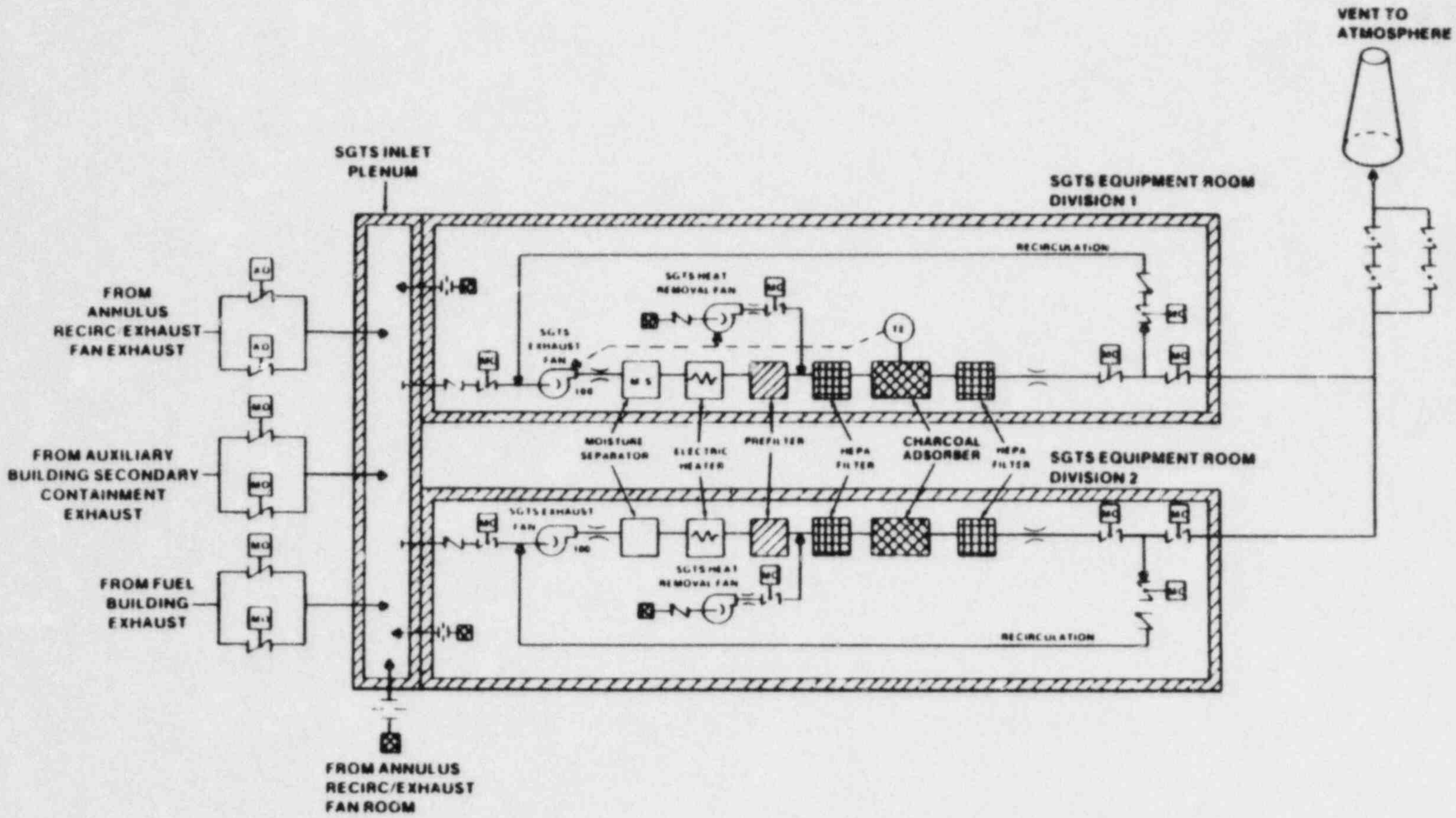

Fig. 3.12. Standard MK III standby gas treatment system.

ORNL-DWG 83-5624 ETD

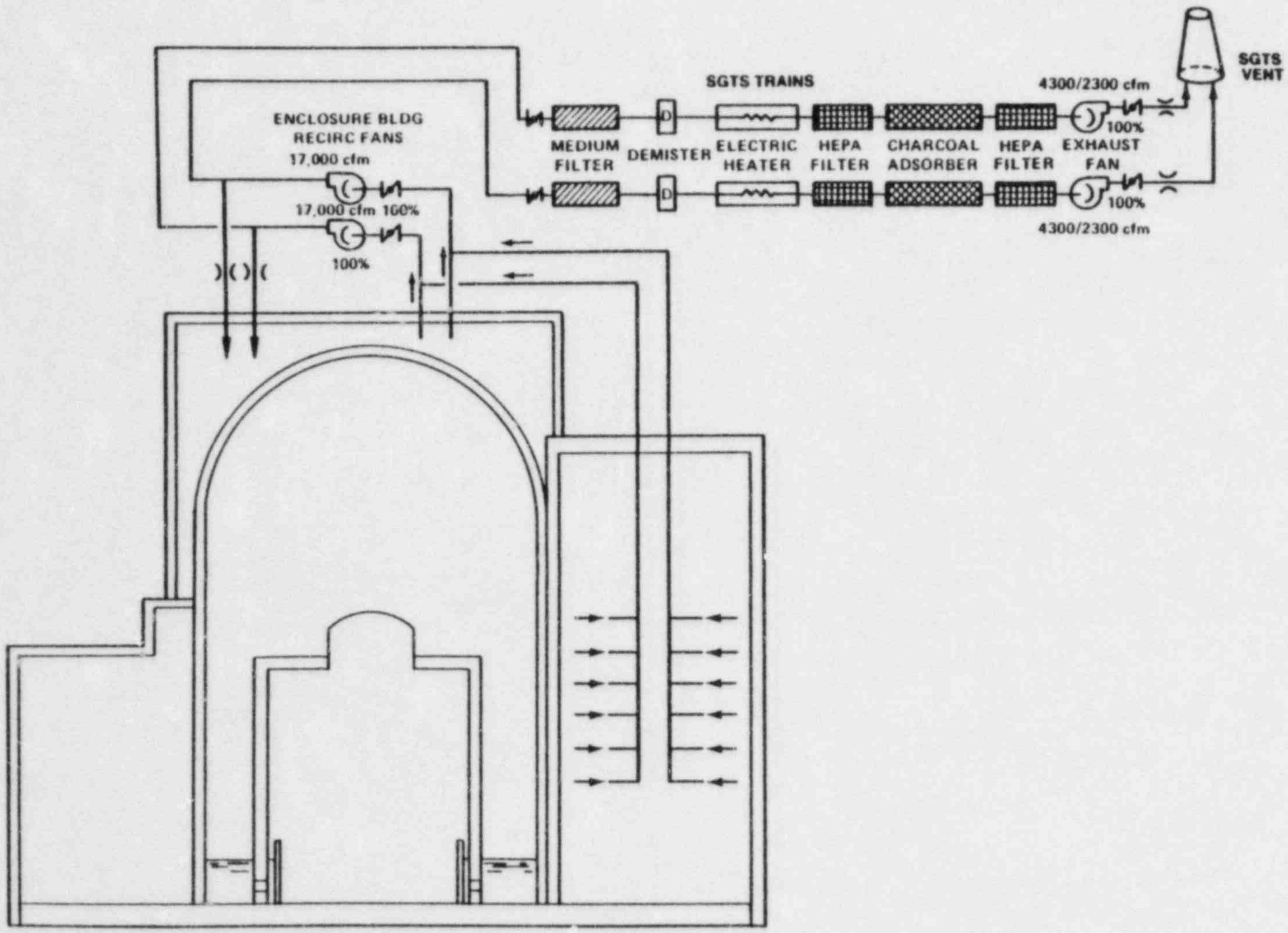

Fig. 3.13. Alternative MK III standby gas treatment system.

ORNL-DWG 83-5625 ETD

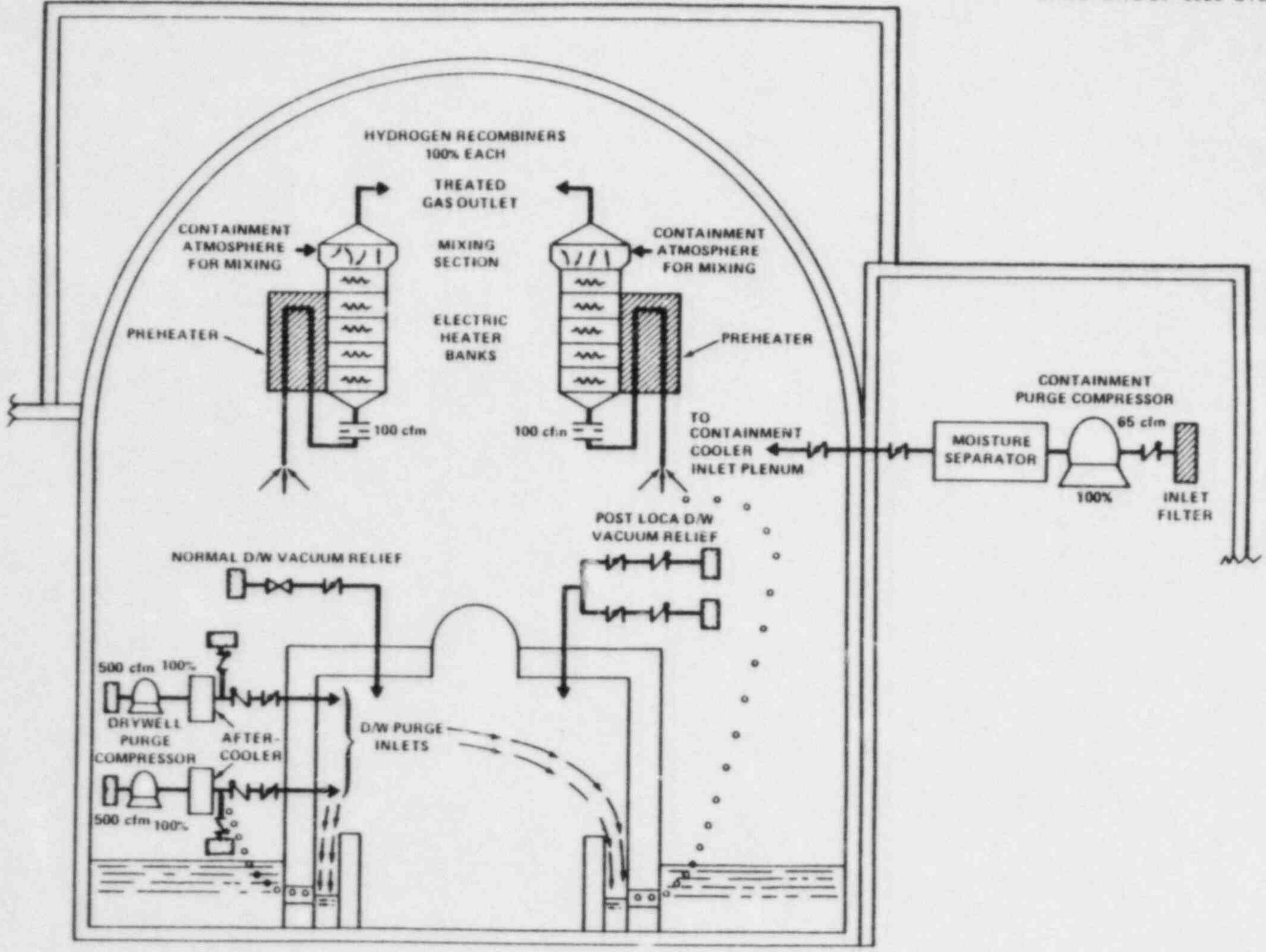

Fig. 3.14. Alternative (Grand Gulf) MK III combustible gas control and vacuum relief systems.

ORNL-DWG 83-5627 ETD

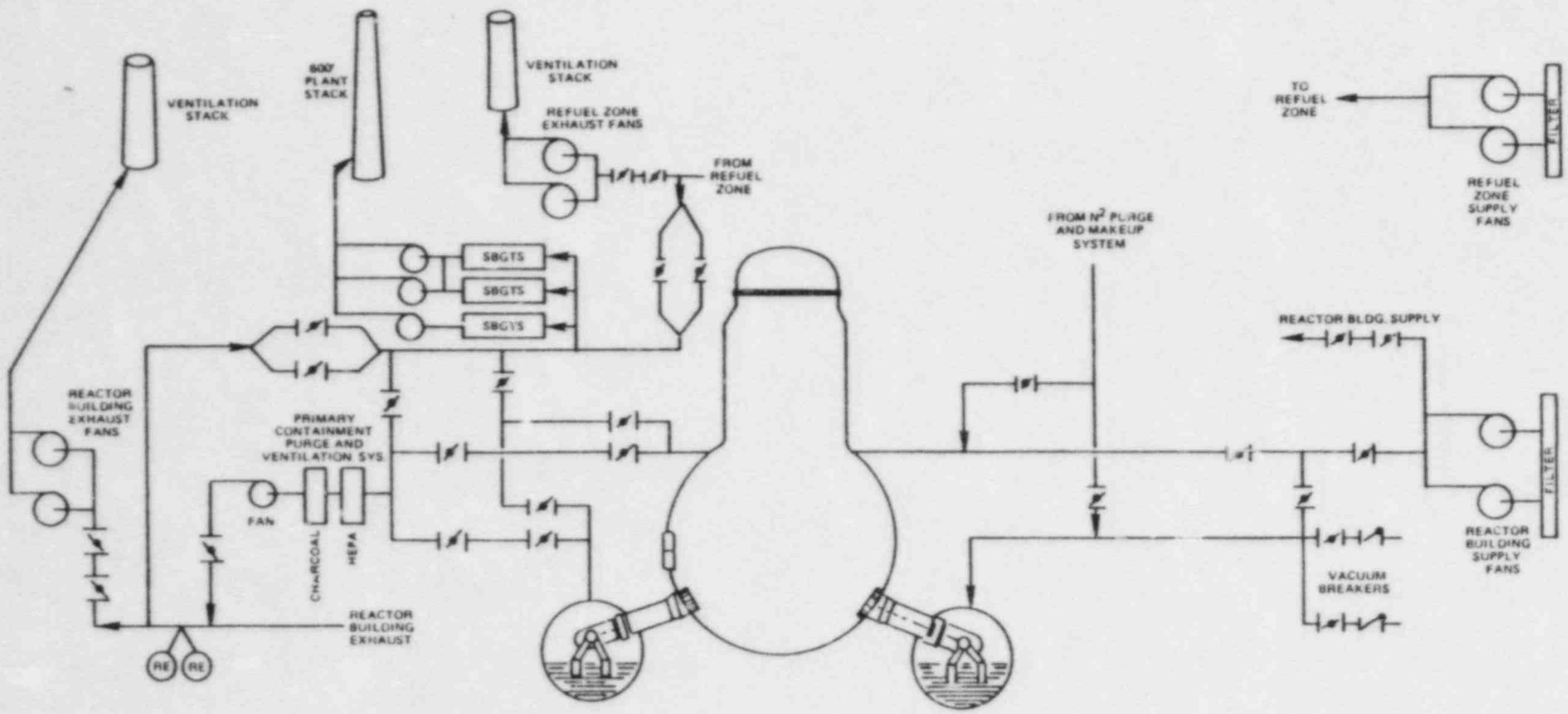

Fig. 3.15. MK I containment ventilation system.

ORNL-DWG 83-5771

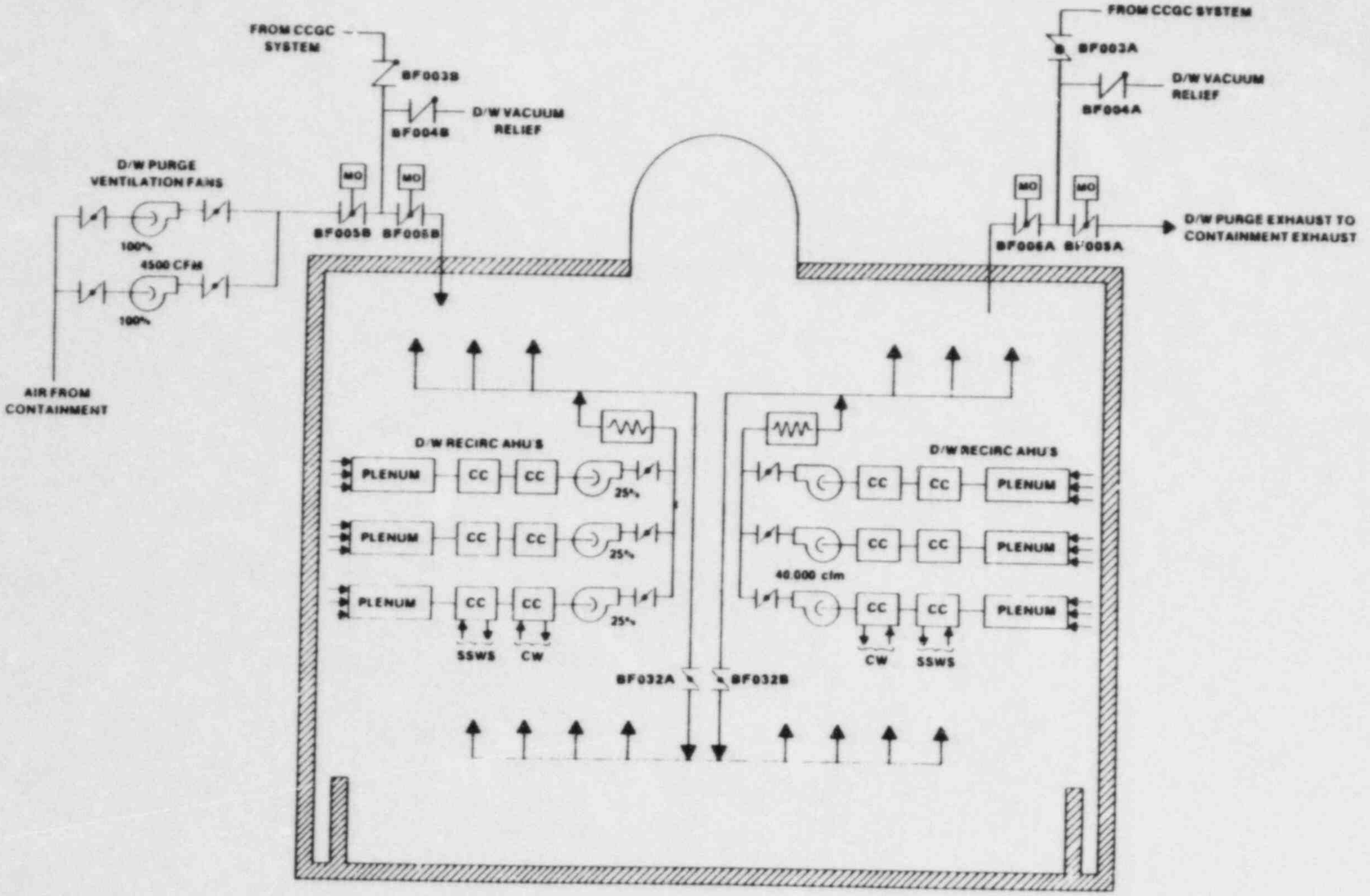

Fig. 3.16. Standard MK III drywell recirculation and purge ventilation systems.

ORNL-DWG 83-5770

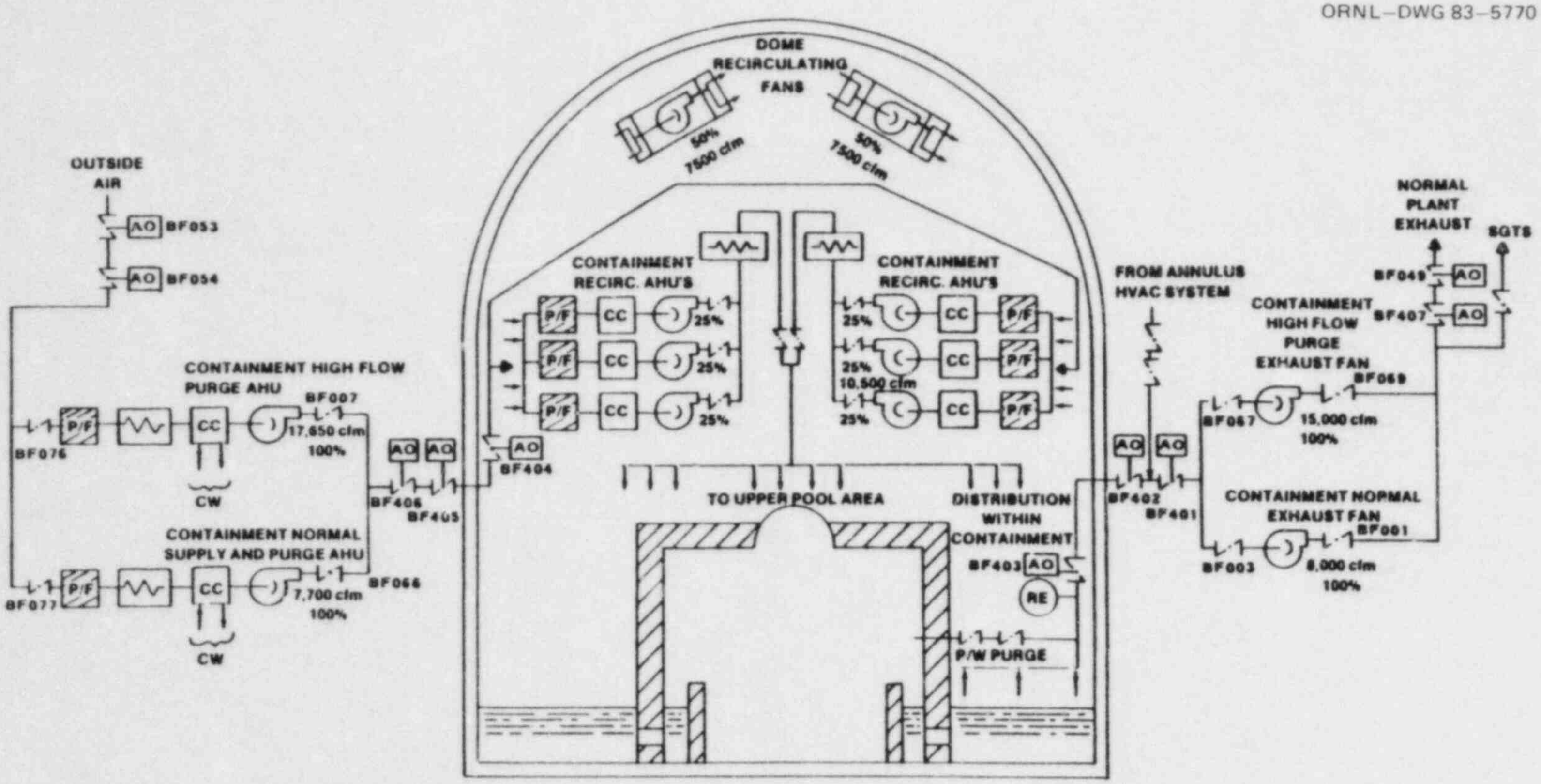

Fig. 3.17. Standard MK III containment normal ventilation, high flow purge, and recirculation systems.

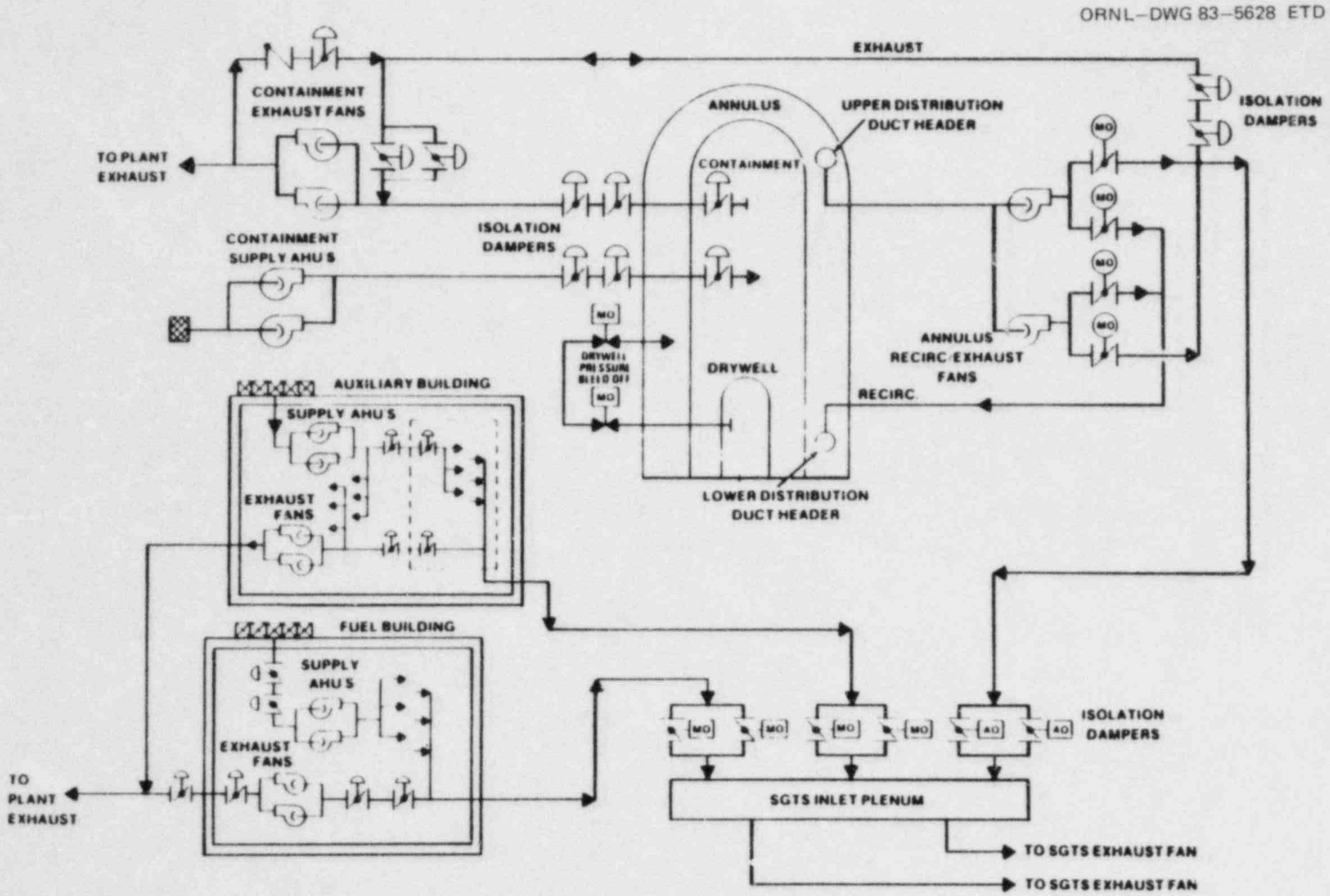

Fig. 3.18. Standard MK III secondary containment HVAC systems.

ORNL–DWG 83–5626 ETD

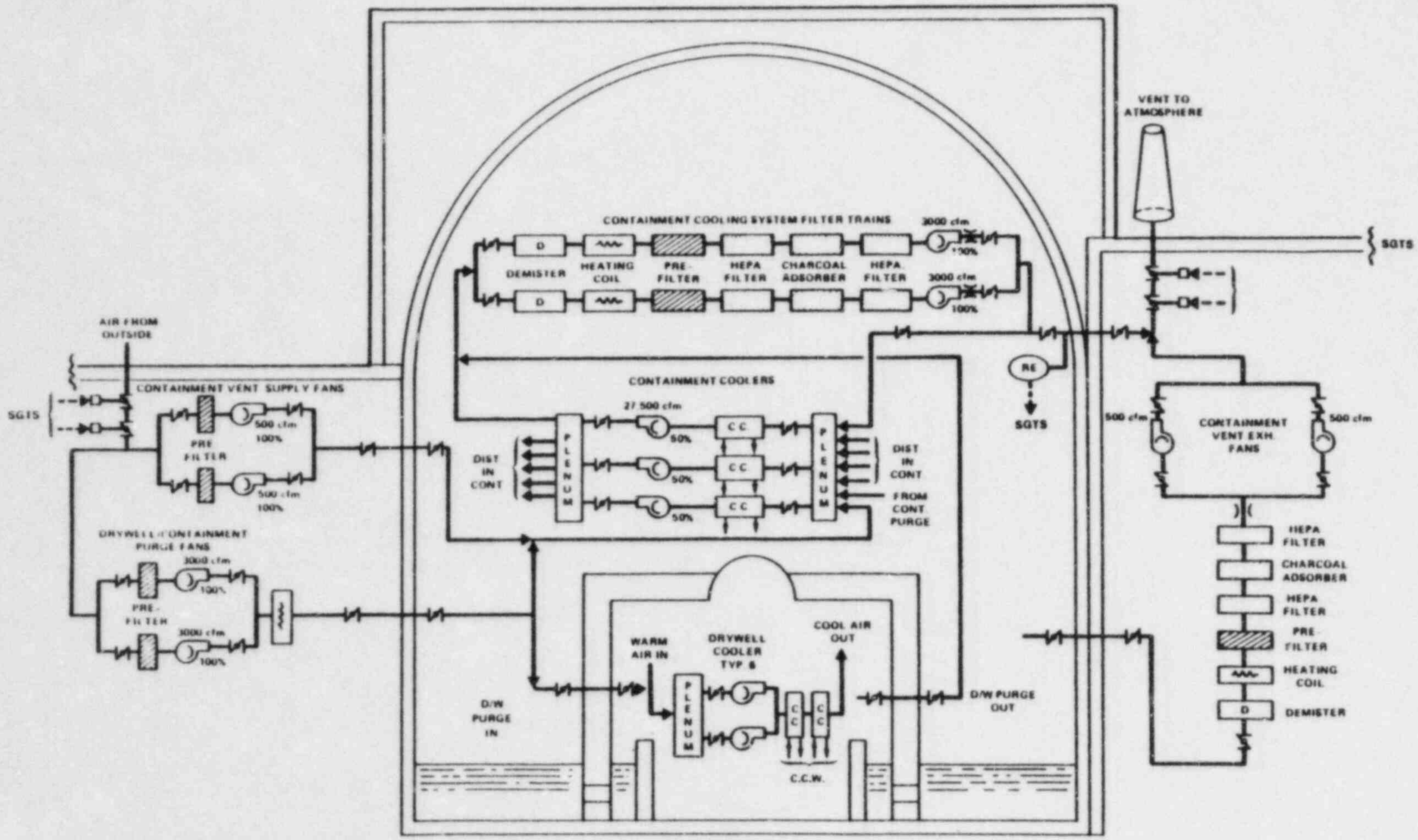

Fig. 3.19. Alternative MK III containment and drywell ventilation, filtration, cooling, and purge systems.

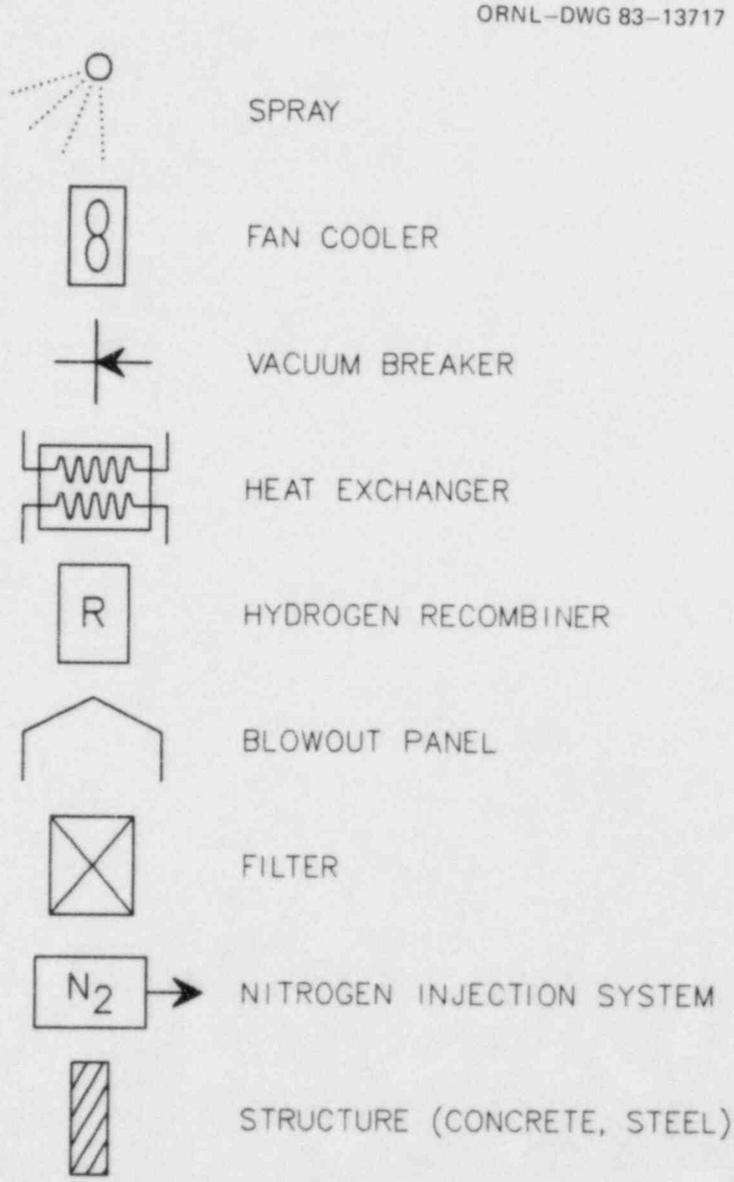

Fig. 3.20. Symbols employed in Figs. 3.21-3.24.

ORNL-DWG 83-12783

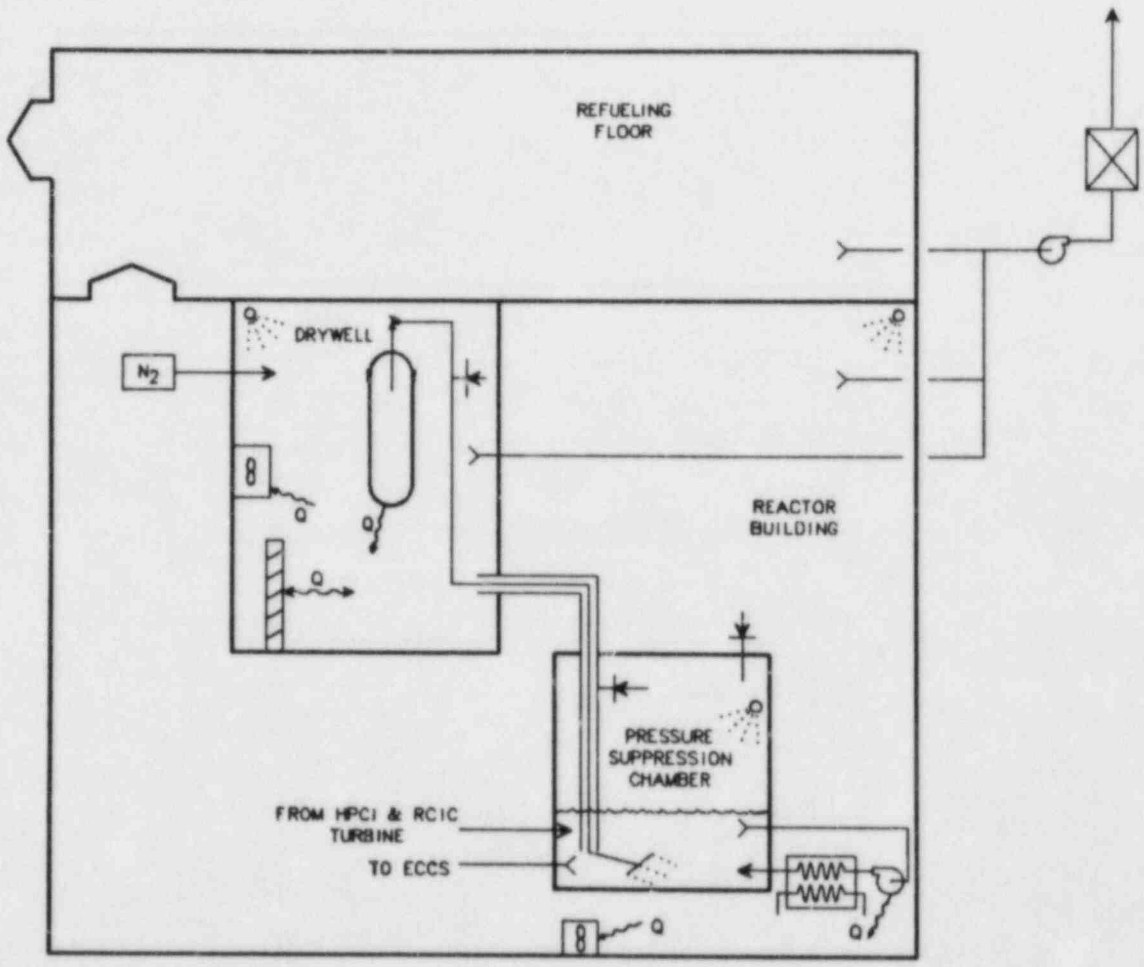

Fig. 3.21. Generic MK I containment systems and structures.

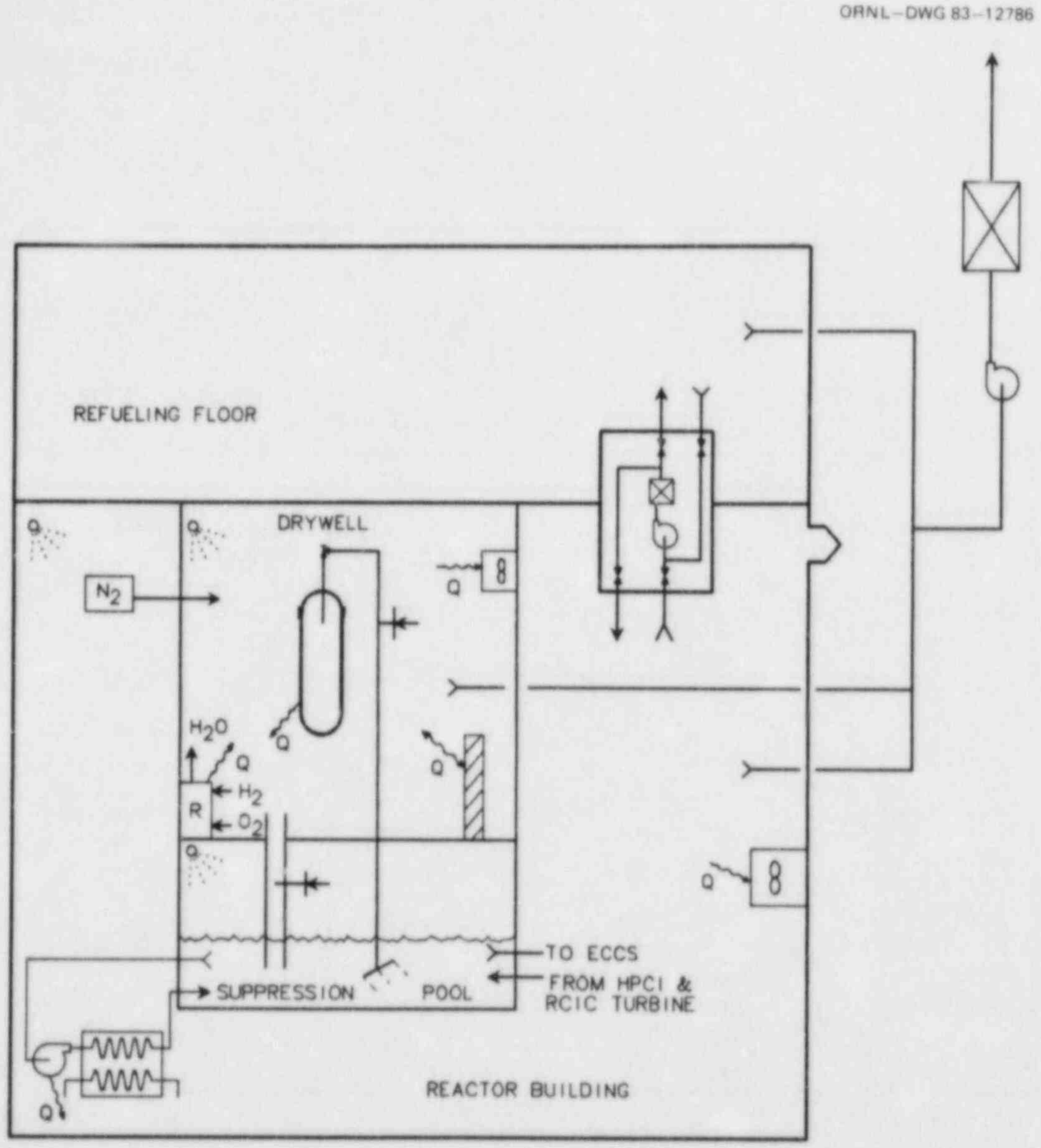

Fig. 3.22. Generic MK II containment systems and structures.

ORNL-DWG 83-12784

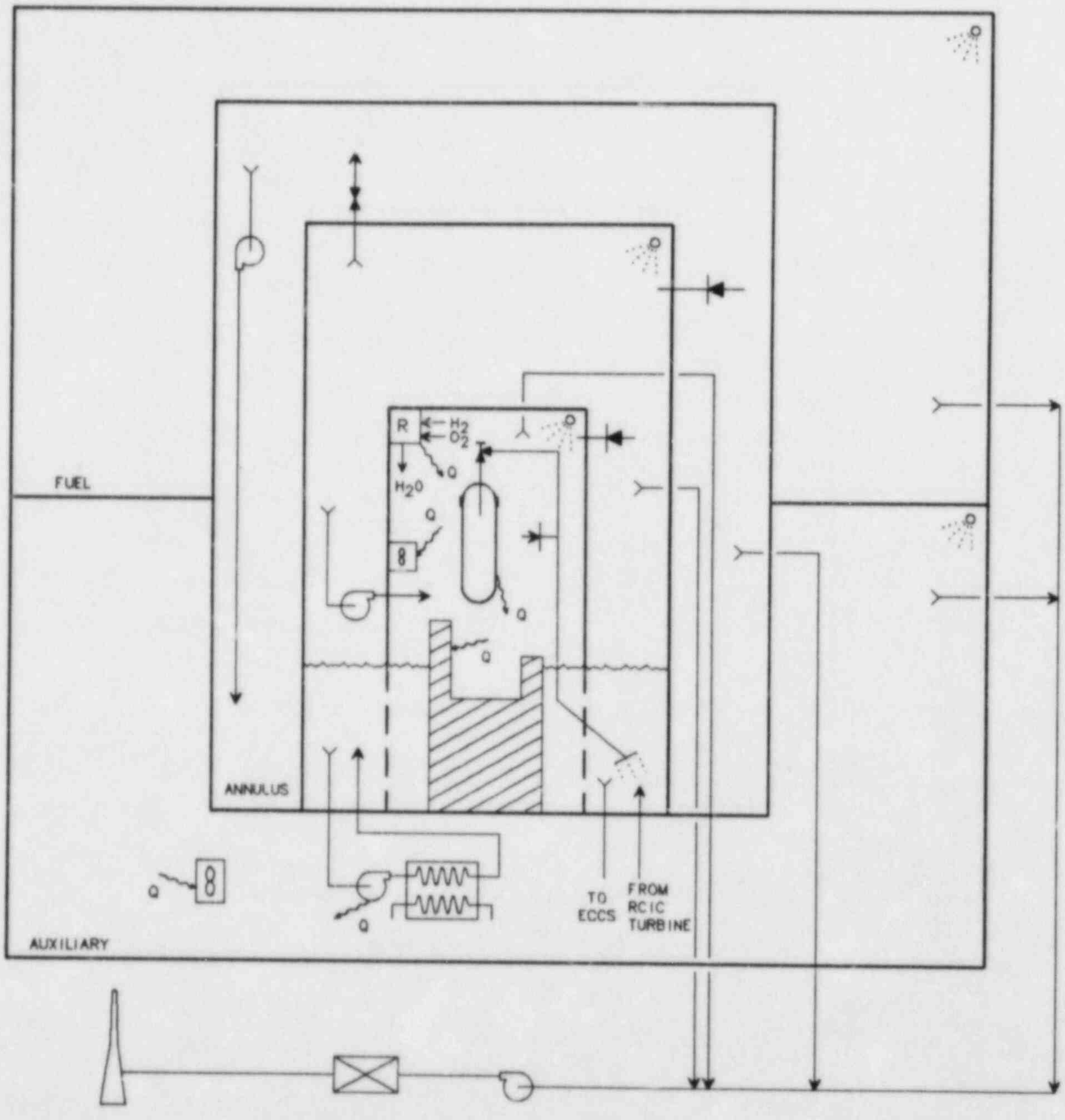

Fig. 3.23. Standard MK III generic containment systems and structures.

ORNL-DWG 83-12785

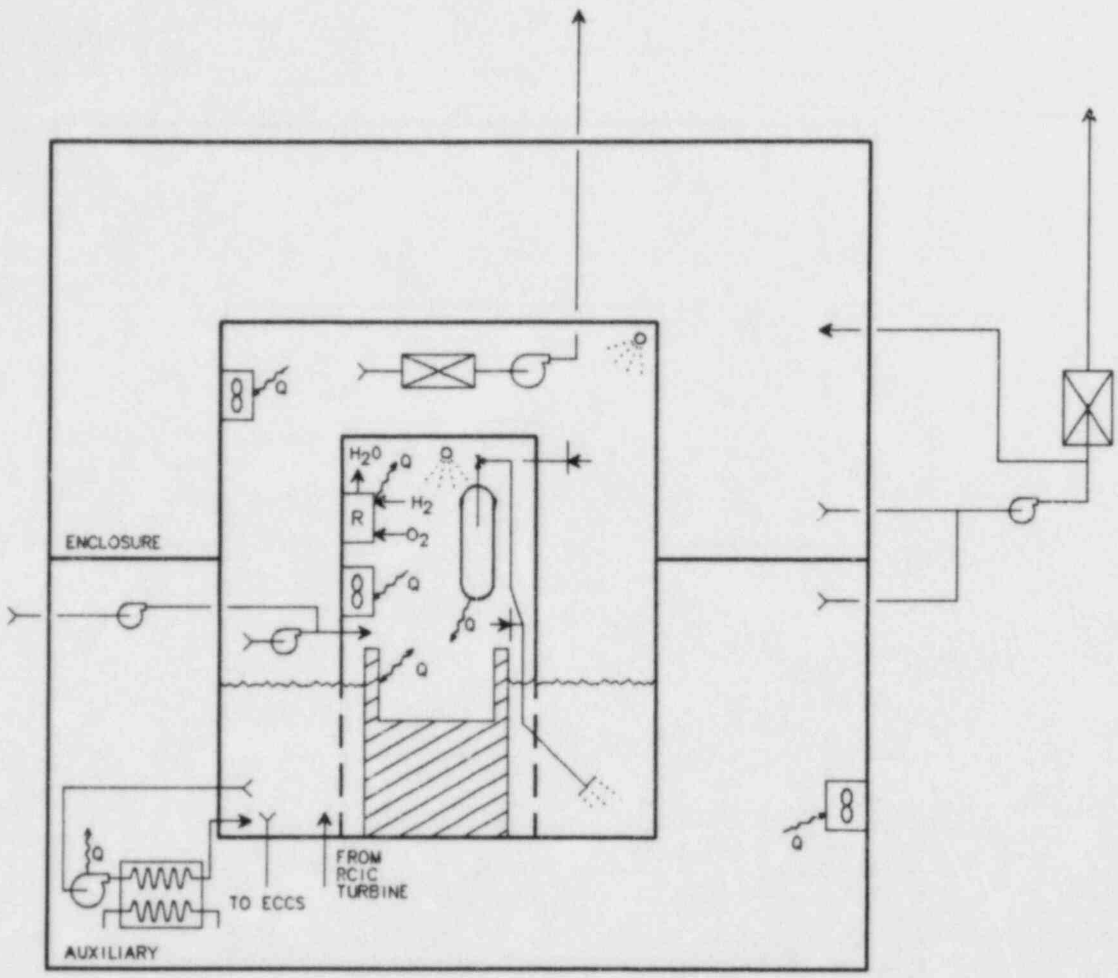

Fig. 3.24. Alternative MK III generic systems and structures.

4. BWR PRA RESULTS SURVEY — DOMINANT SEQUENCES

Table 4.1 contains a listing of domestic BWRs for which probabilistic risk assessments (PRAs) have been completed or are currently underway.[4.1] The majority of these studies have employed the general methodology and in most cases the models and codes (i.e., MARCH, CORRAL, CRAC) that were developed during and as a result of the Reactor Safety Study (RSS).[4.2] While these PRAs have been performed over an 11-year period, the results are generally consistent and indicate that the four "risk dominating" sequences are: (1) transients coupled with failure to provide makeup water to the reactor, (2) transients accompanied by loss of containment (suppression pool) heat removal capability, (3) transients coupled with failure to achieve reactor subcriticality, and (4) loss of coolant (pipe break) accidents. Although these four accidents are consistently ranked as the major or risk dominating sequences, their relative order and absolute core damage probabilities vary widely from study to study.

Table 4.2 is a summary listing of the RSS BWR accident sequence symbols. Following this terminology, the four sequences named above are designated as TQUV, TW, TC, and A or S transients.

Chapter 4 References

4.1 V. Joksimovich, *Insights from on-going PRA Studies*, IEEE Transactions on Nuclear Science, Vol. NS 29, No. 1, February 1982.

4.2 *Reactor Safety Study*, WASH-1400, NUREG-75/014, U.S. Nuclear Regulatory Commission (1975).

Table 4.1. Domestic BWR PRAs

Plant	Reactor type	Containment type
Peach Bottom 2	4	MK I
Oyster Creek	2	MK I
Millstone 1	3	MK I
Browns Ferry	4	MK I
LaSalle 1	5	MK II
Susquehanna	4	MK II
Shoreham	4	MK II
Limerick	4	MK II
Grand Gulf	6	MK III-A
GESSAR	6	MK III

Table 4.2. BWR accident sequence symbols

A	Rupture of reactor coolant boundary with an equivalent diameter of greater than 6 in.
B	Failure of electric power to ESFs.
C	Failure of the reactor protection system.
D	Failure of vapor suppression.
E	Failure of emergency core cooling injection.
F	Failure of emergency core cooling functionability.
G	Failure of containment isolation to limit leakage to less than 100 volume %/d.
H	Failure of core spray recirculation system.
I	Failure of low pressure recirculation system.
J	Failure of high pressure service water system.
M	Failure of safety/relief valves to open.
P	Failure of safety/relief valves to reclose after opening.
Q	Failure of normal feedwater system to provide core make-up water.
S_1	Small pipe break with an equivalent diameter of ~2 to 6 in.
S_2	Small pipe break with an equivalent diameter of ~0.5 to 2 in.
T	Transient event.
U	Failure of HPCI or RCIC to provide core make-up water.
V	Failure of low pressure ECCS to provide core make-up water.
W	Failure to remove residual core heat.
α	Containment failure due to steam explosion in vessel.
β	Containment failure due to steam explosion in containment.
γ	Containment failure due to overpressure - release through reactor building.
γ'	Containment failure due to overpressure - release direct to atmosphere.
δ	Containment isolation failure in drywell.
ϵ	Containment isolation failure in wetwell.
ζ	Containment leakage greater than 2400 volume %/d.
η	Reactor building isolation failure.
θ	Standby gas treatment system failure.

5. BWR SEVERE ACCIDENT MODELING NEEDS — GENERAL CONSIDERATIONS

The purpose of this report is to identify the BWR systems, structures, and accident phenomena that should be modeled in order to perform a realistic evaluation of severe accidents in BWRs. The fundamental goal of severe accident analysis is to determine the probability, timing, mode, and magnitude of releases of fission products to the environment. It is clear, therefore, that any BWR system, structure, or phenomena which has the capacity to significantly influence the probability, timing, mode, or magnitude of fission product release should be represented in future BWR severe accident analysis codes.

The appropriate structure of a severe accident analysis code should be determined by intercompartmental issues such as the number and type of flows entering and leaving a given compartment and the modes by which various compartments can communicate with each other. The overall size and sophistication of a code is determined by intracompartmental issues such as the level of detail employed in the modeling of the thermophysical phenomena which occur within a reactor vessel or a containment compartment. Additionally, it is apparent that, given the appropriate code structure, improvements to various intracompartment phenomenology models can be readily accommodated as our understanding of these phenomena improves. Accordingly, significant effort has been expended during this assessment to identify the various and energy mass and energy flows that might cross the reactor and containment system boundaries during different phases of a severe accident.

For purposes of discussion, all BWR plant designs can be represented by four compartments: reactor, drywell, wetwell (or containment building), and secondary containment. Table 5.1 lists each of these compartments, together with the various systems and structures that are present in or connect to each compartment. Realistic simulation of BWR severe accidents will necessitate some treatment of each of the systems and structures in Table 5.1. Code developers should therefore incorporate representations of these systems and structures in future BWR severe accident analysis codes. The following chapters discuss the system and structure interactions and accident phenomena which occur in each of these four compartments during a core melt accident.

Table 5.1. BWR systems and structures summary

Compartment	Interacting systems	Interacting structures
Reactor	MSIV SRV HPCI HPCS RCIC LPCI LPCS SLC RWCU FW CRDHS Recirculation Head spray	Vessel walls Core shroud Shroud head Standpipes Steam separators Steam dryers Fuel rods Zr canisters Control rods Core plate Top guide Control rod guide tubes Stub tubes Instrument tubes Lower head Head drain
Drywell	Atmospheric (fan) coolers Recombiners Sprays SGTS Vacuum breakers Compressors (outlet) Purge venting Nitrogen injection	Reactor vessel walls Vents Drywell walls Reactor pedestal Pedestal cavity Penetrations
Wetwell/ containment building	RHR Sprays SRVs Vacuum breakers Atmospheric fan coolers HEPA/charcoal filters Purge venting SGTS ECCS turbine ECCS pump suction Compressor suction Pool makeup	Vents Vacuum breakers Walls Drywell vents Drywell floor melt-through Misc. equipment (heat loads) Penetrations

Table 5.1. (continued)

Compartment	Interacting systems	Interacting structures
Secondary Containment	SGTS Fire Protection System sprays Vacuum breakers ECCE (heat loads) ECCE Room Coolers HVAC systems	Blowout panels Walls, ceiling, floor

6. SIMPLIFIED BWR CORE MELTDOWN SCENARIO

6.1 Introduction

The discussion of in-vessel BWR severe accident phenomena and modeling needs will be enhanced by first describing a simplified BWR core melt scenario in which the reactor is subcritical, isolated, and no injection is available as makeup to the reactor vessel (this is the TQUV accident with no operator action and no CRD hydraulic system flow). The first scenario description presented here is based on calculations performed for the Browns Ferry Unit 1 reactor with the ORNL MARCH 1.1B code, which has been extensively modified to represent BWRs in a more realistic manner.[6.1,6.2] *However, due to the many remaining MARCH 1.1B BWR modeling limitations, the reader is cautioned that this scenario should be viewed only as an approximation to the sequence of events which might actually occur during such an accident.* For analytical purposes the core has been nodalized into a 10 radial zone, 10 axial zone (Fig. 6.1) format.

6.2 Sequence of Events

Immediately following accident initiation, the SRVs begin actuating in the safety mode (automatic operation with 50 psi primary system blowdown with each actuation), resulting in a reduction in reactor water inventory. Since the reactor core is initially covered with several feet of water, approximately 30 min pass before the core begins to uncover. Due to steadily decreasing decay heat levels, the interval between SRV actuations continually increases such that at the time of core uncovery the SRVs are actuating once every 3 to 5 min. Each SRV pop results in an immediate but brief vessel water level swell of as much as 1.8 m (6 ft) and a substantial flow of steam and water through the core, upper vessel internals, and the open SRVs. During the period between SRV pops, the vessel slowly repressurizes to the SRV set point, and there is relatively little flow through the core.*

Core structural temperatures begin rising immediately following core uncovery. Fuel pin, canister wall, and control blade surface temperatures in the uncovered portions of the 8th radial and 9th axial zone are predicted to reach 1360 K (2000°F) about 70 min after accident initiation. At these temperatures, oxidation of the zircaloy fuel cladding and canister accelerates, accompanied by heat and hydrogen generation. Although less than 0.6 m (2 ft) of the core is covered at this time, the continuing SRV actuations are producing momentary water level swells which briefly cover significant portions of the core. The structural integrity of the core has not yet been challenged, and no fission

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*During the actual accident there is, or course, natural circulation of water through the core during the earlier stages of the accident when the water level is near the top of the steam separator outlets. This phenomenon is not modeled in the MARCH code.

products have escaped the fuel. The temperatures of the reactor upper internals (shroud head, separators, dryers) are predicted to be ~590 K (600°F). The highest fuel temperature in the outer zone of assemblies is ~870 K (1100°F).

Heatup of the core structures continues until ~78 min into the accident when the stainless steel control blade sheaths and control rod cladding begins melting in the 8th radial, 8th axial region of the core (Fig. 6.1). At this time the maximum fuel pin and canister wall temperatures are near 1644 K (2500°F). Less than 2% of the total fuel pin cladding and 1% of the total canister surface area are oxidized at this time. The highest fuel temperature in the outer zone of fuel assemblies is ~920 K (1200°F). Although the control blades in the 8th radial zone have begun to melt, all fuel pins are intact and there has been no significant release of fission products from the fuel. The SRV with the lowest set pressure is cycling at ~7 min intervals.

By ~90 min into the transient (Fig. 6.2), fuel and canister wall temperatures in several zones have risen to the point where melting of the zircaloy fuel cladding and canisters is expected to occur [2150 K (~3400°F)]. Less than 4% of the total fuel cladding is oxidized at this time. These low oxidation levels are a result of steam starvation of the $Zr-H_2O$ reaction which is, in turn, due to the long intervals between SRV pops. During these intervals, there is relatively little steam flow across the fuel pins. Fuel clad melting will result in prompt release of some fission products (primarily noble gases) to the reactor interior, followed somewhat later by release of the volatile fission products such as cesium and iodine as the fuel temperatures continue to rise. "Burst" type cladding failures are improbable since the reactor pressure is significantly higher than the interior fuel pin pressure. Continued melting and vaporization of core structural materials generate substantial quantities of aerosols within the reactor vessel. Much of the fission product and aerosol material is being deposited on internal reactor structures (separators, dryers, etc.) which are predicted to be significantly cooler than the core [i.e., ~588 K (600°F)]. Fission products which escape during SRV actuations are deposited in the pressure suppression pool. The region of molten cladding, canister, and control blades continues to grow until 95 min after the start of the accident, at which time fuel melting begins (2nd radial, 8th axial zone, Fig. 6.3). Fuel pin melting is accompanied by rapid release of volatile fission products. The region of molten fuel, cladding, canister, and control rod material is gradually relocating downward via a melt/flow/refreeze cycle.

Figure 6.4 is a schematic representation of the core status at 110 min into the accident. The entire outer zone of fuel assemblies is still intact, containing its full inventory of fission products. In addition, it should be noted that only a small fraction of the fuel in the core is molten. Upper vessel internal structural temperatures are typically predicted to be less than 755 K (900°F), and the lowest set SRV is cycling every 15 to 20 min. Due to the low vessel structural temperatures and the extended period between SRV pops, significant opportunities exist for deposition and plateout of fission products within the vessel.

Figure 6.5 is a schematic representation of the status of the core at 140 min into the accident as predicted by MARCH 1.1B. Approximately 67% of the fuel is predicted to be molten at this time, although the fuel pins and canisters in the outer zone of assemblies have not yet begun to melt.

Since MARCH does not model axial conduction in the fuel structures or heat transfer from assemblies in partially uncovered nodes to the surrounding water, the water level is predicted to stay within the first axial node (bottom few inches of the core) for the entire period between 115 and 145 min into the accident. Since little steam is being produced, a single SRV is predicted to cycle very infrequently during this period.

The entire core (melted and unmelted assemblies) is predicted to slump into the core plate at 145 min, when the core is 75% molten.* The temperatures of the upper vessel internals range between 560 and 1030 K (550 and 1400°F) just prior to core slump. The core plate is predicted to melt through in less than 1 min, allowing the core debris to attack the control rod guide tubes. The guide tubes are predicted to melt within 1 min, allowing the debris to begin attacking the bottom head at 147 min into the accident. All of the water in the bottom of the vessel is predicted to flash in the 2-min period as the core slumps into the bottom plenum. All the SRVs are predicted to open at this time to relieve the associated pressure spike. The bottom head of the reactor vessel is pre dicted to fail at 211 min into the accident. Following head failure, the entire mass of core debris is immediately transported into the reactor pedestal cavity.

6.3 Alternative Scenario

Section 6.2 presented a simplified BWR TQUV scenario (without operator action or CRD hydraulic system injection) as predicted by the ORNL MARCH 1.1B computer code. Although this code has been extensively modified to improve its BWR simulation capabilties, it still contains many modeling simplifications which are inappropriate for BWRs. The purpose of this section is to provide an alternate description of the previously described scenario that attempts to account for known BWR modeling deficiencies in the MARCH code. The reader should recognize that the scenario presented in this section is based on engineering judgment rather than computer code simulations.

The alternate scenario is identical to the scenario of Sect. 6.2 prior to the time that melting of the control blades begins. It is possible that melting of the control blade sheaths would begin later than predicted by MARCH, since radial radiation and axial heat conduction

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*75% criteria is user input to MARCH.

mechanisms may tend to smooth axial and radial core temperature profiles. Eventually, however, the temperature of the control blade sheaths in some areas will reach the melting point of the stainless steel. The molten stainless steel material will relocate downward and refreeze. Once the control rod sheath has melted, the interior stainless steel clad boron carbide tubes would undergo thermal attack and soon melt. The boron carbide powder might react with the steam, releasing hydrogen, carbon monoxide, and heat. Those portions of the control rod above the molten zone might fall into the molten zone, forming temporary flow blockages in the interstitial region.

As the control rods continue to melt, the downward relocation of control rod material will result in increased radiative coupling of the fuel assembly canisters which were originally separated by the control blade. This will cause further flattening of the radial core temperature profiles. The degree of enhancement of the canister-to-canister radiative heat transfer is highly uncertain since much depends on the rate of aerosol generation during control rod melting. If significant quantities of aerosols are generated, adjacent canisters might radiate to the aerosols instead of each other. This phenomena of radiative aerosol heating is of importance throughout the entire accident since it adds great uncertainty to both the core heatup and fission product transport evaluations.

Melting of the control rod sheaths will generally be followed by degradation of fuel assembly canisters and cladding in the near vicinity. Figure 6.6 is a graphic representation of the core at this early stage of the accident. Loss of structural integrity of the canister and cladding can be caused either by melting or shattering of oxidized material. For cases in which one or more SRVs are actuating frequently or there is a break in the primary system boundary (to provide quenching and steam for the $Zr-H_2O$ reaction), it is possible that cladding and canister material could shatter prior to melting. Portions of the fuel assembly canisters above the melted or shatter zones might fall into these zones. Failure of the fuel cladding, whether due to melting or shattering, would result in rearrangement of the UO_2 fuel pellets. It is probable that localized pockets of debris would exist in the upper 1/4 to 1/3 of the core. At this time, significant amounts of aerosols would be generated by the melting of the steel and zircaloy. Occasional SRV actuations would result in levitation of some of the smaller debris particles. The debris would consist of solid UO_2 fuel pellets, molten and solid slugs of zircaloy and steel mixtures, pieces of unreacted zircaloy and steel, and pieces of various metal oxides.

The isolated pockets of core debris would gradually coalesce, resulting in the situation depicted in Fig. 6.7. At this stage of the accident, a large debris bed would be located in the upper half of the core. The central region of the debris bed would be molten UO_2. Copious quantities of aerosols would be generated. It is possible that these aerosols would hinder radiation heat transfer from the top of the debris bed to the shroud head. If the aerosols and shroud head do become heated by direct radiation, they might give off volatile fission products that had earlier plated out or been absorbed onto their surfaces.

The debris bed would continue to grow radially and to relocate downward. Axial heat conduction in the canisters and fuel assemblies would result in significant additions of heat to the water in the lower core, which would, in turn, provide additional steam to feed $Zr-H_2O$ reactions in the debris bed.

Eventually the debris bed would penetrate the outer zone of assemblies (Fig. 6.8). The distance between the outer zone of assemblies and the inner surface of the core shroud is ~25 cm (10 in.). Core debris would fall into this annulus and collect on the core plate. If water is standing above the bottom of the core, this debris would initially be quenched. This might, however, add significantly to the core steaming rate, thereby initiating additional hydrogen generation and SRV actuations during this phase of the accident. The molten region of the debris bed would increase in size due to lack of coolant flow in the interior of the debris bed. The core shroud and shroud head might become quite hot during this phase of the accident due to thermal radiation from the debris bed unless such radiation is impeded by aerosols. The vessel water level will continue to recede in the face of the advancing debris front.

Eventually the debris bed would contact the dry core plate and the plate would fail. It is also possible that the core plate would fail in the annulus region between the inside of the core shroud and the outer row of assemblies. The core shroud might melt through in some locations due to direct contact with the core debris. If melting of the core shroud is widespread, it is possible that the shroud would buckle, resulting in a realignment of the upper vessel internals. After failure of the core plate, core debris will begin penetrating into the lower plenum along both the inside and outside of the hollow control rod guide tubes. Depending on steam availability, additional amounts of hydrogen and heat might be generated due to oxidation of the stainless steel guide tube assemblies.

The attack of the control rod guide tubes and boiloff of water within the lower reactor vessel plenum would continue until the core debris contacts the control rod drive housings and stub tube assemblies which are welded to the bottom of the reactor vessel. Due to the curvature of the vessel bottom head and the variation in radial decay power distribution, this might first occur in a region near the outside of the control rod guide tube array. The stainless steel stub tube weld area would be subjected to three-dimensional thermal attack by the hot core debris, and would eventually fail. It is probable that melt-through and failure of the stub tube assemblies would occur prior to (and perhaps preclude) gross melt-through of the reactor bottom head.

Following melt-through of the vessel bottom head (or head penetrations), the reactor will begin depressurizing through the failure opening. The rate at which the reactor depressurizes will depend both on the initial failure size and the rate of ablation of the failure opening by the hot core debris. Thus, the amount of molten material in the lower head just prior to vessel failure will influence the rate of depressurization of the vessel. For coherent core melt scenarios, the vessel might be fully depressurized in a very few minutes.

It should be noted that, at the time of vessel failure, the core debris might contain unmelted fuel pellets. It is also possible that intact fuel assemblies would be standing in the reactor at this time. These are extremely important concerns since, subsequent to vessel failure, fission products released by cladding failure and fuel melting can be expelled directly into the drywell atmosphere, bypassing the fission product scrubbing features of the pressure suppression pool.

Chapter 6 References

6.1 S. R. Greene et al., *SBLOCA Outside Containment at Browns Ferry Unit One — Accident Sequence Analysis*, Appendices B and C, NUREG/CR-2672, Vol. 1, ORNL/TM-8119/V1, Oak Ridge National Laboratory (1982).

6.2 R. M. Harrington and L. J. Ott, *The Effect of Small-Capacity High Pressure Injection Systems on TQUV Sequences at Browns Ferry Unit One*, Appendix B, NUREG/CR-3179, ORNL/TM-8635, Oak Ridge National Laboratory (1983).

ORNL-DWG 83-13716

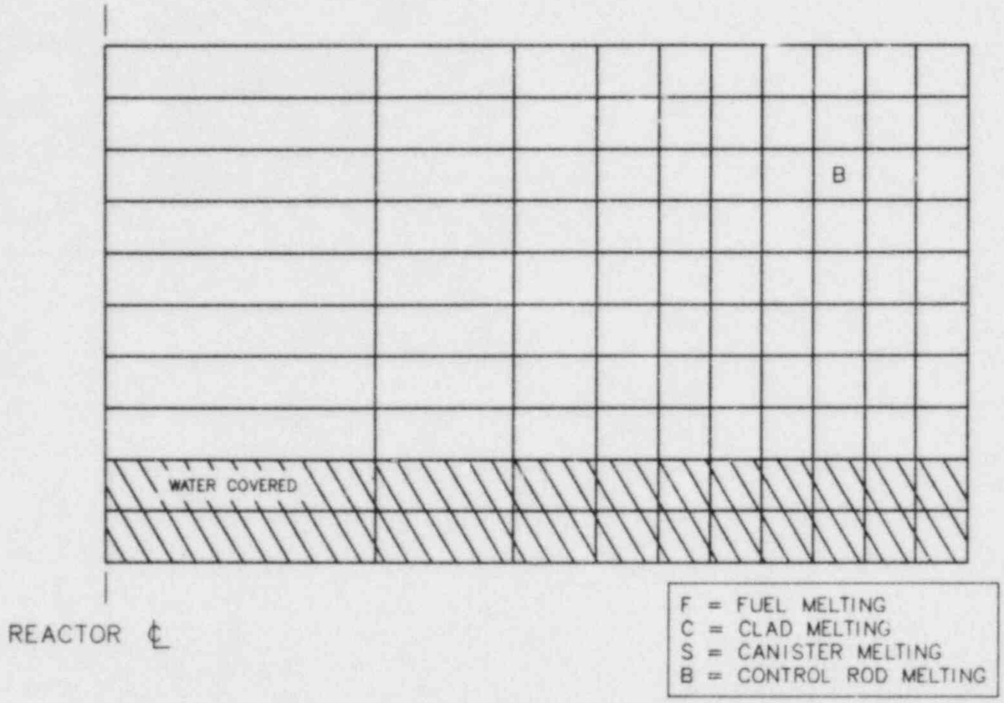

Fig. 6.1. Core status at 78 minutes into accident.

ORNL-DWG 83-13715

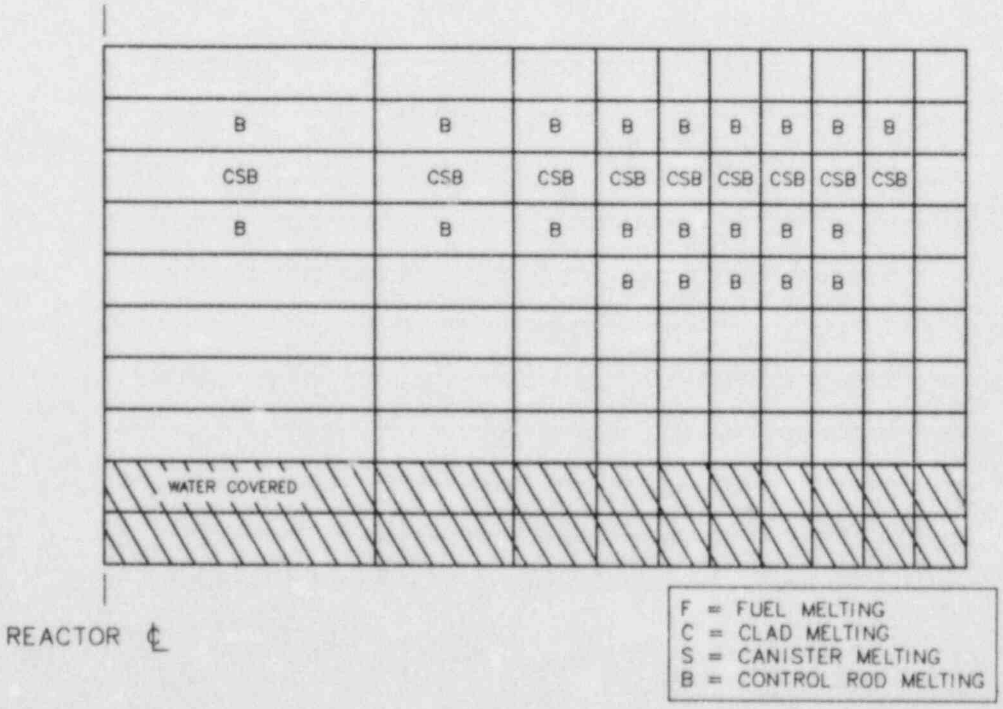

Fig. 6.2. Core status at 90 minutes into accident.

ORNL-DWG 83-13714

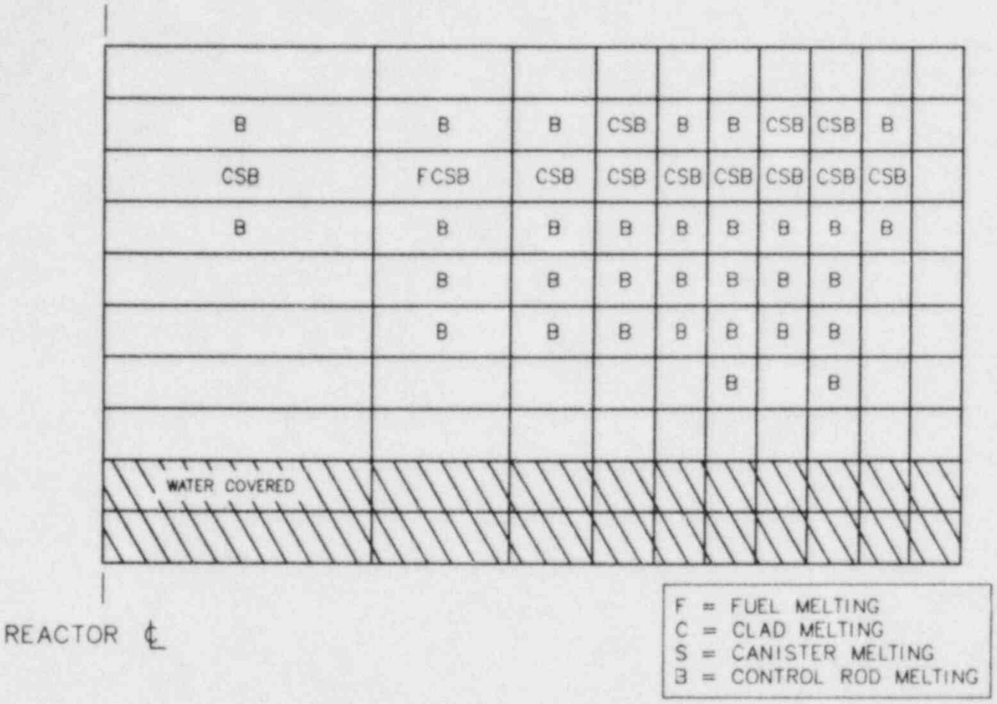

Fig. 6.3. Core status at 95 minutes into accident.

ORNL-DWG 83-13713

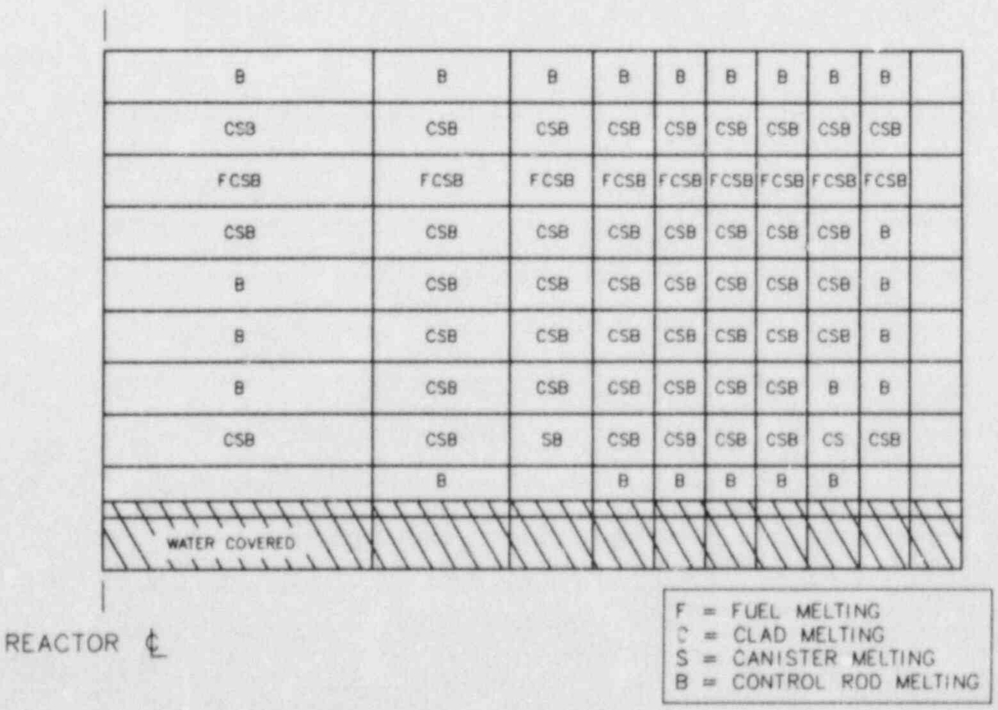

Fig. 6.4. Core status at 110 minutes into accident.

ORNL-DWG 83-13712

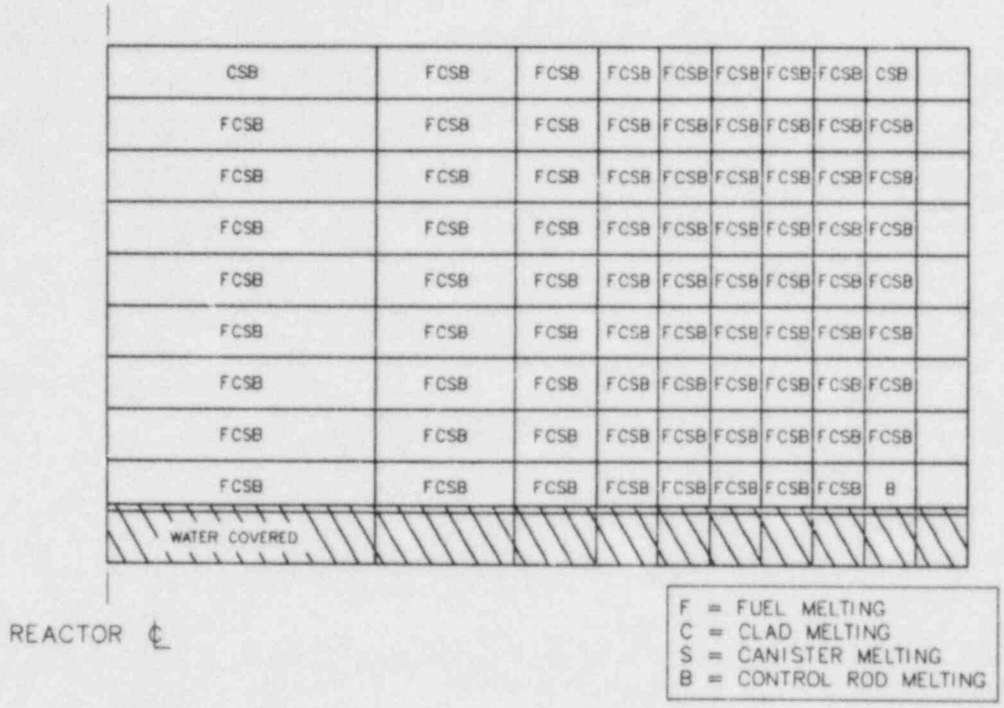

Fig. 6.5. Core status at 140 minutes into accident.

ORNL-DWG 83-13711

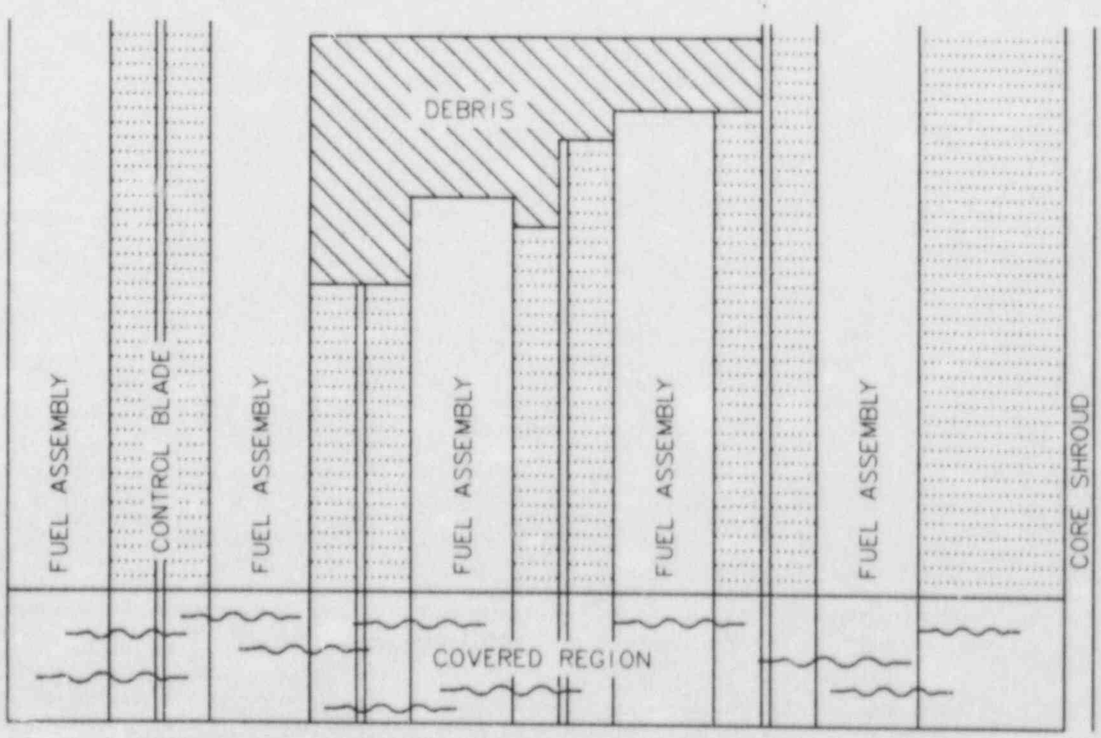

Fig. 6.6. BWR core melt progression — stage 1.

ORNL-DWG 83-13710

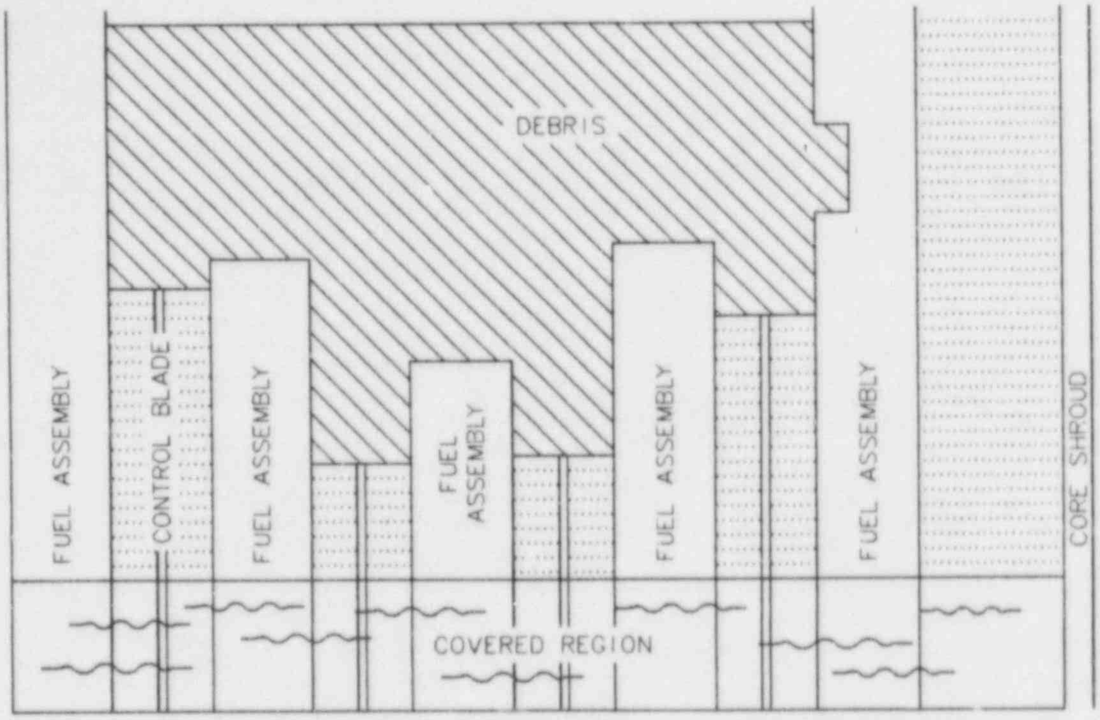

Fig. 6.7. BWR core melt progression — stage 2.

ORNL-DWG 83-13709

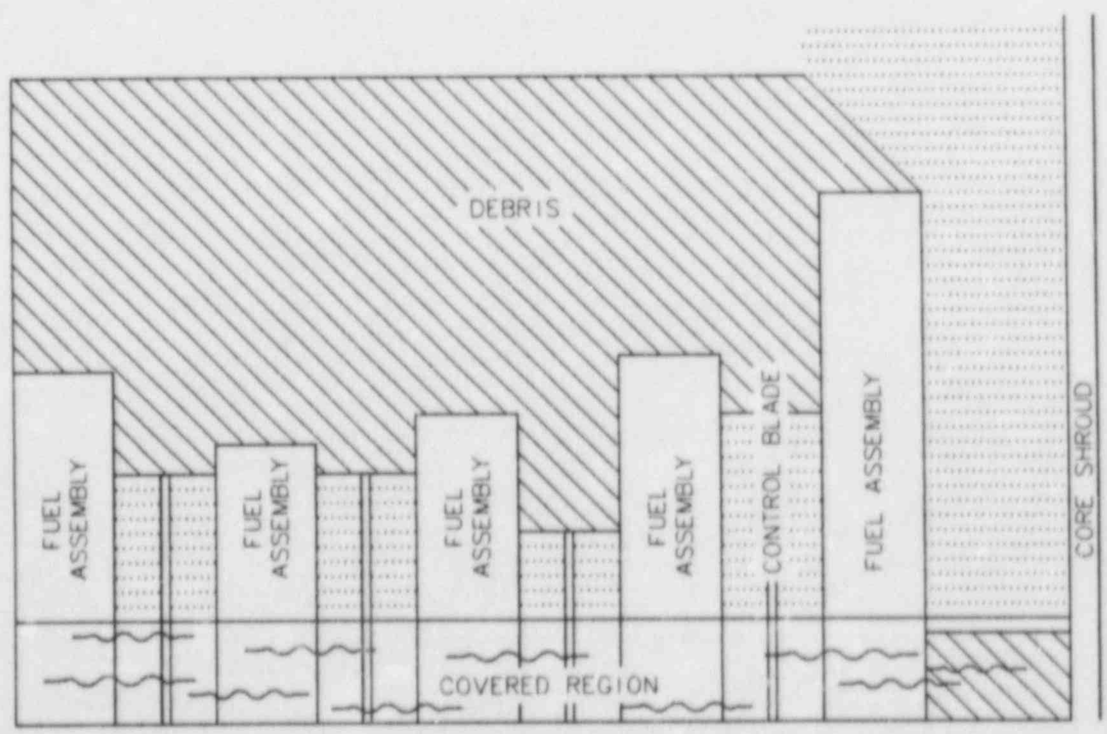

Fig. 6.8. BWR core melt progression — stage 3.

7. BWR IN-VESSEL SEVERE ACCIDENT MODELING REQUIREMENTS

7.1 Global Modeling Issues

7.1.1 Primary system model structure

The accident sequence described in Sect. 6.2 is a simplified scenario which involves neither primary system blowdown (due to pipe breaks) nor injection of water into the reactor vessel. It is clear, however, that accurate simulation of the scenario described in Sect. 6.2 will require the ability to predict structural temperatures throughout the entire vessel — vessel walls, lower vessel structures, core structures, and upper internal structures. This capability is necessary for accurate evaluation of hydrogen evolution, core structural deformation, vessel failure, and fission product retention within the primary system.

Figure 7.1 is a schematic representation of the most simplistic BWR primary system nodalization which would permit explicit representation of all major internal vessel structures and injection and leakage (pipe break, SRV, MSIVs, ECC turbine extraction steam) pathways into and out of the reactor vessel. Water can be injected into volumes 1, 2, 4, 5, and 7, and water, steam, and gases can leave the reactor vessel via breaks, SRVs, or MSIVs, in volumes 1, 2, and 7. Such a nodalization scheme would also facilitate examination of the impact of core and head spray injection (volumes 5 and 7) on vessel pressure and core oxidation. This scheme would also accommodate accurate annulus water level models, which are necessary for correct simulation of BWR emergency core cooling system operation, and the cooling effect of natural circulation within the vessel. A nodalization scheme of this type is also necessary to correctly treat fission product retention within the vessel during the latter stages of a pipe-break-initiated severe accident.

An adequate severe accident analysis code must be capable of analyzing internal vessel phenomena (structural temperatures, fuel melting, fission product transport, etc., subsequent to failure of the reactor vessel bottom head. MARCH's inability to treat invessel phenomena subsequent to vessel failure is a significant deficiency that should be avoided in future severe accident analysis codes.

7.1.2 Reactor vessel coolant leakage models

7.1.2.1 Background. Prior to vessel melt-through, the reactor water inventory decreases due to (1) open or leaking MSIVs, (2) pipe breaks, (3) periodic SRV actuations, or (4) stuck open SRVs. The object of this section is to briefly discuss the factors which must be considered in modeling these leakage mechanisms.

7.1.2.2 MSIV flow models. The great majority of BWR severe accidents would involve closure of the main steam isolation valves (see Sect. 2.11). However, for cases in which these valves fail to close or are leaking, flow through the open MSIVs must be modeled. For situations in which the MSIVs are more than 25% open, MSIV flow is actually limited by the main steam line flow restricting orifice rather than the valves themselves.[7.1,7.2] MSIV flow models must incorporate such considerations.

As previously stated (Sect. 2.5), BWR MSIVs are often found to leak excessively when they are tested during reactor refueling periods. Although the leakage is reduced to within Technical Specification limits prior to each startup, such leakage might occur during an accident following extended reactor operation. The coupled effect of this primary system leakage on reactor pressure, water inventory, and fission product transport, should also be represented in MSIV flow models. A simple flow vs pressure model (based on available experimental information) should suffice for most applications.

Prior to core uncovery, MSIV flow would consist of dry, saturated steam. Subsequent to core uncovery, the steam might be saturated or superheated, and possibly mixed with hydrogen and carbon monoxide.

7.1.2.3 Pipe break flow models. The pipe break flow models shoul be capable of accommodating critical and sub-critical gas, gas/liquid mixtures, and liquid flows. Prior to core uncovery, the gas will consist solely of saturated steam or water with a quality dependent upon the break location. Subsequent to core uncovery, the gas leakage might be saturated steam and water or superheated steam/H_2/CO mixtures. Liquid leakage conditions might range from saturated to several degrees subcooled — depending upon the break location and water level.

7.1.2.4 SRV flow models. As described in Sect. 2.4, BWR SRVs are critical flow devices, which are attached to the main steam lines upstream of the MSIVs. In general, SRV flow will consist of dry saturated steam prior to core uncovery, and dry saturated or superheated steam/hydrogen (and possibly carbon monoxide) mixtures subsequent to core uncovery. In addition to accommodating flows of this nature, the SRV model control logic should be capable of simulating the open — blowdown — close cycling characteristics of actual SRVs. The capability to model multiple SRVs with different opening/closing pressure setpoints should also be incorporated into future BWR severe accident analysis codes.

A simplistic flow model such as:

$$W = (I)\, W_r \left[\frac{\rho P}{\rho_r P_r}\right]^{0.5}$$

where

I = SRV control parameter (0 or 1)
W_r = rated SRV flow
ρ = average density of gas in upper reactor head
P = gas pressure in upper reactor plenum
ρ_r = steam density corresponding to rated SRV flow and pressure
P_r = rated SRV pressure (pressure at which W_r is achieved)

is probably sufficient for most applications.

7.1.3 Vessel water level models

As discussed in Sect. 2.6, reactor vessel water level (the single-phase collapsed level sensed in the shroud-to-vessel wall annulus region — volume 1 in Fig. 7.1) is the single most important BWR operating parameter. Vessel water level is a control parameter for almost all BWR emergency core cooling systems and the indicated level in the control room guides most operator actions. Because of this, a BWR severe accident analysis code must have an accurate annulus water level model. The model should correctly account for the axial variation of vessel internal cross sectional area due to steam dryers, separators and standpipes, core, shroud head and shroud, and the control rod drive guide tubes.

7.1.4 ECCS flow and control

For cases in which BWR ECCS systems operate as designed, the reactor vessel would be flooded in an extremely brief period of time. Severe accident analysis codes need not therefore strive to analyze such highly dynamic phenomena. However, severe accident analysis codes must be capable of treating cases involving operation of degraded ECCS systems or nondegraded systems at pressures near their shutoff head. Correct simulation of ECCS operation necessitates consideration of the actual ECCS control parameters, suction and injection locations, and pump performance characteristics. Table 7.1 is a listing of ECCS model capabilities which are necessary for realistic simulation of these systems. Since steam turbine driven HPCI and RCIC systems are generally operated in a constant flow mode, the steam demand of these turbines might be modeled simply as a steam demand vs primary system pressure curve which could be user input. The steam should be removed from the reactor during system operation and the turbine exhaust should be input to the suppression pool.

Variable flow systems might be modeled with a user input pump head curve (flow vs reactor or containment pressure). Since many systems can draw suction from diverse sources, it is necessary that the ECCS model utilize the correct pressure differential with these head curves. Accommodation of multiple pump suction reservoirs, reservoir elevations, etc. will also enable the model to correctly account for reservoir mass, pump NPSH requirements, and automatic switch of suction from one reservoir to another.

7.2 Pre Core Uncovery Modeling Issues

For purposes of discussion, it is convenient to divide the severe accident sequence into two phases: pre core uncovery and post core uncovery. While the reactor core is covered, the temperature of the in-vessel structures will remain relatively stable. If the accident is initiated while the reactor water level is near its normal operating range, several hours might pass before the core begins to uncover (core uncovery time will depend upon decay heat level, vessel leakage rate, injection flow, and initial vessel water inventory). The major modeling requirements for evaluation of this phase of the accident are, therefore, quick running, accurate models for (1) MSIV leakage, (2) pipe break flow, SRV actuation and flow, (4) vessel water level, and (5) vessel injection systems control and flow. Detailed core structural and heat transfer models are probably not necessary for evaluation of this phase of the accident.

7.3 Post Core Uncovery Modeling Issues

7.3.1 Background

Subsequent to core uncovery, the temperature of the upper core structures will immediately begin to increase due initially to decay heat input and later to decay heat and heat from the $Zr-H_2O$ and $Fe-H_2O$ reaction. If the reactor is pressurized, periodic SRV actuation will produce vessel water level swells which briefly cover all or portions of the upper core. Water might be introduced, either in the form of spray or liquid, at different locations in the vessel. If this injection flow is insufficient to cool the core, melting will eventually occur.

Melting and relocation of the core debris can proceed at different rates in different regions of the core. Since the individual BWR fuel assemblies are supported from beneath, it is highly unlikely that the so-called "core slump" phenomenon (as modeled in MARCH) will occur. Instead, it is probable that a debris bed would form in-core. This debris bed would gradually relocate downward as the water is boiled from the lower plenum. The bed will eventually melt through the core plate and attack the control rod drive housing, instrument, and stub tube assemblies. It is possible that unmelted fuel and perhaps even unfailed fuel pins will exist in the reactor vessel at the time of bottom head failure.

The structure and modeling strategy of future severe accident analysis codes should be designed to accommodate the phenomena described above. This will necessitate a code structure in which downward melt progression can proceed at different rates in different regions of the core.

7.3.2 BWR core heat transfer models — intact geometry

7.3.2.1 Significant structures. Figure 7.2 is a simplified plan view of a BWR core. The core basically consists of two types of fuel assemblies. The first type of assembly is adjacent to other assemblies on two sides and adjacent to control rods on two sides (Assembly A, Fig. 7.2). The second type of assembly (Assembly B, Fig. 7.2) is adjacent to control rods and/or other canisters on one or two sides and to the core shroud on the other sides. Within each fuel assembly, the individual fuel pins can be grouped into three general types: interior fuel pins which communicate both with interior and peripheral pins, peripheral pins adjacent to channel walls which are adjacent to other channel walls, and peripheral pins which are adjacent to channel walls which are adjacent to control rods. As described in Sect. 2.3.4, the top and bottom of each fuel pin is comprised of natural, rather than enriched uranium. The ends of the pins are slip fit or thread attached to the upper and lower assembly tie plates.

Table 7.2 is a summary listing of significant BWR core structural heat transfer communication pathways for intact geometries. Interior fuel pins transfer energy via thermal radiation to other interior fuel pins, peripheral pins, and the coolant (water, steam, H_2). The pins also communicate with the surrounding coolant via convection and conduction. In addition to these structures and fluids, peripheral fuel pins radiate to adjacent canister walls. Axial conduction in the pins will result in smoothing of axial pin temperature profiles and heating of the upper and lower assembly tie plates.

The canister walls exchange energy via convection, conduction, and radiation to the assembly coolant on one side, and to the bypass coolant on the other. The inside canister walls also communicate via thermal radiation with the peripheral fuel pins, while the outer surfaces of the canisters radiate to either control rods, other canister walls or the core shroud. Axial conduction in the canister walls will also result in flattening of axial canister wall temperature distributions and heating of upper and lower assembly tie plates.

A realistic simulation of BWR core melt may necessitate explicit representation of all the structures and fluids listed in Table 7.2, together with explicit representation of both fuel and cladding in the fuel pin models (see Sect. 7.3.2.3). Axial fuel pin and canister conductive heat transfer models are necessary both for evaluation of fuel pin, canister, and tie plate temperatures, and for correct simulation of core steaming rate when the water level is near the bottom of the core. Inter-structure radiative heat transfer models should accommodate changing emissivities, transmissivities, and view factors due to aerosol clouding of the vessel atmosphere during the core melt process.

7.3.2.2 Thermohydraulic phenomena. In addition to modeling the basic structures in the BWR core, BWR severe accident analysis heat transfer models must be capable of correctly representing core thermohydraulic phenomena. These phenomena include flashing, level swell, and rewetting of structures during SRV actuation. Simplified countercurrent flooding models are required for analysis of accidents involving degraded core spray and LPCI operation. Separate calculations for fuel bundle and interstitial zonal water levels are also desirable for this purpose. The ability to model core spray system operation is particularly desirable since operation of degraded core spray systems might significantly alter the axial fuel cladding and canister oxidation profiles.

All significant heat transfer paths between core fluids and structures should be modeled (Fig. 7.2). Heat transfer coefficient correlations should be appropriate for the full range of conditions from liquid natural convection to film boiling and radiation, for laminar and turbulent flows of subcooled and saturated liquid and saturated and superheated steam and steam/hydrogen mixtures. Physically based fuel pin, canister, and control rod quench models should be utilized wherever possible rather than models based on arbitrary quench time constants (such as these employed in MARCH).

Vessel steaming rate and flashing calculations should accurately simulate the nature of invessel steam generation due to heat transfer and vessel depressurization. It is probable that this requirement will necessitate a multinode treatment of the water pool within the reactor vessel. Significant difficulties have been encountered with oscillatory behavior of MARCH's invessel steam generation models. Many of these problems can be avoided with correct nodalization of the primary system.

7.3.2.3 Structural oxidation kinetics. Realistic BWR severe accident core models must accommodate accurate models for fuel cladding and fuel canister oxidation reactions. Although the control rod blades are stainless steel clad, significant iron oxidation is not expected below temperatures approaching that of the melting point of the metal. It may, therefore, be acceptable to ignore oxidation of the stainless steel control rods prior to rod deformation.

Accurate prediction of cladding and canister oxidation rates requires accurate (1) clad and canister surface temperatures, (2) coolant flow rates, (3) coolant mass composition, and (4) appropriate oxidation rate equations. The need for accurate fuel clad temperatures will necessitate a multinode fuel pin model with separate treatment of cladding and fuel. It is unclear at this time whether multinode models for the canister wall are required. A single node canister wall model might suffice if coolant flow rates and mass compositions are correctly treated on each side of the canister wall.

Existing MARCH 1.1 models have no explicit treatment of either the fuel cladding or canister walls. The ORNL MARCH 1.1B code does incorporate a simplistic BWR core model which explicitly treats the canister wall and control rod structures — but not the fuel cladding. The MELRPI BWR core model developed by Rensselaer Polytechnic Institute for ORNL does incorporate explicit treatment of fuel cladding, canisters, and control rods.

Accurate prediction of structural oxidation rates will require accurate information regarding steam availability at the metal surface. This, in turn, requires accurate simulation of invessel steaming rates. This has been a significant problem for users of the MARCH code, since MARCH typically predicts highly oscillatory vessel flashing rates.

Appropriate oxidation rate equations should be utilized. The equations utilized in the MARCH 1.1 code (Baker-Just and Cathcart) are generally acknowledged to be appropriate only for situations in which there is excess steam present.[7.3] In many accident situations, various regions of the core will be deprived of steam due to low vessel steaming rates, flow blockages, steam utilization by oxidation of lower core regions, or a combination of these circumstances. Oxidation rate equations appropriate for such steam depleted conditions should be incorporated in future severe accident analysis codes. The oxidation kinetics equations should also accommodate simulation of hydrogen blanketing phenomena which might occur during low core flow conditions (such as the periods between SRV actuations).

7.3.3 Structural deformation concerns

7.3.3.1 Control rod deformation. Following core uncovery, the stainless steel control blades are expected to begin deforming first, due to their relatively low melting point (2600 F vs 3400 F for zircaloy). Depending upon local steam availability at the time blade melting begins, either of two scenarios are possible. If sufficient steam is available to rapidly oxidize the molten stainless steel, an iron oxide and boron carbide slug would be formed.[7.3] The boron carbide may react exothermically with the steam to produce additional hydrogen and carbon monoxide. This slug could remain in place, effectively filling the half inch gap between the two adjacent canister walls in the reaction zone. It is possible that the portion of intact control blade above the reaction zone would also remain in place.

If sufficient steam is not available to rapidly oxidize the metal, molten material would begin falling down along the inside and outside of the steel-sheathed control blades, ultimately melting the internal stainless steel tubes which contain the boron carbide powder. This molten material would gradually relocate downward along the surface of the control rod via a series of melt/flow/freeze phenomena. If melting of a particular control rod begins at a point below the top of the rod, the intact portion of the control rod above the molten zone will fall downward. Localized interstitial region flow blockages may occur. Since melting of the control blade is primarily driven by heat input from adjacent canister walls, the upper intact portions of the control rod may eventually fall into the hot zone and melt.

Existing severe accident analysis codes do not analyze these phenomena. Both MARCH 1.1B and MELRPI assume that molten control rod material stays in place until the total core collapses. Appropriate control rod melt models should accommodate either slug formation of molten material relocation. Models should also accommodate thermal communication of can ister surfaces once the control blade has relocated downward.

7.3.3.2 Fuel pin deformation and failure. The basic methods by which fuel cladding can deform are rupturing, buckling, and shattering. These phenomena occur due to changes in material properties brought about by excessive temperatures, oxidation or melting, and by excessive temperature or pressure gradients across the cladding material. Failure of the fuel cladding will, in any case, result in the release of gaseous fission products and subsequent admittance of steam to the inner surface of intact cladding material. Depending on the nature of the failure, a localized debris bed may be formed as the UO_2 fuel pellets fall out of their original configuration.

A reasonable condition for signaling clad rupture is when the clad hoop stress exceeds the equivalent yield stress of the composite Zircaloy/zirconium oxide clad material. For cases in which the reactor pressure exceeds the internal fuel pin pressure, a buckling criterion is a more appropriate cladding deformation signal. Shattering of clad material due to thermal shock may occur during situations in which very hot fuel pins are recovered with water. This situation could be brought about by latent ECC injection or, possibly, by the level swell which accompanies SRV actuation. Recent work performed at Argonne National Laboratory[7.4] could, perhaps, provide the basis for a shattering criterion.

Due to the nature of the decay heat source, it is possible that interior fuel pins within the fuel assembly would reach failure conditions prior to the failure of peripheral fuel pins and the canister wall. This could lead to the existence of localized debris beds within intact fuel assembly canisters. Molten cladding and cladding/fuel mixtures would flow down the fuel pins, eventually freezing in lower, cooler regions of the core. Recent work by Moore and Broughton[7.5] indicates that axial heat conduction in the fuel rod may have a significant influence on the redistribution and freezing of the liquified fuel material. Fuel pin axial heat conduction models should, therefore, be incorporated in future severe accident analysis codes. Significant fractions of the assembly cross-sectional flow area might become blocked due to accumulation of this resolidified material.

None of the phenomena described above are modeled by MARCH. The MELRPI model does incorporate simplified rupture, buckling, and shattering failure calculations of the type previously described. The criteria for switching from intact geometry to a rubble bed structure is specified in terms of beam buckling theory or a quenching criterion. Molten cladding relocation is modeled in MELRPI with a simplified falling liquid slug model.

7.3.3.3 Canister deformation. ORNL's experience with the MARCH 1.1B code indicates that canister wall temperatures typically follow fuel cladding temperatures very closely. This is due primarily to radiative heat transfer between the two surfaces. It is, therefore, expected that canister wall melting will begin at approximately the same time as melting of the cladding on adjacent fuel pins.* Experience with MARCH 1.1B also indicates, however, that canister wall radial oxidation profiles are not, in general, symmetrical about the center of the wall. This is due primarily to differences in steam availability inside and outside the canister.

Since canister wall melting would generally begin below the top of the core, portions of the canister above the melt zone are expected to fall downward into the rubblized zones as the melting process continues. The canister wall will, in general, be subject to the same types of deformation phenomena as the fuel cladding, with the exception of rupture mechanisms.

7.3.3.4 Summary — desirable model characteristics. The three preceding sections have presented a summary of control rod, fuel pin, and canister wall deformation phenomena under severe accident conditions. Based on these descriptions, a reasonable scenario for progression of the core melt process within a single core zone is presented in Table 7.3. Table 7.4 presents a comprehensive listing of desirable BWR core melt model capabilities.

No currently available models have all of the capabilities listed in Table 7.4. Existing MARCH 1.1 core models are inappropriate in many respects. MARCH 1.1 has no explicit treatment of canisters or control blades, no structural deformation models, no explicit treatment of cladding, no in-core debris bed models, no axial conduction, and inappropriate fuel pin quench models.

The ORNL MARCH 1.1B code does explicitly treat canister walls and control blades and has improved fuel pin quenching models, but has most of the other limitations cited for MARCH 1.1. With the exception of multiple fuel pin representations, MELRPI has most of the capabilities listed in Table 7.4.

7.3.4 Lower plenum melt progression

If the BWR core melt sequence is not terminated, core debris will eventually contact the assembly lower tie plates, fuel support pieces, and core plate. The portions of the stainless steel core plate contacted by the debris are expected to rapidly melt, allowing material to penetrate into the lower plenum region along both the inside and outside of the control rod guide tubes. This downward relocation may proceed at different rates in different regions of the core in some accidents. In

*The fuel pin cladding is Zr-2; the canister walls are Zr-4. The melting temperatures of the two alloys are not significantly different.

some accidents downward progression of the debris into the lower plenum may be limited by the boiloff rate of water in the lower plenum and the control rod guide tubes.

Due to the geometric arrangement of the control rod guide tubes, it is possible that debris would penetrate along the inside of the guide tubes more quickly than along the outside. This is illustrated by Fig. 7.3, which is a plan view of the lower plenum region, showing the available debris flow areas inside and outside the guide tubes. It is reasonable to assume that significant amounts of core debris would fall into the region inside the guide tubes as well as between the guide tubes. As in the case of the control rods, molten stainless steel from the guide tubes may oxidize if sufficient steam is available, forming localized slag plugs inside or between adjacent guide tubes.

An appropriate model for BWR lower plenum melt progression might incorporate ingredients previously discussed for the fuel bundle and control rod melt models. A reasonable approach would be to model a single typical guide tube in each of the radial regions previously defined for the core model. Each guide tube would be segmented into axial nodes in a fashion similar to the core model. Following localized core plate failure, some fraction of the overlying debris could be assumed to fall into the regions inside, between, and around the outer periphery of the guide tubes. The amount of material which might actually fall into these regions is extremely uncertain, but would be determined by geometric factors such as the average size of the core debris and the distribution of the debris above the core plate. The problem could be bounded by assuming that (a) no material falls directly into the lower head area or (b) that enough material falls into the free areas in the lower plenum to completely fill the regions inside and between the guide tubes. A check must, of course, be made to ensure that sufficient debris is available to fill the interstitial regions.

For the case in which no debris is assumed to fall directly into the lower plenum, an appropriate model must accommodate axial conductive heat transfer and radiative heat transfer from the overlying debris bed to the guide tubes, radiative heat transfer from the debris to the water, and oxidation and melting of both the debris and the guide tubes. Due to the relative thinness of the guide tube walls, a simple two-dimensional cylindrical heat transfer model might suffice.

The case in which the core debris does fall into the lower plenum is more complex. The internal heat generation rate of the debris is very uncertain. The internal energy of the debris is utilized both to heat the pool and the lower portions of the guide tubes, stub tubes, and perhaps the bottom head of the reactor vessel. An appropriate model for this scenario must have all the capabilities of the previous model plus the capability to handle debris quench and radial conductive heat transfer from the debris into the guide tubes, stub tubes, and vessel walls. Both of these scenarios would require a model which has the capability to accommodate different rates of downward melt progression in different radial zones and the effects of CRD hydraulic system flow inside the CRD guide tubes (for cases in which this system is not assumed to fail).

The author is unaware of any existing computer model which simulates either of the aforementioned lower plenum melt progression scenarios. Existing MARCH models for lower plenum melt progression are extremely unmechanistic. Debris relocation in the lower plenum is based on a temperature criterion which allows the entire mass of debris to relocate when the average temperature of the core debris and lower plenum structures exceed a user input "grid plate failure temperature." Such models are inappropriate for application to boiling water reactors.

7.3.5 BWR vessel head attack and failure

7.3.5.1 Phenomena. As noted in Sect. 2.2, the bottom head of a BWR vessel contains numerous penetrations for control rod drive housings, incore instrument housings, and low point drains. Current designs utilize between 137 and 193 control rod drive housing penetrations, 53 to 55 incore instrumentation tube penetrations, and 1 vessel drain penetration. Under accident conditions, it is likely that one or more of these penetrations would fail prior to gross melt-through of the bottom head. For reasons described below, it is unclear which type of penetration would fail first, or the exact manner in which the failure would occur.

Consider the situation in which a large debris bed is progressively relocating downward into the lower plenum region. As the bed slowly moves downward, the stainless steel control rod guide tubes and housings and the instrument tubes will melt. If insufficient steam is available to rapidly oxidize the molten stainless steel, the material will run down the sides of the guide and instrument tube assemblies, eventually refreezing on lower portions of the drive housings, stub tubes, and instrument tubes. The internal heat generation rate of this material may be fairly low, since it will consist primarily of molten steel. As the debris continues to relocate downward, axial heat conduction in the tubes and radiative heat transfer from the advancing debris would result in attack of the CRD housing/stub tube and instrument tube welds, and the collection of a mol ten pool of stainless steel directly over the vessel drain.

Since the CRD housing/stub tube welds are located above the surface of the lower head, it is likely that the integrity of these welds would be challenged prior to the buildup of a significant amount of molten material on the bottom head. Failure of this weld could result in a 1 to 3 in. downward movement of the associated CRD mechanism and the opening of a very small (<0.01 in. wide) annular gap between the housing and the vessel head opening, through which the reactor would begin depressurizing. Complete failure of these welds would not result in the immediate ejection of the CRD housing from the vessel. Significant additional melting of the CRD mechanism and housing would be would be necessary before the entire 6-in. diameter CRD housing penetration would be opened. During this period, it is possible that the continued accumulation of molten material over the reactor vessel drain would result in failure of the hollow, thin-walled (<0.4 in.) nozzle and the ejection of a limited amount of molten stainless steel onto the CRD mechanism support structures below the vessel. Ablation of the drain opening by the flowing molten steel could result in a significant increase in the size of the failure opening. It is, of course, also possible that molten material could migrate down inside the 2-in. instrument tubes, failing the thin walled tubes outside the reactor vessel.

To summarize, it is likely that the reactor vessel would fail due to melt-through of one of the many bottom head penetrations rather than direct melt-through of the 20 cm (8-in.) thick head (as modeled in MARCH). It is not clear, however, which type of bottom head penetration would fail first during a core melt accident. Total circumferential failure of a CRD housing/stub tube weld would not result in ejection of the associated mechanism from the vessel, but rather the opening of a thin annular crack through which the reactor could depressurize. Failure of the bottom head drain or an instrument tube [both approximately 5 cm (2 in.) in diameter] could occur prior to complete failure of any CRD housing penetrations. Finally, it appears likely that failure of bottom head integrity would occur prior to the accumulation of a large molten pool in the bottom head.

7.3.5.2 Existing models. The author is aware of only one existing mechanistic model for localized failure of the bottom head. The model, developed by Henry,[7.6] is incomplete in that it does not predict the occurrence or location of the failure. Henry's model assumes (1) a large overlying pool of molten material and (2) a circular failure opening. (As previously stated, it is unclear what fraction of the core debris will be in the molten state at the time of head failure, and the failure opening geometry is also unknown.) Henry's model does, however, simulate ablation of the opening by the flowing molten material, the discharge rate of the molten material from the vessel, and the depressurization rate of the vessel.

Hagen[7.7] has developed a set of steady-state, one-dimensional, moving boundary models for gross reactor head melting. The models do account for heat transfer on the outside surface of the vessel and for several possible layer configurations of solid and molten steel and UO_2. The referenced report presents a transient vessel melting model which utilizes a finite difference two-dimensional, moving, phase change boundary formulation. None of the models described in Ref. 7.7 accommodate in-vessel debris bed geometries or stress failure considerations.

Existing MARCH head failure models are designed to evaluate only gross head melting rather than localized penetration degradation. The models employed in MARCH assume that the outside of the vessel is insulated. Heat conduction in the bottom head is modeled using the concept of a thermal penetration distance in a solid, homogeneous shell. The bottom head is assumed to fail in a gross fashion when the total tensile stress at the interface of the head hemisphere and vessel cylinder or in the vessel wall at the surface of the debris, exceeds the temperature dependent strength of the material. When this occurs, MARCH assumes that all of the core debris (liquid + solid) is instantly transferred to the drywell floor.

7.3.5.3 Modeling recommendations. A mechanistic bottom head failure model cannot be developed independently from a lower plenum melt transport model. This is true because prior to development of a mechanistic lower plenum melt relocation model, one does not know (a) the degree of coherence of the melt relocation, (b) the physical state and composition of the debris, (c) the internal heat generation rate of the debris, nor (d) the mechanism by which the debris reaches the bottom head.

and A_2 is the equivalent horizontal projection of some apportioned fraction of the bottom head surface in the given radial zone. The weld joint could be assumed to fail when the total internal energy of the debris above the weld area exceeds some critical value. This approach is similar to that currently used to model grid plate failure in MARCH. If desired, the model could be designed to accommodate the falling of solid material from the debris bed onto the area A_2. Models similar to those developed by Henry could be employed following weld failure, and models similar to those developed by Hagen or those currently employed in MARCH could be utilized to simulate heat transfer to and failure of the area of the head (A_2) surrounding the housing and instrument tube. In the central region of the vessel, the above models would have to be modified to account for the possibility of failure of the head drain nozzle.

7.4 In-vessel Fission Product Transport Phenomena and Modeling

The intent of this section is not to provide a detailed discussion of generalized fission product transport phenomena, but rather to highlight those areas where differences in BWR and PWR designs dictate different fission product transport modeling approaches. The major unique characteristics of BWRs are (a) the size and arrangement of in-vessel structures, (b) the impacts of SRV operation during pressurized accident situations, (c) steam demand of turbine driven ECC systems, and (d) core and head spray operation.

Figure 7.5 is a simplified schematic of the BWR in-vessel structural arrangement. The normal flow path for material leaving the core is identified as path A in the figure. Material flows up through the core, into the steam dome, through the standpipes, separators, and dryers, into the upper head, and out of the vessel via the main steam lines. For situations other than those involving a break somewhere in the reactor coolant boundary, the majority of the fission products and aerosols leaving the melting core during the early stages of the accident would follow this flow path prior to release to the main condenser or pressure suppression pool.

Table 7.5 presents a summary of the surface areas of the BWR upper internal structures. Due to the substantial surface areas available in these structures, fission product plateout and deposition are major concerns. Accurate evaluation of these phenomena necessitates realistic estimates of structural temperatures. During the early stages of a core melt accident, these structures will be significantly colder than the gases and aerosols which are evolving from the core. Substantial amounts of fission product deposition on these structures are expected during this phase of the accident. Later in the accident it is probable that some or all of the structures would heat up significantly, perhaps driving off much of the more volatile material which had previously deposited in and near their surfaces. The effectiveness of the steam separators in removing aerosols from the gas stream is not fully understood. Any aerosols trapped by the separators would either deposit on amounts of fission product deposition on these structures are expected during this phase of the accident. Later in the accident it is probable that some or all of the structures would heat up significantly, perhaps driving off much of the more volatile material which had previously deposited in and near their surfaces. The effectiveness of the steam separators in removing aerosols from the gas stream is not fully understood. Any aerosols trapped by the separators would either deposit on the interior surfaces of the separators or be directed to the outside of the separators where they might settle out for possible resuspension later by one of the bypass flows described below.

Two alternative flow paths are shown in Fig. 7.5 bypass some or all of the upper internal structures. Path B is a bypass path which exists in the reactor when the water level is below the dryer skirt. This path is present due to the design of the steam dryer skirt which extends down past the top of the steam separators, but does not form a seal between the regions of the vessel above and below the dryer assembly. The second bypass path (path C, Fig. 7.5) is open anytime the reactor water level is below the outlet of the jet pumps. It is possible that this path would only become significant during the later stages of the core melt accident, when core debris has penetrated into the lower plenum region. It should be noted that failure of the core shroud or shroud head would also enable gas and aerosol flows to bypass the upper core internals.

The second major design characteristic of BWRs which influences fission product transport within the reactor vessel is the operating characteristics of the SRVs. As described in Sect. 2.4, BWR SRVs are not continuous flow devices. Once open, an SRV will remain open until the primary system pressure drops 34 to 69 KPa (50 to 100 psi). For accident conditions in which the reactor remains isolated and pressurized, this operating characteristic of the SRVs will result in primary system pressure and SRV steam flow histories similar to those shown in Fig. 7.6. SRV actuation has two major in-vessel impacts: vessel water level swell and an increase in steam flow through the core due to flashing of the water. The water level swell results in a brief recovering of hot core debris and scrubbing or dissolution of the associated fission products.

It is also apparent that the operating characteristics of the SRVs will result in brief periods of significant steam and gas flows through the core debris and structures, separated by substantial periods of time during which the reactor vessel is slowly repressurizing. The internal natural circulation patterns within the reactor vessel during the relatively quiescent periods between SRV actuations and the capability to model natural circulation paths within the vessel should be included in future codes. The impact of SRV actuation on in-vessel fission product retention and transport of fission products to the pressure suppression pool should also be modeled.

For accidents involving operation of steam turbine driven HPCI and RCIC systems, the impact of this steam demand on internal reactor vessel circulation patterns and steaming rates should be considered. The steam demand of these turbines ranges between 40,000 and 184,000 lb_m/h. For accidents in which these systems operate subsequent to core damage, the impact of these steam flows on in-vessel fission product retention and transport of fission products to the suppression pool should be modeled.

BWRs have two separate systems by which water can be sprayed into the reactor vessel. High and low pressure core spray systems inject into the shroud head region above the core. A second spray nozzle in the upper reactor vessel head might be used during accident conditions to spray water into the upper vessel plenum. The effects of these sprays would include entrainment of deposited materials, cooling of structures, and alteration of circulation patterns within the vessel. The efficiency of these sprays in removing fission products and aerosols from the internal vessel atmosphere is highly uncertain. It is desirable, however, to include some simple treatment for this phenomenon in the analysis of accidents in which these spray systems are operable.

It is possible that unmelted fuel pellets, and perhaps even unfailed fuel rods, will be present in the reactor at the time of bottom head failure. The history of this material subsequent to head failure is particularly important, since the fission product scrubbing capability of the suppression pool may be bypassed subsequent to vessel breach (containment type and accident sequence dependent). Realistic fission product transport models must therefore have the capability to evaluate the status and distribution of fission products within the reactor vessel throughout the entire course of the core melt accident — both prior to and following vessel failure.

It is clear from the discussion in this chapter that a realistic assessment of severe accidents requires a coupled heat transfer/thermohydraulic/fission product transport analysis approach — rather than the decoupled (MARCH/CORRAL) approach currently utilized for such evaluations. This is particularly true due to the decay heat source associated with fission products and the possible effects of aerosol induced atmospheric clouding and deposition on various heat transfer mechanisms within the reactor vessel.

Chapter 7 References

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7.7 R. C. Hagan, Transient Thermal Analysis of Molten Core-Reactor Vessel Systems, Ph.D. Thesis, University of Virginia, (1974).

Table 7.1. ECCS modeling requirements

Automatic actuation on low reactor water level
Automatic actuation on high drywell pressure
System trip on high reactor water level
System isolation on low reactor pressure
Steam demand of HPCI and RCIC pump turbines
Constant flow pumps
Variable flow pumps
Multiple suction sources
Auto switch of suction from one source to another
Correct vessel injection types (liquid or spray)
Correct vessel injection locations (lower plenum, shroud head, etc)
Manual operator actuation

Table 7.2. Core structural heat transfer paths — intact geometry

	Interior fuel pins	Peripheral[a] fuel pin (can type)	Peripheral[b] fuel pin (CR type)	Assembly coolant	Canister[a] wall (can type)	Canister[b] wall (CR type)	Assembly tie plates	Interstitial coolant	Control rod	Core shroud
Interior fuel pin	R	R	R	R,V			D			
Peripheral fuel pin (can type)	R	R		R,V	R		D			
Peripheral fuel pin (CR type)	R		R	R,V		R	D			
Assembly coolant	R,V	R,V	R,V		R,V	R,V	V			
Canister wall (can type)		R		R,V	R		D	R,V		R[c]
Canister wall (CR type)			R	R,V		R	D	R,V	R	
Assembly tie plates	D	D	D	V	D	D				
Interstitial coolant					R,V	R,V			RV	(R,V)[c]
Control rod						R		R,V		
Core shroud					R[c]			(R,V)[c]		

[a]Canister wall adjacent to canister wall.

[b]Canister wall adjacent to control rod.

[c]Outer row of assemblies only.

NOTE: R — Radiation
V — Convection
D — Conduction

Table 7.3. BWR zonal core deformation scenario

1. Control Rod Melts. In steam rich environment, rapid iron oxidation may form a slug which remains in place. In steam depleted environment, molten material will relocate downward along blade surface.
2. Interior fuel pins fail — Intra-assembly debris bed formed.
3. Peripheral fuel pins adjacent to canister walls fail. Entire intra-assembly area is rubblized.
4. Canister wall fails.
5. Entire zone rubblizes.
6. Upper zones in same radial region relocate downward.
7. Fuel pellet melting begins.

Table 7.4. Desirable BWR core melt modeling capabilities

1. Representation of 1 fuel assembly and control rod per radial zone.
2. Representation of 3 fuel pins/assembly (interior, canister/canister peripheral pins, and canister/control rod peripheral pins)
3. Two radial nodes per pin (fuel, clad)
4. Single node canister wall with nonsymmetric radial oxidation profile.
5. Cladding failure on burst, buckle, or shatter.
6. Debris bed formation on buckle or shatter of peripheral fuel pins.
7. Falling slug model for molten material (clad, canister, control blades).
8. Frozen plug formation model for melting control blades.
9. B_4C-H_2O reactions.
10. Downward relocation of intact structure above deformation zone.
11. Axial conduction in all structures.
12. Appropriate radiation heat transfer models (including effects of aerosol clouding and appropriate structures).
13. Oxidation kinetics appropriate for both steam rich and steam starved condition.
14. Physically based fuel pin, canister, and control rod quench models.
15. Decay power with actinide contribution.

Table 7.5. Estimated structural surface areas

Structure	Surface area [m^2 (ft^2)]
Shroud	81 (870)
Shroud head	21 (225)
Standpipes	209 (2250)
Separators	260 (2800)
Dryers	2973 (32000)
Upper head	121 (1300)

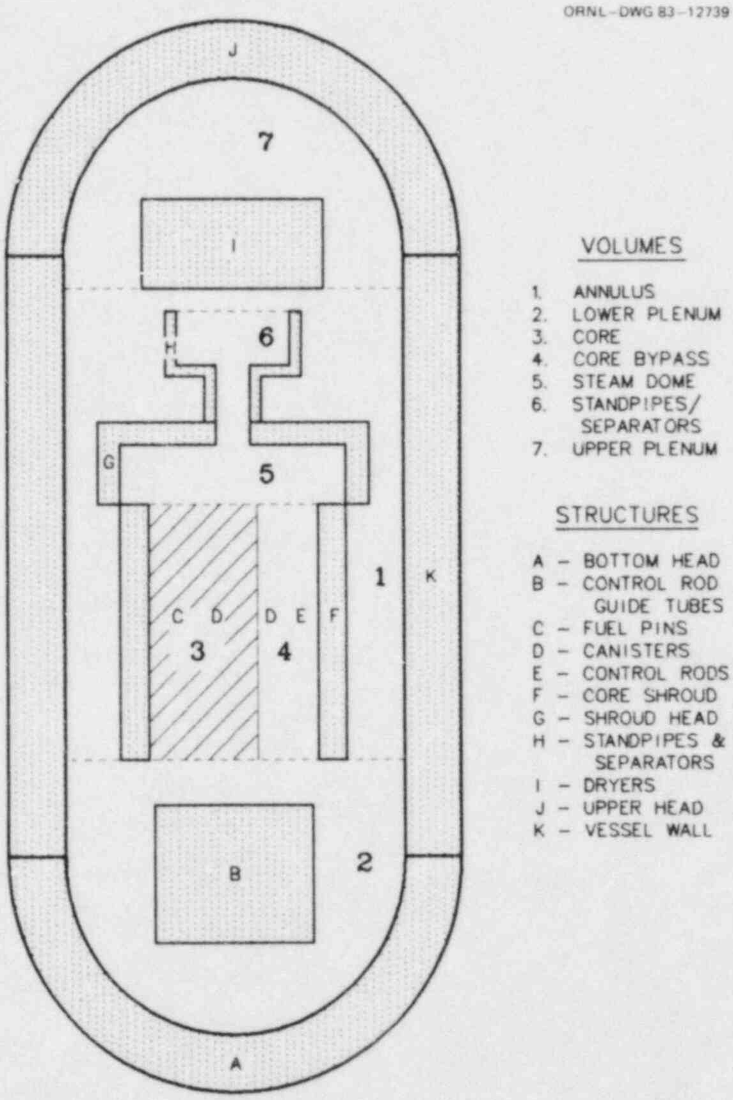

Fig. 7.1. Simple BWR primary system nodalization.

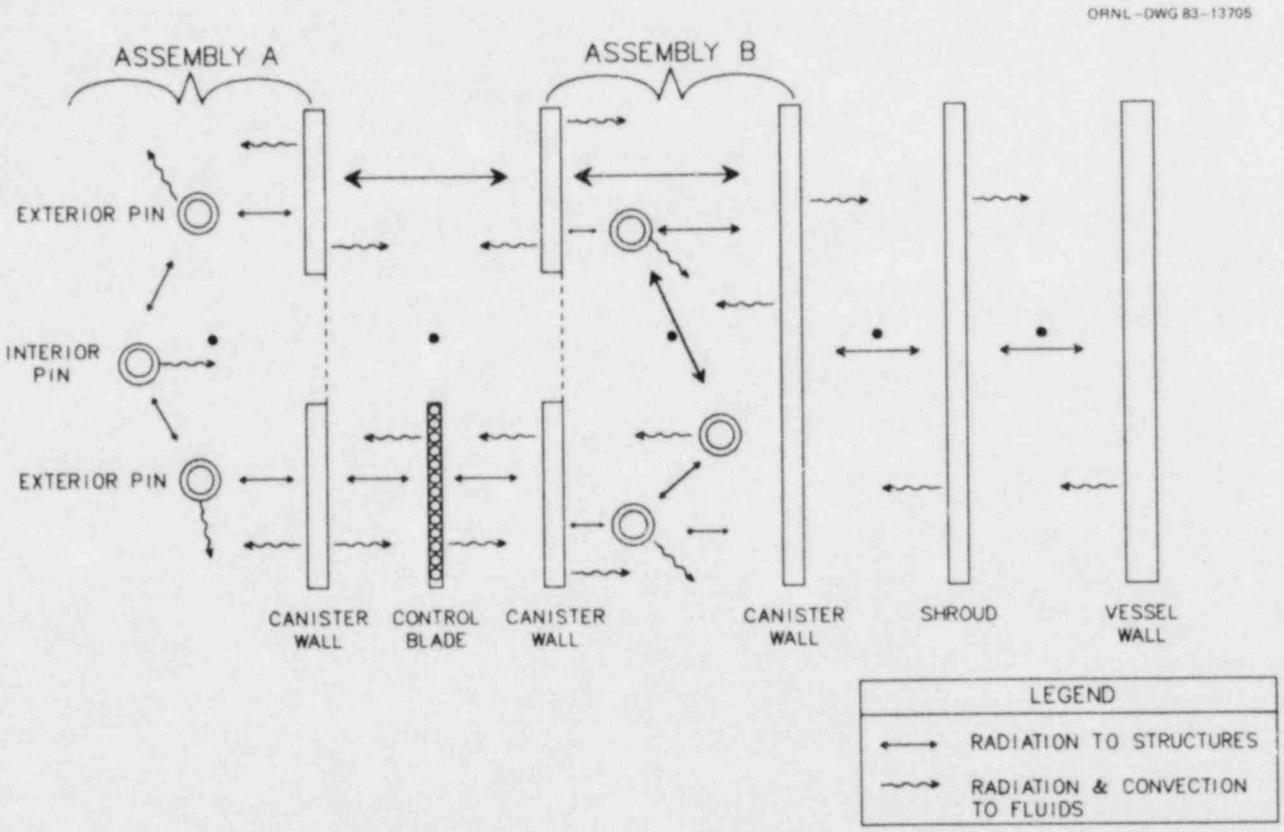

Fig. 7.2. BWR core radial heat transfer paths.

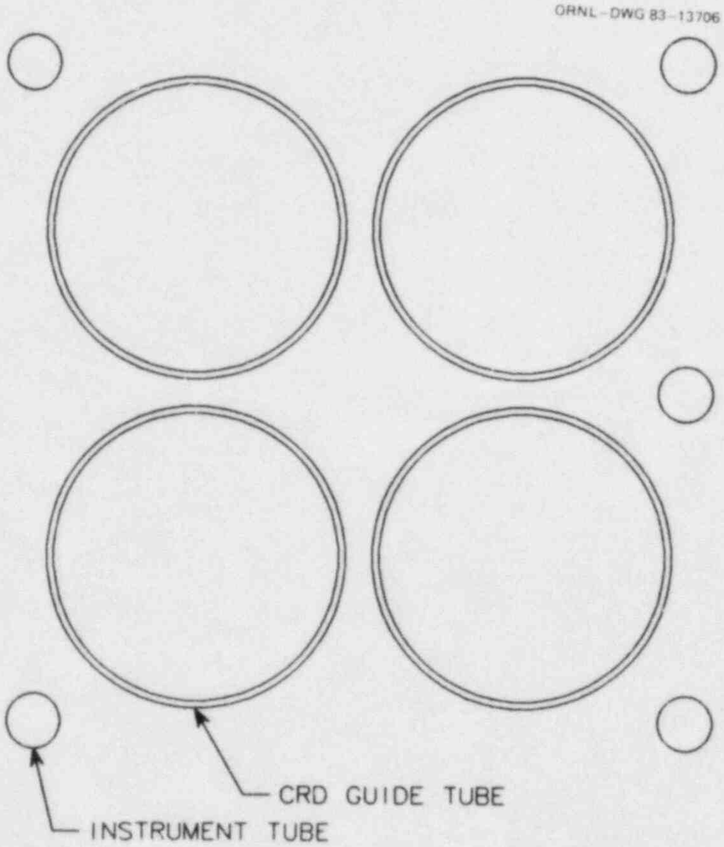

Fig. 7.3. BWR lower plenum structures (plain view).

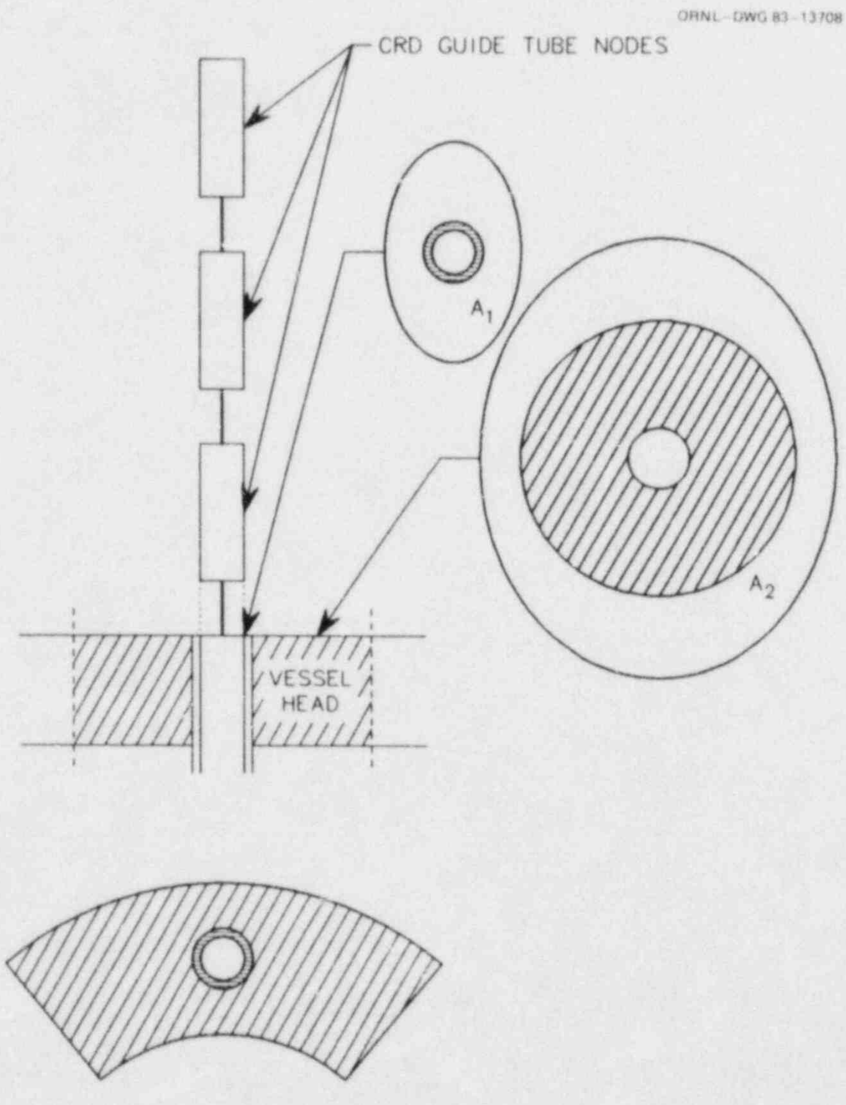

Fig. 7.4. BWR vessel head failure model structure.

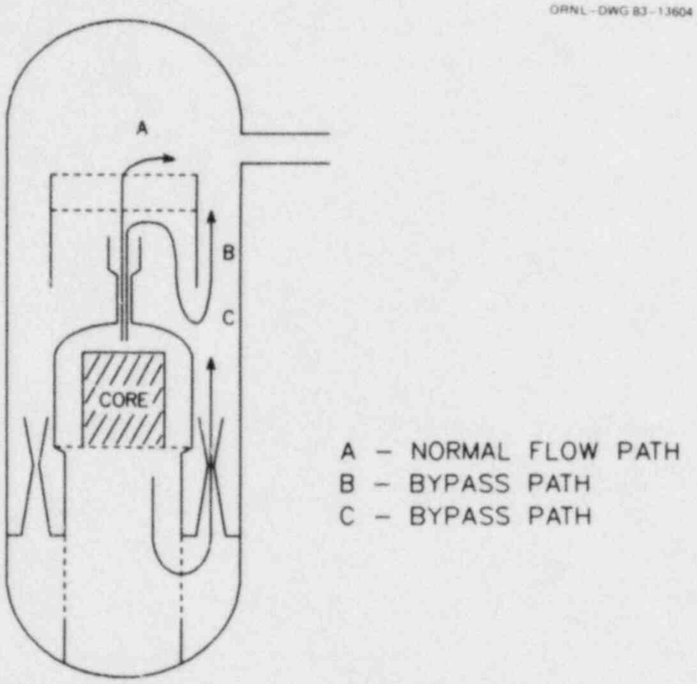

Fig. 7.5. BWR internal flow paths.

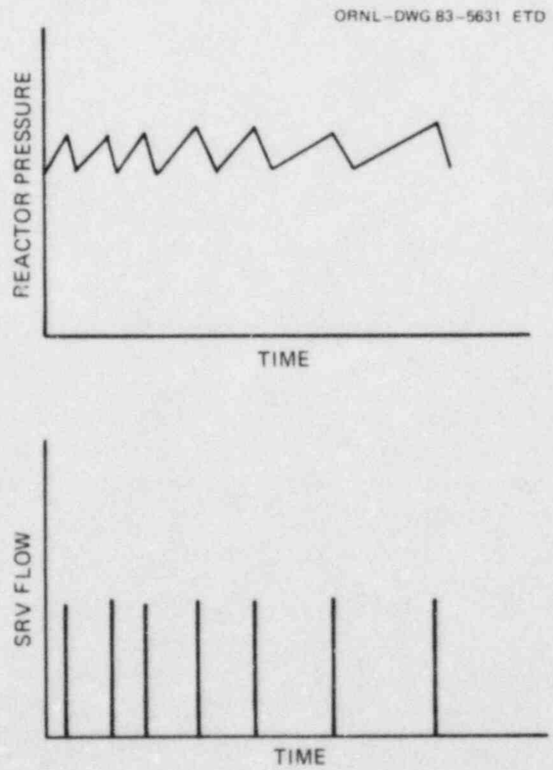

Fig. 7.6. Typical BWR vessel pressure and SRV steam flow histories for accidents with pressurized vessel.

8. BWR CONTAINMENT SEVERE ACCIDENT PHENOMENOLOGY AND MODELING CONCERNS

8.1 Background

The discussion in the remainder of this chapter will be best facilitated by first briefly describing the mass and energy flows from the reactor vessel into various containment compartments during the course of a severe accident.

If the accident does not involve a break in the reactor coolant system boundary or MSIV leakage, all material (steam, hydrogen, fission products) leaving the reactor vessel prior to vessel failure is routed to the pressure suppression pool via either the SRVs or HPCI/RCIC turbine exhausts. In the case of a pipe break accident, some or all of this material will be dumped into the compartment where the break occurred. In the case of MSIV leakage, material will be dumped to the main condenser or to the standby gas treatment system if a leakage control system is installed. Leakage from the main condenser to the local compartment atmosphere is possible if turbine gland sealing steam is not available. In addition to these mass flows, the hot reactor vessel and SRV pipe surfaces will heat the drywell atmosphere if the drywell coolers are not operating.

Subsequent to vessel head failure, core debris will drop onto the floor beneath the reactor and begin attacking the concrete. (In some MK II designs, some of the core debris could fall into the suppression pool via downcomers located directly beneath the vessel.) The attack of the concrete by the hot core debris will generate copious quantities of hot gases which are released directly into the drywell atmosphere. The core/concrete reaction will continue until a stable heat transfer geometry is reached or (in MK II designs) the floor fails, allowing the debris to drop into the suppression pool below. It is also possible that during this process the reactor pedestal could fail, allowing the vessel to drop down onto the drywell floor.

The scenario described above, together with the generic BWR containment systems described in Sect. 3.6, provides the basis for identifying the BWR containment systems, structures, and phenomena which should be modeled in realistic BWR severe accident analysis codes. Figures 8.1 through 8.5 illustrate the complex design and interaction of BWR containment systems. Tables 8.1 through 8.5 summarize the systems and structures depicted in Figs. 8.1 through 8.5. *These figures and tables form a concise set of general BWR severe accident modeling requirement guidelines, and future BWR severe accident analysis codes should incorporate models for each of the systems and structures contained in these tables and figures. The reader should consult Sect. 3.6 for an illustration of the application of Figs. 8.1 through 8.5.* The remaining sections of this chapter will discuss the systems and severe accident phenomena related to each of the BWR containment compartments.

8.2 Global Modeling Issues

The basic goal of containment modeling is to determine the timing, location, magnitude, and chemical form of fission product releases to the environment. This can only be accomplished by accurately quantifying the effects of various fission product transport and retention phenomena within each containment compartment. In the case of BWRs, both primary and secondary containment compartments must be analyzed. It will also be necessary to model the turbine building for accidents in which MSIV leakage is a concern.

Realistic evaluation of BWR containment geometries illustrated in Figs. 8.1 through 8.5 will require a code which is capable of treating both series and parallel inter-compartment flow paths. In general, the code must have the capability to track compartmental atmospheric and pool temperatures and pressures, atmospheric gas compositions and mass and energy flows into, from, and between compartments. Figure 8.6 is a schematic representation of the various BWR containments, showing possible inter-compartment flow paths which exist during normal plant operations and those which might be opened following a severe accident due to gross structural failure or actuation of blowout panels. In addition to these flow paths, failures in the reactor coolant system pressure boundary can result in leakage into both primary and secondary containment compartments. The structure and solution techniques employed in future severe accident analysis codes should accommodate such complex containment geometries.

8.3 Suppression Chamber Modeling

As described in Chap. 3, the design of BWR pressure suppression chambers varies significantly between the MK I, MK II, and MK III configurations. All suppression chambers, however, basically consist of a compartment which contains a pool of water. Table 8.5 is a composite listing (based on Tables 8.1 through 8.4) of systems, structures, and phenomena responsible for mass and energy transfer to, from, or within the various types of BWR suppression chambers. Each of the mechanisms in Table 8.5 must be modeled in a realistic BWR containment analysis code.

Almost all existing suppression chamber analysis codes treat the suppression pool as a well mixed single node. A review of Table 8.5 and Sect. 3.6 will reveal that the points of injection to and suction from the pool are widely distributed. Under certain accident conditions (such as a stuck open relief valve with no pool circulation or certain accidents in which the RHR system is inoperable), this design characteristic can lead to significant thermal stratification within the pool, resulting in localized pool boiling and pressurization of the containment before the bulk pool temperature reaches saturation. For cases such as this, a single node pool model is incapable of accurately predicting suppression chamber response.

Cook[8.1] has developed a distributed, lumped parameter MK I suppression pool model for discharge of a single SRV into an initially well mixed pool. This model could provide the basis for a comprehensive distributed node pool model. The CONTEMPT-LT code[8.2] incorporates a simple two region (liquid and vapor) wetwell model which incorporates many desirable characteristics-although the pool is modeled as a single node. The MARCH[8.3] suppression pool model also utilizes a single, well mixed node approach, though the MARCH models are significantly less advanced than the CONTEMPT-LT pool models. Since any distributed node pool model will undoubtedly require significantly longer computer execution time than will single node models, future severe accident analysis codes should provide the user a choice between single and multi-node pressure suppression pool representations.

A single node suppression chamber atmosphere model might be sufficient for most purposes, however, a multi-node atmosphere model would be desirable for analysis for localized hydrogen deflagration phenomena.

In the case of MK II plants, the pool model should accommodate direct interaction of pool water and core debris following vessel head failure (via vents under the reactor) and drywell floor melt-through. Existing containment models do not accommodate this phenomenon.

8.4 Drywell Compartment Modeling

Table 8.6 is a summary listing of mass and energy transfer mechanisms which can influence the response of the drywell compartment during a severe accident. A realistic BWR containment analysis code should have the capability to model these mechanisms.

Due to the relatively small size of MK I and MK II containments it is extremely important to accurately simulate the impact of reactor vessel and SRV line surface heat transfer, core/concrete reactions, and penetration leakage. These issues have been areas of major concern for BWR users of the MARCH code since MARCH typically predicts extremely high (>900 K) temperatures in the MK I drywell following the onset of core-concrete reactions.

Both MARCH and CONTEMPT model many of the mechanisms listed in Table 8.6. However, neither CONTEMPT-LT nor MARCH has any representation of hydrogen recombiners, reactor vessel surface heat transfer, hydrogen mixing systems, drywell purge systems, or containment atmospheric dilution systems. Many of the phenomenological models employed in CONTEMPT are more mechanistic than those employed in MARCH, however CONTEMPT-LT does not incorporate any treatment of core/concrete reactions or the gases produced by these reactions. Furthermore, neither CONTEMPT-LT nor MARCH incorporates models for drywell floor or reactor pedestal wall melt-through (both of which are major issues of concern for severe accidents in MK II plants).

8.5 Secondary Containment Modeling

The secondary containment compartments in present BWR designs are the reactor building reactor zone and refueling zone, the annulus compartment, fuel building, auxiliary building, and the enclosure building. These do not all exist in any single design.

Drywell or suppression chamber failure in the MK I and II designs will result in direct leakage from the primary containment into the reactor building reactor or refueling zone. In the standard MK III design, failure of the reactor building (annulus boundary) will result in leakage to the atmosphere, fuel building, or auxiliary building. Failure of the containment building in the alternate MK III design will result in leakage from the containment building to the enclosure or auxiliary buildings. Finally, failure of the MK I and II reactor building, or the MK III fuel, auxiliary, or enclosure buildings can result in direct leakage from these compartments to the surrounding atmosphere. This is particularly true in cases when SGTS failure results in positive gauge pressures in the containment. A BWR severe accident containment analysis code should be capable of simulating each of these failures and the resulting inter-compartment flows.

Table 8.7 is a listing of the mass and energy transfer mechanisms associated with each of these compartments. Due to the open nature of the reactor building refueling zone, the annulus, and the enclosure building, it is probable that a single volume modeling representation of these compartments would provide adequate accuracy in compartmental temperature and pressure predictions, etc. However, a multi-volume representation of the reactor building reactor zone and the auxiliary and fuel buildings is desirable due to the complex structural design of these buildings. As noted in Table 8.7, much of the ECCS equipment is located in rooms in the reactor and auxiliary buildings. Since high room temperature is an isolation signal for many of these systems, a multi-volume representation of these compartments is very desirable. Accurate evaluation of hydrogen deflagration events and fission product transport phenomena will, however, require a detailed treatment of atmospheric composition regardless of the modeling structure employed.

8.6 Fission Product Transport in Containment

8.6.1 Introduction

As previously stated, the basic purpose of all severe accident modeling is to determine the timing, magnitude, location, and chemical form of all fission products releases to the environment. The complex configuration of BWR containments greatly complicates this task. The purpose of this section is not to provide a detailed discussion of generalized fission product transport phenomena, but rather to highlight those areas where differences in BWR and PWR containment designs might dictate different fission product transport modeling approaches.

During a severe accident, fission products may escape the reactor vessel via the SRVs, MSIVs, pipe breaks, HPCI and RCIC steam supply lines, and, of course, vessel melt-through. Potential leakage paths also are present in the core spray, RHR, recirculation and scram discharge volume systems. (A detailed discussion of these fission product release pathways (for the Browns Ferry plant — BWR 4/MK I) is presented in Appendix A of Ref. 8.4.) In BWR accidents, the primary fission product release paths prior to vessel melt-through would be via the SRVs to the suppression pool or via a pipe break to one of the primary or secondary containment compartments. The fission products would be in the form of gases, vapors, and particulates, and would be waterborne when they enter the containment in some instances.

The natural fission product removal processes in the containment include

> ".... sorption and condensation onto surfaces and particles, condensation into aerosol particles, chemical reactions with surfaces and other species in the atmosphere, and dissolution in any water present. The natural removal processes undergone by particulate matter would include agglomeration into larger particles (by processes such as Brownian, gravitational, and turbulent coagulation) and subsequent removal of particulates from the containment atmosphere (by processes such as gravitational, diffusional, thermophoretic, and diffusiophoretic deposition. In all cases, processes would occur which would partially counteract some of these removal processes. For example, condensed vapors could be revaporated and deposited particles could be resuspended. Thus the concentration of materials in the containment atmosphere typically would be a complex function of the many processes which would take place. ... Inasmuch as most of the radionuclides other than the noble gases, and perhaps some of the halogens, could form aerosol particles in the containment, the post-accident behavior of the majority of radionuclides could be determined by the overall aerosol behavior."[8.5]

Accurate simulation of these processes will necessitate detailed models for atmospheric gas, steam and aerosol compositions and some simulation of the various chemical reactions which produce changes in chemical species compositions.

8.6.2 Suppression chamber fission product transport concerns

Since severe accidents would generally involve the release of fission products into the suppression pool and/or the drywell prior to gross primary containment failure, systems and structures which might significantly influence fission product retention in these compartments are of special interest. As previously stated, many accidents would result in the release of significant quantities of fission products into the suppression pool via the SRVs or possibly the drywell vents. These gaseous and particulate materials would be carried in flows of steam and noncondensable gases. The suppression pool can remove and retain significant quantities of fission products from the SRV and vent inputs, but the effectiveness of the pool scrubbing function is a complex, and as yet not well understood, function of pool temperature, depth, pressure, and material flow rates and compositions (fraction of noncondensable gases). Heating of the pool due to the deposited fission products should be modeled. An additional complication is the fact that considerable fractions of the pool may flash (and subsequently boil) when the primary containment fails, resulting in resuspension of significant quantities of materials which had previously been deposited in the pool. Operation of suppression chamber RHR sprays, plateout of material on vent/pool boundary structures and flow of material through the vents, and reactor building-to-wetwell and wetwell-to-drywell vacuum breakers also must be considered in mechanistic fission product transport calculations.

Few physical models for fission product retention in the pool currently exist, but relatively good models can be formulated for SRV, vent, vacuum breaker, and standby gas treatment system flows. A reasonable approach to this problem at the current time is to utilize experimentally based, time dependent pool decontamination factors (see Sect. 4.5, Ref. 8.4) which would vary, based on the chemical and physical form of the incoming flow (particulates, organic iodine, etc.), the source of the incoming flow (SRV quenchers or vents), and the physical state of the pool (subcooled or saturated). Simplistic models of this type are described in Appendix E of Ref. 8.6.

Rastler[8.7] has summarized available suppression pool decontamination experimental results. These data could provide the basis for establishment of appropriate decontamination factors for present applications.

8.6.3 Drywell fission product transport concerns

Prior to vessel melt-through, fission products may enter the drywell via a reactor coolant boundary pipe break, or via the drywell vent and vacuum breaker system. Subsequent to vessel failure, the core/concrete reaction will generate tremendous quantities of noncondensable gases and aerosols which will disperse directly into the drywell atmosphere. All of the basic fission product transport phenomena described in Sect. 8.6.1 will occur. In addition to the natural phenomena, BWR severe accident analysis codes should have the capability to simulate the impact of drywell fan coolers, sprays, SGTS, H_2 mixing, and purge system operation on the time dependent distribution of fission products within the drywell. The SGTS, H_2 mixing and purge systems basically impact the drywell fission product distribution by removing portions of the drywell atmosphere. The drywell fan coolers condense steam and directly remove fission products by plateout and deposition on the cooling coils. The drywell spray system condenses steam and washes aerosols out of the drywell atmosphere. All of these phenomena can significantly affect the time dependent distribution of fission products in the drywell.

8.6.4 Primary containment failure — impact on fission product distribution

If the primary containment is intact prior to vessel melt-through, evolution of gas from the core/concrete reaction will result in pressure and temperature increases in the drywell that will eventually challenge the integrity of the containment. The timing, location, and nature of the primary containment failure can significantly impact the ultimate distribution of fission products following a severe accident.

The four basic types of primary containment failure mechanisms are static overpressurization of the containment resulting in (1) liner rupture, or (2) penetration leakage due to differential expansion between the penetration assemblies and the liner: (3) degradation of the elastomer seals in primary containment electrical penetration assemblies due to excessive containment temperatures, and (4) primary containment basemat melt-through. Due to their small size, MK I and MK II containments would generally fail by one of the first three mechanisms prior to basemat melt-through. It is unclear whether MK III systems would behave in the same manner.

Greene (Chap. 7, Ref. 8.1) has reviewed and summarized existing work on BWR severe accident static overpressurization containment failure analysis for MK I containments. Historically, only shell failure modes (type 1 above) have been analyzed even though most investigators have noted that penetration leakage (type 2 failure) is likely prior to gross containment failure to by overpressurization.

Available studies[8.8,8.9] indicate that the MK I containment liner would fail at the intersection of the spherical and cylindrical sections in the drywell at gauge pressures ranging between 0.81 and 1.22 MPa (117 and 177 psig). MK II containment failure would occur in the upper third of the suppression pool liner at a gauge pressure of ~0.92 MPa (~133 psig). Failure of the standard MK III containment liner would occur near the junction of the hemispherical head and the cylindrical wall, at a gauge pressure of ~0.69 MPa (~100 psi).

None of the available studies have made any definitive statements regarding the size of such liner failure opening since existing analytical methods are incapable of addressing such issues.

Existing structural mechanics models are not sufficiently refined to allow a detailed analysis of type 2 failures on a scale that would be practical for application to BWR severe accident analysis codes. It is possible, however, that failures of this type could forestall or perhaps even circumvent gross containment liner (type 1) failures.

Excessive temperatures may be reached in the containment due both to the evolution of hot gases and aerosols from the core/concrete reaction and to direct radiation to the containment liner and other structures from the hot core debris. Excessively high temperatures could result in the degradation of elastomer seal material employed in containment electrical penetration assemblies.[8.10,8.11] Such degradation might allow the primary containment atmosphere to leak directly into the reactor building in the MK I and MK II designs. It is possible, however, that such leakage paths would be partially or even totally plugged due to aerosol deposition along the flow path.[8.12] (It should be noted that such plugging would not necessarily yield reduced fission product releases since it might result in a violent static-overpressurization failure of the containment boundary.)

A sufficiently large failure in the drywell (MK I and II) or containment building (MK III) liner would result in a depressurization of the associated compartment. At some point, the containment pressure could drop below the saturation pressure of the suppression pool, resulting in pool boiling and an attendant redistribution of fission products from the pool. If the liner fails in a gross (type 1) fashion, tremendous quantities of pool water would flash and boil in a very brief time. In the MK I and II designs this steam would flow through the drywell vacuum breakers and vents, through the drywell, and out the failure openings. In the MK III design suppression chamber, steam could flow directly out any breach in the containment building wall. Depending on the location and configuration of the containment breach, material could flow directly into the reactor building reactor and/or refueling zone (MK I and II), the annulus (standard MK III), or the fuel and auxiliary building (alternate MK III). The steam could, of course, carry substantial quantities of fission products.

Due to the great uncertainty currently associated with containment failure mechanisms, a reasonable approach to the modeling of these phenomena in BWR severe accident analysis codes is to allow the user to specify the time, pressure, or temperature and location at which the containment failure will occur. The initial size of the failure would also be input. Existing information[8.12] relating to aerosol plugging rates could be incorporated to account for the effective time-dependent decrease in failure size due to these phenomena.

8.6.5 Secondary containment fission product transport concerns

As previously described, the secondary containment in MK I and MK II designs is comprised of the reactor and refueling zones of the reactor building. The secondary containment in the standard MK III design consists of the reactor, fuel, and auxiliary buildings, and the enclosure and auxiliary buildings in the alternate MK III design.

In addition to the natural fission product transport and deposition processes discussed in Sect. 8.6.1, realistic BWR fission product transport analysis codes must be capable of considering the impact of BWR secondary containment systems which might continue to operate during a severe accident. These systems include reactor building-to-suppression chamber vacuum breakers, annulus-to-containment building vacuum breakers, reactor and refueling zone blowout panels, standby gas treatment systems, reactor enclosure recirculation systems, and fire protection system sprays.

In many accidents in MK I and MK II plants, failure of the primary containment could lead to failure (opening) of the reactor building reactor zone-to-refueling zone and refueling zone-to-atmosphere blowout panels, i.e., loss of secondary containment integrity. Such phenomena must be modeled accurately in fission product transport codes.

Standby gas treatment systems draw suction from the secondary containment volumes during accident conditions. Operation of these systems can significantly impact fission product release magnitudes due to removal of aerosols by the HEPA filters and removal of iodine by the charcoal beds. These phenomena must be modeled in realistic BWR severe accident analysis codes. A second impact of SGTS operation which is often over-looked is that, in some plants, the SGTS capacities are sufficiently large to maintain negative pressures in the reactor building even after the reactor building blowout panels open.[8.4] The net impact of this is to significantly decrease the magnitude of direct fission product releases to the environment, allowing more time for natural deposition processes to occur within the containment. It should be noted, however, that continued operation of SGTS systems under core melt (high aerosol loading) conditions could result in tearing of the SGTS HEPA filters. This tearing would allow aerosols to penetrate into and possibly deposit in the SGTS charcoal beds. The decay heat source from any aerosols retained in the bed could be significant and might even result in ignition of the charcoal.

Some MK II plants incorporate a reactor enclosure recirculation system (Sect. 3.5.3). This system can draw suction from either the reactor or refueling zone of the reactor building, filter the atmosphere through charcoal and HEPA filters, and return the cleaned gas to either zone of the reactor building. This system would function as a fission product trap during accidents in which it remains functional. However, the survivability of the RERS under the extreme environmental conditions produced during severe accidents is questionable.

Although not a part of the secondary containment system, fission product transport phenomena in the main turbine building must be modeled for cases involving fission product leakage into the building via the main steam lines and turbine gland seals.

Chapter 8 References

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Table 8.1. MK I systems and structures[a]

	Q_{in}	Q_{out}	M_{in}	M_{out}
Drywell	Recombiners	DC	DV	DV
	RV	DS	Sprays	SGTS
			VB	
			Break flow	
			CAD	
Wetwell		RHR	HPCI/RCIC	VB
			Turbine exhaust	ECCS suction
			SRVs	
			Sprays	
			VB	
			DV	
RB/RXZ	ECCE	RC	Sprays	BP
			Break	VB
				SGTS
RB/RFZ			Sprays	SGTS
			BP	BP

[a]
RV Reactor Vessel
DS Drywell Structures
DV Drywell Vents
DC Drywell Coolers
VB Vacuum Breakers
CAD Containment Air Dilution System
ECCE Emergency Core Cooling Equipment
RC Emergency Core Cooling Equipment Room Coolers
BP Blowout Panels
RB/RXZ Reactor Zone of Reactor Building
RB/RFZ Refueling Zone of Reactor Building

Table 8.2. MK II systems and structures[a]

	Q_{in}	Q_{out}	M_{in}	M_{out}
Drywell	Recombiners RV	DC DS	Sprays DV Break flow CAD VB	DV SGTS
Wetwell		RHR	HPCI/RCIC Turbine exhaust SRV Sprays DV	ECCS suction
RB/RXZ	RERS ECCE	RERS RC	RERS SGTS Sprays Break	RERS SGTS BP
RB/RFZ	RERS	RERS	RERS SGTS	RERS SGTS

[a]
RV Reactor Vessel
DS Drywell Structures
DV Drywell Vents
DC Drywell Coolers
VB Vacuum Breakers
CAD Containment Air Dilution System
ECCE Emergency Core Cooling Equipment
RC Emergency Core Cooling Equipment Room Coolers
BP Blowout Panels
RB/RXZ Reactor Zone of Reactor Building
RB/RFZ Refueling Zone of Reactor Building
RERS Reactor Enclosure Recirculation System

Table 8.3. Standard MK III systems and structures[a]

	Q_{in}	Q_{out}	M_{in}	M_{out}
Drywell	RV Recombiners	DRS DS	RHR/sprays Break flow VB H_2 mix sys. DV	DV SGTS
Containment		RHR	VB DV RHR/sprays RHR/flood RCIC turbine exhaust SRV	SGTS H_2 mix sys. Purge sys. ECCS suction
Annulus	Conduction through cont. walls		Drywell purge	SGTS VB
Fuel Bldg			Fire sprays	SGTS
Auxiliary Bldg	ECCE	RC	Fire sprays	SGTS

[a] RV Reactor Vessel
DS Drywell Structures
DV Drywell Vents
DC Drywell Coolers
VB Vacuum Breakers
DRS Drywell Recirculation System
CAD Containment Air Dilution System
ECCE Emergency Core Cooling Equipment
RC Emergency Core Cooling Equipment Room Coolers
BP Blowout Panels

Table 8.4. Alternative MK III systems and structures[a]

	Q_{in}	Q_{out}	M_{in}	M_{out}
Drywell	RV Recombiners	DC DS	RHR/sprays VB DV Break flow	DV SGTS DP
Containment		CC	RHR/sprays DV RCIC turbine exhaust	SGTS ECCS suction
Auxiliary Bldg		RC	Fire sprays	SGTS
Enclosure Bldg			SGTS	SGTS

[a] RV Reactor Vessel
DS Drywell Structures
DV Drywell Vents
DC Drywell Coolers
VB Vacuum Breakers
DRS Drywell Recirculation System
CAD Containment Air Dilution System
ECCE Emergency Core Cooling Equipment
RC Emergency Core Cooling Equipment Room Coolers
BP Blowout Panels
DP Drywell Purge
CC Containment Cooling

Table 8.5. BWR suppression chamber mass and energy transfer mechanisms

Region	Mechanism	Direction
Liquid		
Mass transfer:	SRVs	+
	Drywell vents	+,-
	ECCS turbine exhaust	+
	Sprays + atmospheric fallout	+
	Condensaion on structures	+
	Pipe break	+
	Evaporation	-
	Boiling	-
	Flashing	-
	ECCS pump suction	-
	RHR drain pump suction	-
	Structural failure	-
	RHRS (Mixing)	
	Drywell floor melt-through (MK II)	+
Heat transfer:[a]		
	Structures	+,-
	RHRS	-
	Sensible heat to compartment atmosphere	+,-
	Fission products	+
Vapor		
Mass transfer:	Sprays	+
	Vacuum breakers	+
	SGTS	-
	H_2 mixing system	+
	Purge system	-
	Penetration leakage	-
	Structural failure	-
	H_2 reactions	+,-
Heat transfer:	Structures	+,-
	Containment coolers	-
	Sensible heat to pool	+,-
	H_2 reactions	+

[a]Without mass transfer.

Table 8.6. BWR drywell mass and energy transfer mechanisms

Atmosphere	Mechanism	Direction
Atmosphere		
Mass transfer:	Sprays	+
	Vacuum breakers	+
	Containment atmosphere	+
	Dilution system	+,-
	H_2 mixing system	+
	Vents	-
	SGTS	-
	Penetration leakage	-
	Structural failure	-
	Drywell purge system	-
	Hydrogen recombiners	+,-
	Condensation	-
	Fallout	-
	Sump evaporation	+
	H_2 reactions	+,-
	Core/concrete reactions	+
Energy transfer:[a]	Reactor vessel and SRV lines	+
	Sensible heat from sump	+,-
	Structures	+,-
	Fan coolers	-
	Hydrogen recombiners	+
Sump		
Mass transfer:	Condensation	+
	Sprays and atmospheric fallout	+
	Evaporation	-
	Boiling	-
	Flashing	-
	Overflow to PSP	-
	Sump pumps	-
	Structural failure	-
	Floor melt-through (MK II)	-
Energy transfer:	Core debris reactions	+
	Fission products	+
	Sensible heat to atmosphere	+,-

[a]Without mass transfer.

Table 8.7. BWR secondary containment mass and energy transfer mechanisms

Region	Mechanism	Direction
Reactor Building Reactor Zone (MK I, II)		
Mass transfer:	Fire sprays	+
	Blowout panels	-
	Reactor coolant boundary breaks	+
	SGTS	-
	Infiltration	+
	Primary containment penetration leakage	+
	Primary containment failure	+
	Vacuum breakers (MK I only)	-
	Exfiltration	-
Energy transfer:	Equipment (ECCS) waste heat	+
	Structural	+,-
	Emergency core cooling	-
	Equipment room air coolers	+
Reactor Building Refueling Zone (MK I, II)		
Mass transfer:	Fire sprays	+
	Blowout panels	+,-
	SGTS	-
	Infiltration	+
	Exfiltration	-
Energy transfer:	Structural	+,-
Annulus		
Mass transfer:	Drywell purge	+
	Vacuum breakers	-
	SGTS	-
	Structural failure	+,-
	Annulus mixing system	Mixing
Heat transfer:	Structural	+,-
Auxiliary Building		
Mass Transfer:	Fire sprays	+
	Annulus failure	+,-
	Pipe break	+
	SGTS	-
Energy transfer:	Structural	+,-
	Equipment room air coolers	-
	ECCS equipment waste heat	+
Fuel Building		
Mass transfer:	Fire sprays	+
	SGTS	-
	Annulus failure	+,-
Heat transfer:	Structural	+,-
Enclosure Building		
Mass transfer:	Annulus failure	+,-
	Infiltration	+
	Exfiltration	-
	SGTS	-
Heat transfer:	Structures	+,-

[a]Without mass transfer.

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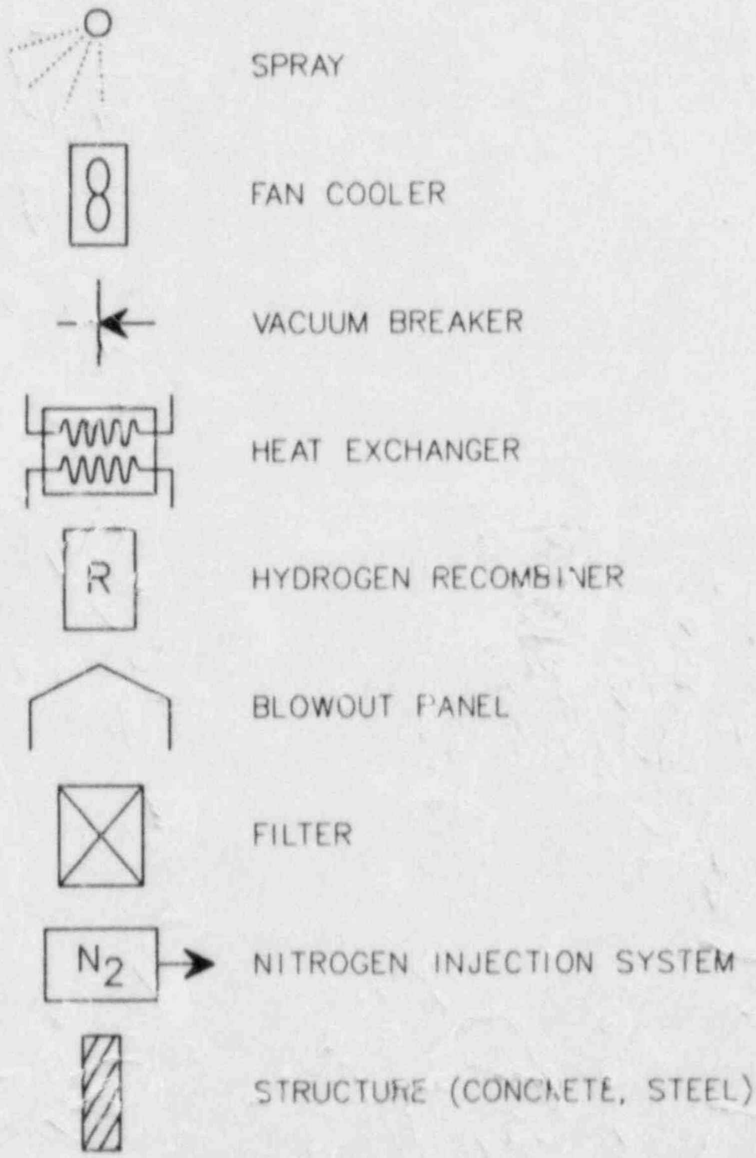

Fig. 8.1. Symbols employed in Figs. 8.2–8.5.

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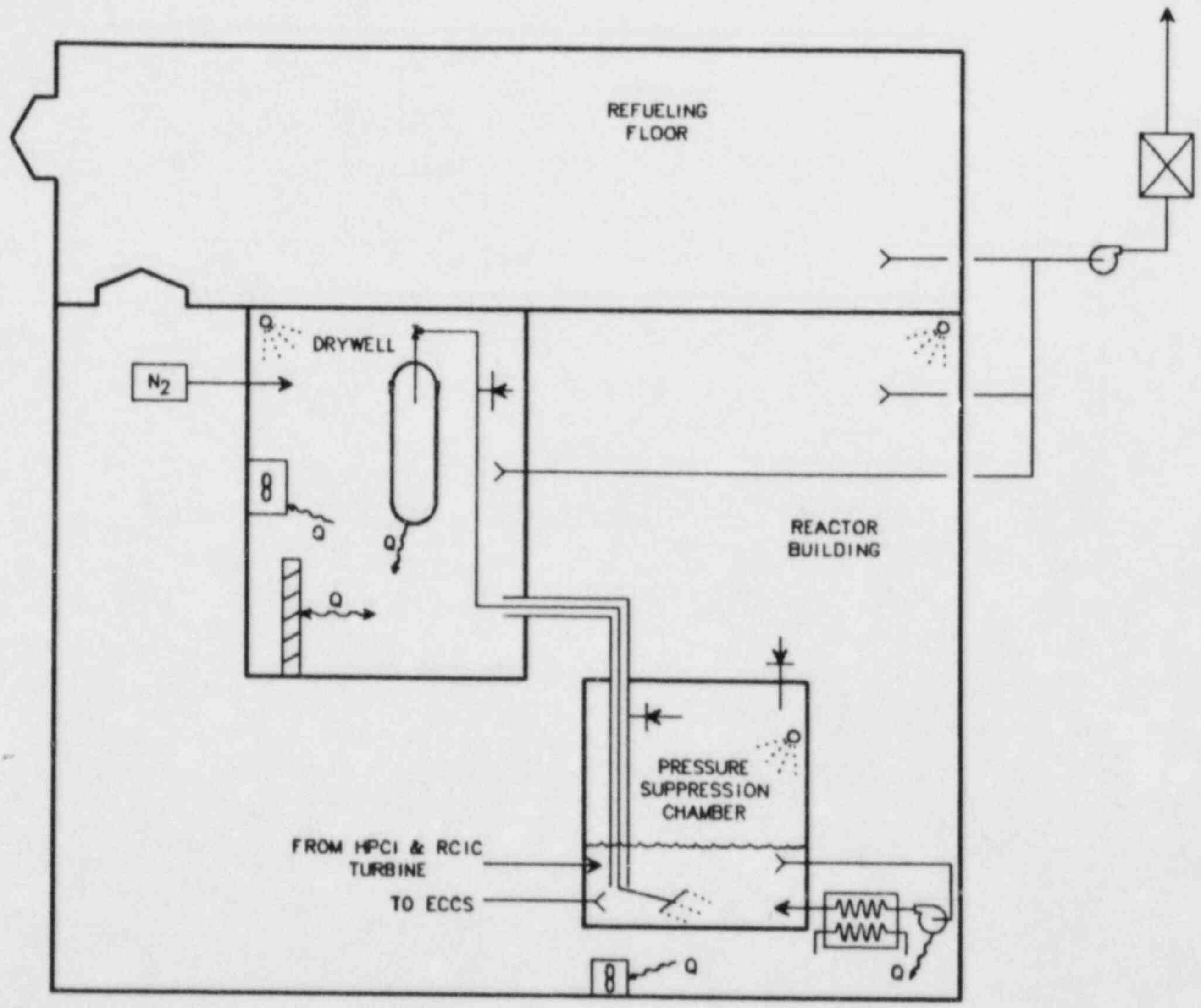

Fig. 8.2. Generic MK I containment systems and structures.

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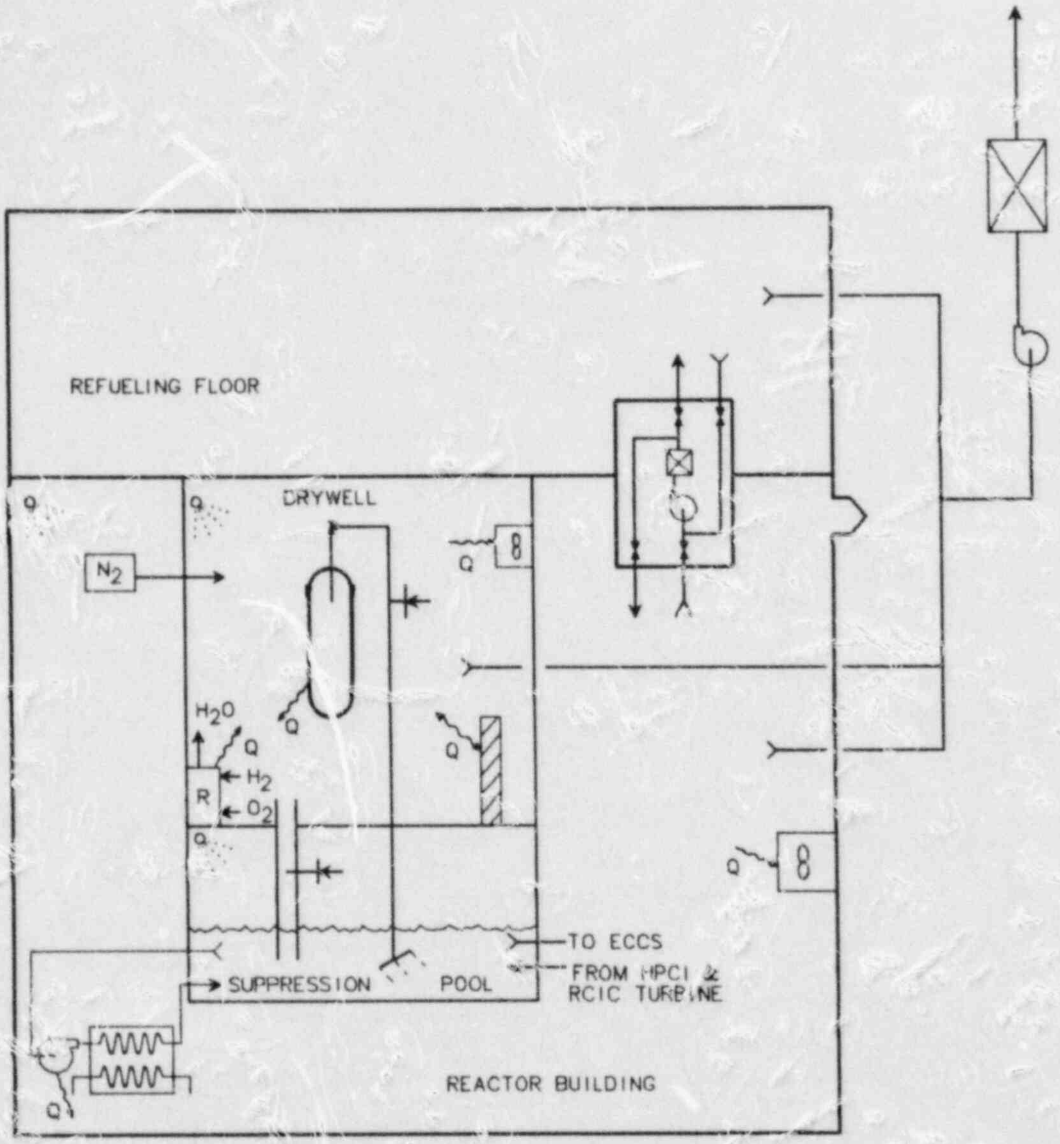

Fig. 8.3. Generic MK II containment systems and structures.

ORNL-DWG 83-12784

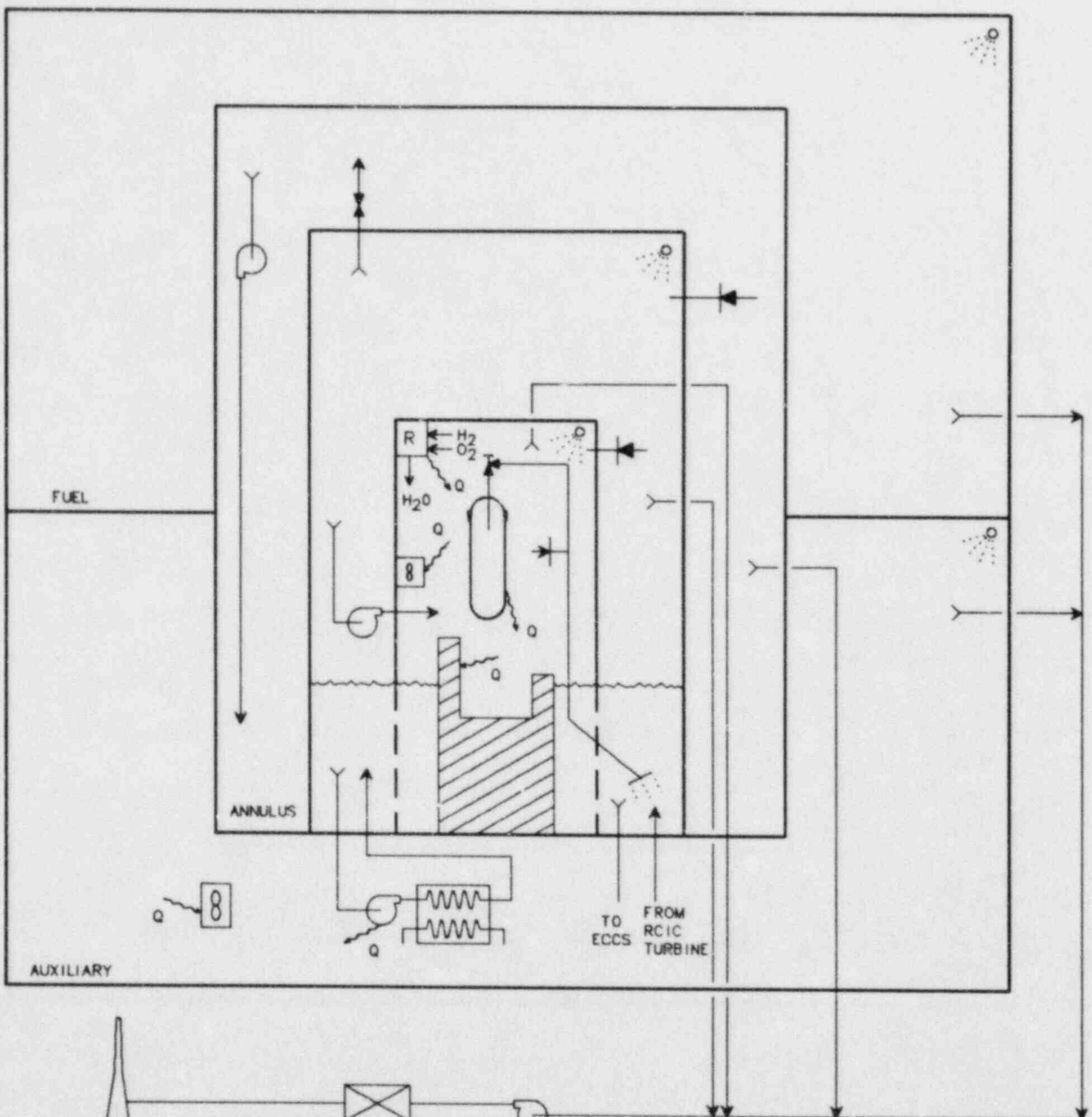

Fig. 8.4. Standard MK III generic containment systems and structures.

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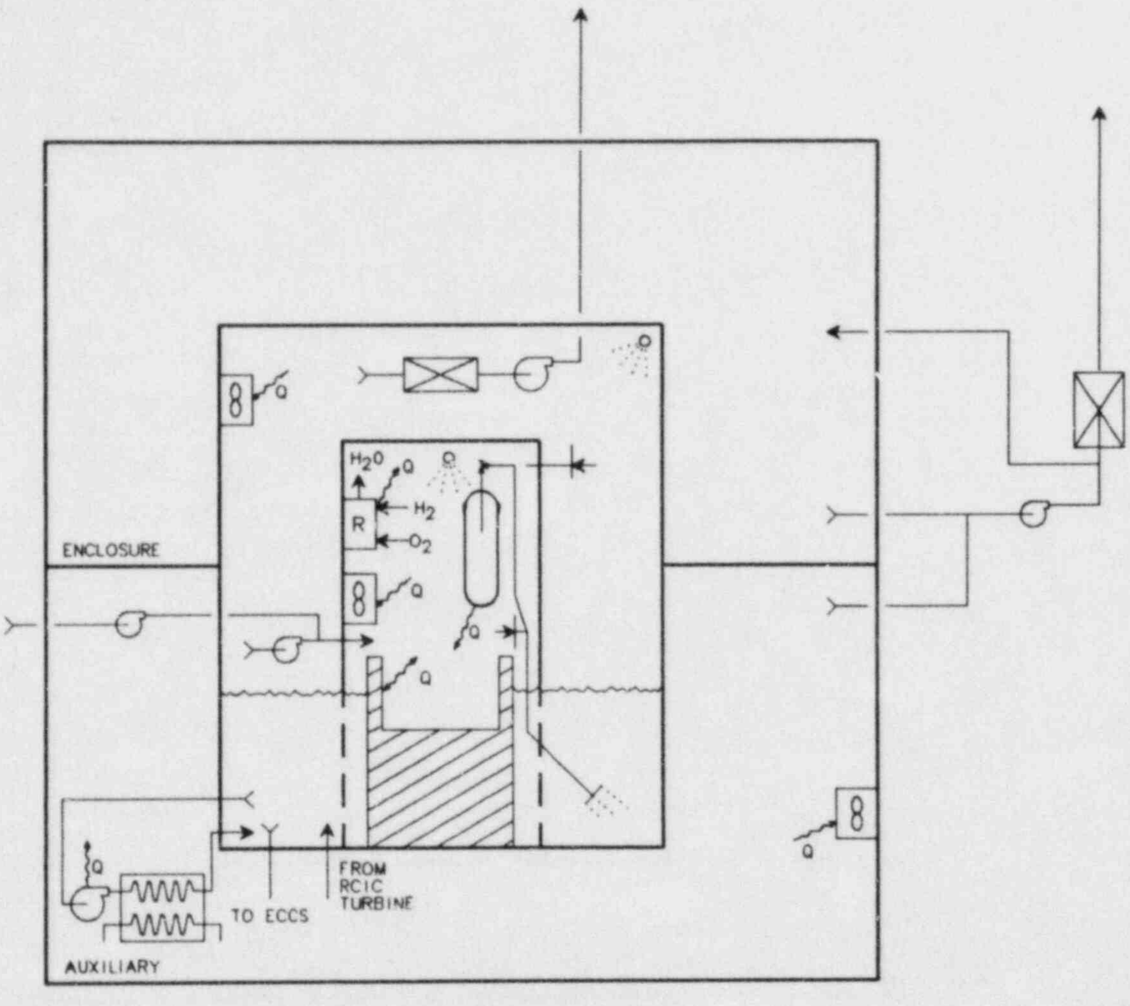

Fig. 8.5. Alternative MK III generic containment systems and structures.

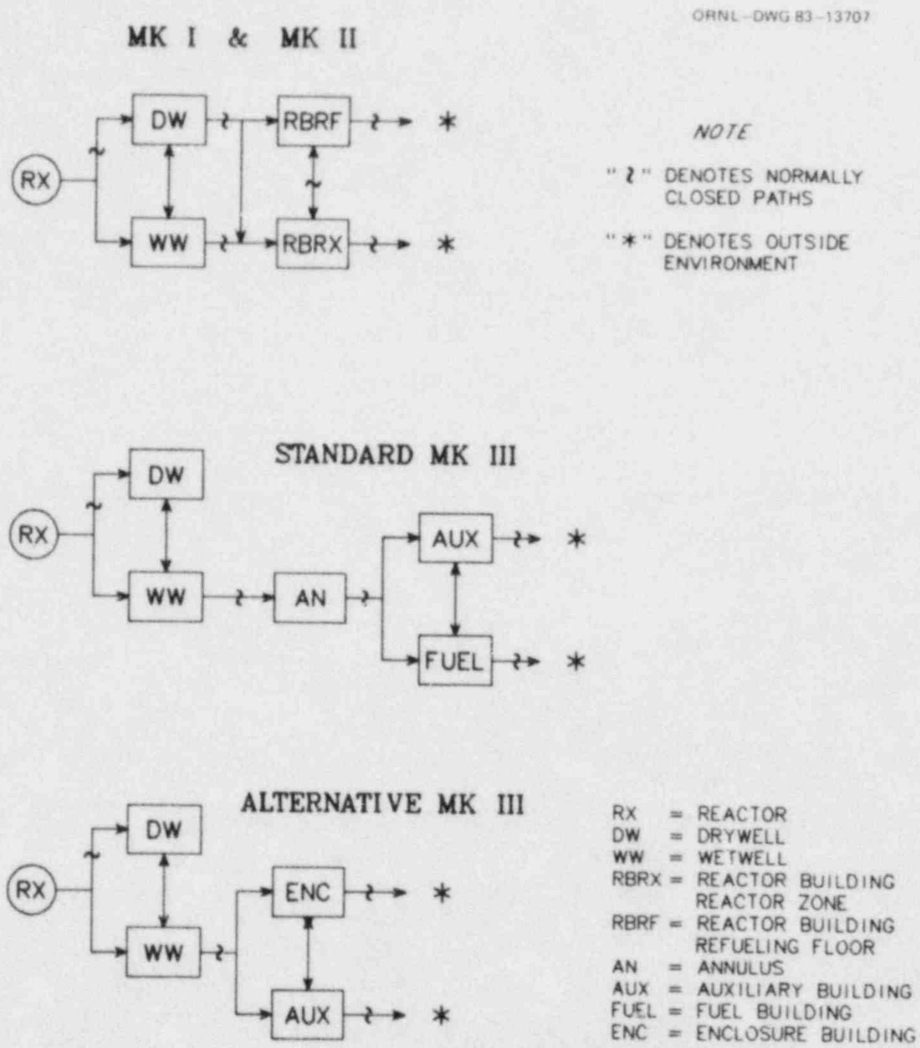

Fig. 8.6. BWR containment flow paths.

9. HUMAN FACTOR MODELING CONSIDERATIONS

The accident at Three Mile Island[9.1] served as a forceful reminder of the importance of the operator's role in the mitigation or exacerbation of accidents in nuclear power plants. The key factor in determining the operator's actions at any point in an accident is the "apparent" plant status (which may be different than the actual plant status) displayed to the operator by the available control room instrumentation. It is, therefore, desirable that the severe accident analyst have the capability of determining what information would be available to the control room operator throughout the course of the accident.

There are two factors which influence the quality of the information available to the operator via the control room instrumentation: (1) non design basis environmental conditions in the vicinity of the sensors which can result in erroneous instrumentation readings, and (2) limits to the physical range of the instrumentation.

An understanding of the first factor can be gained by reviewing the operating principle of BWR in-vessel water level instrumentation systems. The measured water level is the level existing in the reactor downcomer annulus. To measure this water level, two connections are made to the reactor vessel. One connection penetrates the reactor vessel in the steam volume area. This high pressure side penetration connects to a condensing chamber which is an enlarged volume in the piping. This chamber is not thermally insulated and remains at approximately the same temperature as the surrounding dry-well atmosphere. Most of the reference leg piping in BWR 5 and 6 plants is routed out of the drywell into the containment to minimize density changes because of changing drywell temperature). Steam entering the condensing chamber condenses on the inside of the chamber. The resulting condensation collects in a reference leg which connects to one side of a level transmitter.

The lower penetration (variable leg) enters the reactor vessel in the downcomer annulus region. This line connects to the low pressure side of the level transmitter. With this arrangement, reactor pressure is felt on both sides of the differential pressure detector and does not affect the measurement. The pressure caused by the reference column of water is compared to the pressure resulting from the water level inside the reactor vessel. Since the reference leg remains constant (under design conditions), because of the action of the condensing chamber, any change in the height of the reactor vessel water level produces a difference in the water column pressures that is proportional to the reactor vessel water level.

The level transmitter converts this differential pressure signal to an electrical signal and transmits it to a control room indicator, a protection or isolation system trip channel, or an alarm trip signal. The instrument is calibrated to read maximum level when zero differential pressure is applied.

This type of level measurement system makes no correction for changes in reactor vessel or reference leg water temperature or density, and is termed uncompensated. *Each level detector is calibrated for a given temperature of the reactor coolant and reference leg. Any deviation from these conditions introduces errors in the level measurement because of changes in the water density.* Each instrument is calibrated at the vessel pressure and drywell temperature in which the instrument is to be used.

Compensated level detector systems operate with a reference leg temperature maintained near the reactor temperature to help compensate for density changes. In this type of system, the reference leg tends to flash to steam during large pressure reduction transients in the reactor vessel. While this condition is only temporary, the indicated level appears higher than actual level during the transient period.

Many severe accidents would result in drywell temperature and pressure extremes and reactor vessel depressurization transients which could yield significant errors in the indicated reactor vessel water level. As shown in Fig. 9.1, most BWRs have only one level instrumentation system which is capable of generating vessel water level readings when the level is below the top of the active fuel. This system, which is designed to monitor reactor vessel level during design basis loss of coolant accidents, is calibrated for saturated water and steam conditions of 212°F inside the reactor vessel and the drywell. This system would yield erroneous core water level indications during many severe accident conditions. Similar limitations exist for the remaining level indication systems. The different calibration points of the level indication systems would also lead to contradictory level readings during many severe accident conditions. This is a particular concern for BWR-4 systems in which the reference leg piping is located within the drywell.

The second factor noted above is that the indication range of much of the control room instrumentation is such that readings would be off scale during many phases of a severe accident. During such situations operators would be forced to make decisions and take actions based on incomplete, and, perhaps, inaccurate information.

An accurate assessment of the operator's role in severe accident management cannot be accomplished without consideration of the issues discussed above. Future severe accident analysis codes should incorporate an "interpretive" algorithm which would be capable of converting predicted values of critical plant parameters to values which would be displayed to operators in the control room.

Table 9.1 is a suggested list of parameters for inclusion in an interpretive processor. Such an algorithm would be relatively easy to design and code, and would utilize relatively meager computer resources. The information generated by such a processor, when combined with prepared emergency procedure guidelines[9.2] would provide the analyst with a powerful tool for investigation of the operator's role in severe accident mitigation.

Chapter 9 References

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Table 9.1. Interpretive parameters

Reactor vessel level (all indicating systems)
Reactor pressure
Drywell pressure
Drywell atmospheric temperature
Wetwell (containment building) pressure
Pressure suppression pool temperature
Pressure suppression pool level
Condensate storage tank level

ORNL-DWG 83-5632 ETD

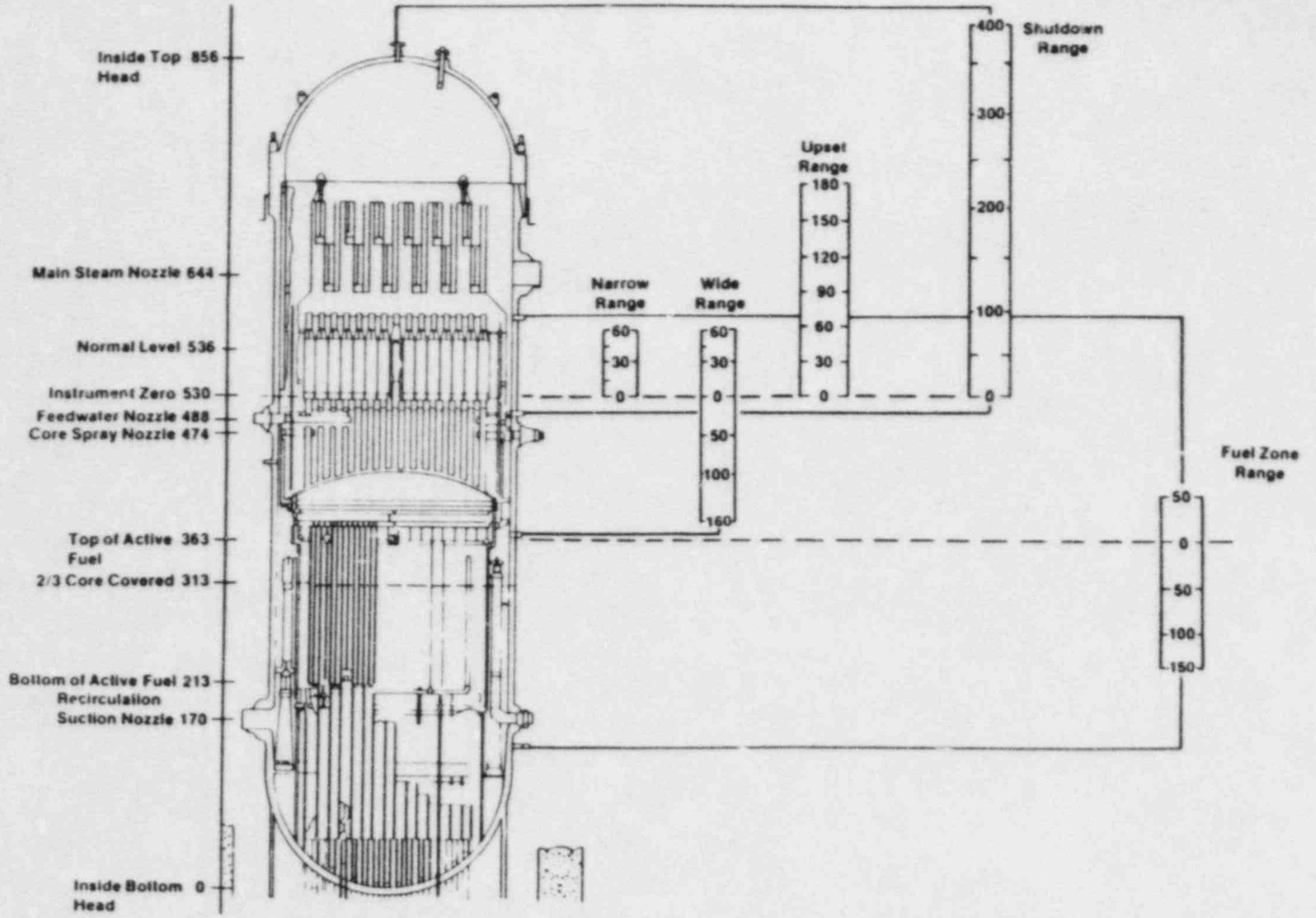

Fig. 9.1. Vessel level instrument ranges.

10. ATWS MODELING ISSUES

10.1 Introduction

The previous chapters of this report have described a wide variety of modeling capabilities necessary for realistic simulation of a broad spectrum of BWR severe accidents. As mentioned in Chapt. 4, the results of existing BWR probabilistic risk assessments indicate that the four "risk dominating" sequences for BWRs are (1) transients coupled with failure to provide makeup water to the reactor, (2) transients accompanied by loss of containment cooling, (3) loss of coolant accidents, and (4) transients coupled with failure to achieve reactor subcriticality (ATWS). It is the author's belief that the first three types of accidents can be adequately modeled by a code having the capabilities previously described in this report. The purpose of this chapter is to briefly summarize the additional modeling capabilities necessary for acceptable representation of ATWS events.

Section 10.2 presents a brief description of the plant response to an event in which the reactor fails to scram when initially called upon, and of subsequent automatic and operator initiated actions which would result in accident mitigation. Section 10.3 describes situations in which failure of mitigative actions would lead to core melting. Finally, Sects. 10.4 and 10.5 describe the additional severe accident modeling capabilities necessary for evaluation of ATWS accident phenomena in the reactor vessel and containment.

10.2 Mitigated ATWS Sequence Description

10.2.1 Introduction

A recent evaluation performed by the General Electric Co.[10.1] reveals that the two major transient without scram sequences of concern in BWRs are (1) ATWS/MSIV closure, and (2) ATWS/turbine trip.

The ATWS/MSIV closure sequence is an accident in which the reactor is isolated and no feedwater flow is available (for plants with turbine driven feed pumps). This is the most limiting of the two transients in that higher vessel pressures and pressure suppression pool temperatures are expected than in the other transient.

The ATWS/turbine trip transient differs from the previous transient in that the MSIVs are open and feedwater flow is available during the early part of the transient. The main condenser is, therefore, available for steam discharge via the turbine bypass valves during a portion of the accident. Peak reactor pressures are calculated to be lower than those for the MSIV closure accident.[10.1] It should be noted that the turbine trip ATWS converts to a MSIV closure ATWS prior to core uncovery, since the MSIVs automatically close on receipt of a low reactor water level signal.

The following section will present a brief description of the ATWS/MSIV closure accident in which normal mitigation procedures result in accident termination prior to core damage. This description is presented to afford the reader an improved visualization of ATWS mitigation procedures.

10.2.2 ATWS/MSIV closure transient description

The MSIV closure accident results in the greatest challenge to the reactor coolant pressure boundary. Following MSIV closure, reactor vessel pressure immediately increases, producing a reduction in core void fraction and a rapid increase in power. Feedwater flow is lost due to the feedwater pump trip associated with MSIV closure. Reactor power is expected to peak at approximately 525% of the initial value at 4 s into the transient.

The pressure spike associated with MSIV closure produces SRV actuation and recirculation pump trip. Recirculation pump trip (which is vital for ATWS mitigation) reduces core flow which, together with loss of feedwater flow, results in increased core voiding and substantially reduced core power levels. Continued SRV actuation results in a steady reduction in vessel water level, triggering HPCI and RCIC injection at ~1 min into the accident. Reactor vessel water level would continue to drop for some period since the combined flow of the HPCI and RCIC systems is insufficient to replace SRV losses at this point in the accident.

The operator should attempt to manually scram the reactor following indication of automatic scram failure. Assuming that the manual scram function is unavailable, the GE Emergency Procedure Guidelines (EPGs)[10.2] call for the operator to manually insert the control rods. (This is a slow process since only one rod can be driven at a time.) The operators are to initiate the SLC system if the reactor is not shut down before the pressure suppression pool reaches 110°F.

If the pressure suppression pool reaches 110°F with reactor power greater than 3% and SRVs actuating, the EPGs require that the operator stop all injection except that from the CRD hydraulic and SLC systems until the power level drops below 3%, or the level drops to the top of the active fuel, or all SRVs close with a drywell pressure less than 2.5 psig. Thereafter the operators are to prevent automatic ADS operation and maintain the level near the top of the active core until the reactor can be brought to a safe shutdown state via rod insertion or SLC system operation. In the event the vessel water level cannot be maintained above the top of the core, the reactor would be depressurized to allow use of low pressure injection systems.

10.3 ATWS Degraded Core Progression

Regardless of the transient initiator, current BWR ATWS emergency procedure guidelines rely upon a combination of recirculation pump trip, manual rod insertion, SLC injection, and lowering of vessel water level to control or terminate the accident. Successful SLC injection will result in power levels of a few percent within 20 min of the start of injection. Recirculation pump trip alone will result in power levels below 30% — even if the SLC system is unavailable. Reduction in water levels to near the top of the active fuel will result in additional reductions in power level due to increases in core voiding.

Preliminary results from an ongoing BWR ATWS study at Oak Ridge National Laboratory[10.3] indicate that, properly executed, the current EPGs will result in *termination* of the ATWS sequence within 20 min *if the SLC system functions as designed*. In the event that the operator cannot or does not manually scram, insert rods, or utilize the SLC system, the mitigating actions described in the current EPGs will result only in "arrest" of the ATWS sequence. For such cases, execution of the EPGs will result in either of two conditions: (1) reactor power of a few percent of the initial value and vessel water level stabilized near the top of the active fuel, with injection via HPCI/RCIC systems; or (2) power status unknown, vessel water level below the top of the active fuel and dropping, injection via HPCI/RCIC systems, and depressurization underway. It is these two EPG exit conditions that are of interest to the severe accident analyst.

In the first case, the reactor is in a quasi-steady state situation in which reactor power is stabilized at a few percent of the initial value, vessel water level has stabilized near the top of the fuel, and the HPCI/RCIC system is supplying the necessary vessel makeup water from the condensate storage tank (CST). The pressure suppression pool is heating up due to steam input from the lifting SRVs. This situation can be driven to a core melt condition by loss of high pressure injection capability coupled with the inability to depressurize. Loss of high pressure injection can occur due to exhaustion of the CST inventory coupled with inability to draw suction from the pressure suppression pool, overheating of the turbine driven HPCI and RCIC pump lube oil, or loss of all injection systems due to blow-down phenomena associated with containment failure. Table 10.1 is a summary listing of this event sequence.

In the second case, the HPCI (HPCS) and RCIC systems cannot maintain the water level above the top of the core (or the suppression pool temperatures are excessively high), and the operator depressurizes to allow use of the low pressure injection systems. The combined flow of the LPCI and LPCS systems is such that the entire reactor vessel can be filled with cold water within 3 to 8 min (if vessel pressure is sufficiently low). This is an extremely significant point since reactor power in this sequence is limited only by the core voiding induced by the maintenance of reduced vessel water levels. Several scenarios are possible once the vessel has been depressurized: (a) injection is controlled by the operator to maintain the level near the top of the core, low pressure injection is lost, core melts; (b) injection is initially controlled by operator to maintain reduced level, RHR pressure suppression pool cooling is lost, containment fails, vessel injection fails, core melts; (c) low pressure injection flow limited by vessel pressure, core is partially uncovered; (d) low pressure injection systems cover core with cold water, producing power spike, possibly damaging fuel and bursting the reactor vessel; or (e) the core shatters into subcritical geometry due to thermal shock. Table 10.2 is a summary listing of the event chronology for these sequences.

Tables 10.1 and 10.2 reveal the large number of system and hardware failures which must occur if the ATWS event is to lead to core melting. This evaluation is, however, based on the assumption that the operator is well trained and familiar with the Emergency Procedure Guidelines[10.2] — and follows them. As in all accidents, incorrect or unplanned operator actions could change the accident sequence and severity.

This description has been presented in an effort to provide a basis for identifying ATWS severe accident analysis code modeling requirements which are discussed in the following sections.

10.4 ATWS Modeling Concerns

10.4.1 Kinetics/thermohydraulics

Loss of injection results in the gradual uncovering of the critical core. As the void fraction in a given zone of the core increases, moderation drops, and at some point the nuclear fission reaction in the zone ceases. It is possible that more coolant is required to maintain criticality than is required to provide adequate cooling. *If this is true, core melting for ATWS sequences would occur under decay heat — not at power.* The exception to this is Case II D (Table 10.2), in which LPECCS injection results in an intense core power spike, followed by vessel breach due to excessive pressures.

During the core uncovery phase of ATWS accidents, steam generated in the covered sections of the core is available for cooling and reaction in the uncovered zones. This could lead to situations where some zones are critical but well cooled, other zones are subcritical and well cooled, and still other zones are subcritical and under-cooled. Detailed evaluation of this scenario is extremely difficult since the time- and space-dependent power levels of the critical zones are closely coupled to core thermohydraulic conditions.

Detailed treatment of the core uncovery phase of this accident requires a coupled multi-dimensional neutron transport and thermohydraulics code such as the RAMONA-3 code,[10.4] which employs a coupled neutronics/thermohydraulic modeling approach. Such detailed modeling approaches are inappropriate for inclusion in integrated severe accident analysis codes because (a) the core dryout and uncovery phase of many ATWS accidents will last for only a brief period of time, and (b) detailed neutronics/thermohydraulics codes typically require prohibitively large amounts of computer memory and CPU time to exercise.

A second method for the study of ATWS accidents would be to run a detailed code, such as RAMONA, independently and feed the time dependent core power distribution into a severe accident analysis code. This approach is impractical, however, since related thermohydraulic parameters such as nodal void fractions, system pressures, fuel temperatures, etc., must be input to the severe accident code as well if realistic and consistent core heatup transients are to result.

Historically, the approach employed for MARCH analysis of ATWS is to assume that "covered" regions (regions below the calculated two-phase level) of the core operate at some fixed percentage of full power, while uncovered core regions produce only decay heat. This is an undesirable approach since it results in artificially discontinuous core power states and the calculated two-phase level may have little physical significance.

From the computer utilization standpoint, the only practical core power model for severe accident analysis code ATWS applications is a point kinetics model. Although these models have zero dimensionality, they can explicitly account for void and doppler feedback mechanisms. These models have not previously been utilized for severe accident analysis applications, but it is probable[10.5] that nodal point kinetics models could be formulated to approximate spatial power distributions during ATWS accidents. Significant care must be exercised to ensure that such models yield results that are consistent with detailed codes.

Case II D (Table 10.2) thermohydraulic modeling requirements are probably beyond the present state of the art for severe accident analysis codes. In this case, reactor vessel depressurization coupled with LPECCS injection results in rapid (less than 2 min) filling of the reactor core with cold water. This produces an immediate positive reactivity addition, resulting in an intense core power increase and concomitant vessel pressure pulse which may fail the vessel. Fuel pins may also melt due to the power burst. Reactor subcriticality would be achieved within a few seconds due to disassembly of the core and vessel.

It should be emphasized that the operator can manually control the LPECCS injection rates to prohibit excessively rapid core submersion. Indeed, such operator action may be necessary to prevent the occurrence of the Case II D ATWS over-cooling accident. Additional analysis is needed to determine the range of core flooding rates, system pressures, etc., under which reactor vessel rupture might occur. In any event, the overall time span for the disassembly phase of this accident is so brief that detailed analysis of the vessel failure phenomena by integrated severe accident analysis codes is impractical and unwarranted.

10.4.2 In-vessel structural modeling considerations

The major structural difference between an ATWS and other severe accidents is that the control rods are partially or fully withdrawn from the core during the course of the accident. This translates to the need for a structural core model in which the control rods can be removed, allowing adjacent fuel assembly canisters to thermally communicate. Removal of the stainless steel sheathed control rods from the core might result in a significant reduction in hydrogen generation during the in-core meltdown phase of the accident due to a reduction in the amount of steel and B_4C available for reaction.

Accurate evaluation of the ATWS accident also necessitates a lower plenum structural (control rod drive guide tube) model which accommodates the presence of the control blades inside the guide tubes.

10.4.3 Containment modeling issues

10.4.3.1 Pressure suppression pool. The ATWS event is characterized by sustained periods of SRV actuation. Since the rated capacity of a single relief valve is ~6% (plant dependent) of the full power reactor steaming rate, several SRVs will be open simultaneously if the MSIVs are closed and the reactor power is above this level. The pressure suppression pool will therefore be subjected to distributed heat loads and high mass flux rates in the vicinity of the SRV exhaust quenchers. Plants with turbine driven HPCI and RCIC systems will also experience pressure suppression pool heating due to ECCS turbine exhaust during periods in which these systems are operating.

Because of the distributed nature of the SRV quencher locations, the pressure suppression pool can probably be treated as a well mixed volume for analysis purposes under ATWS conditions when several SRVs are open. Nevertheless, due to the relatively high SRV discharge mass fluxes involved, it is possible that phenomena in the vicinity of the SRV quenchers would produce accelerated suppression chamber (containment) pressurization rates.

Techniques for treating localized pressure suppression pool phenomena are currently under development, but realistic modeling approaches are beyond the current state of the art.[10.6] A simplified, well mixed, few node pool model such as that needed for modeling of some non-ATWS events is therefore appropriate for current ATWS modeling applications. The capability for the user to specify the local pool temperature or time at which complete condensation of SRV and turbine discharge steam ceases is, however, a highly desirable modification for ATWS modeling applications.

10.4.3.2 Drywell. ATWS Case II D (reactor vessel rupture) presents an imposing containment modeling challenge due to the great uncertainties associated with reactor vessel failure pressure, mode, location, size, etc. Primary containment (drywell) failure could occur either by excessive pressure or by penetration of missiles generated by reactor vessel failure. The large uncertainties associated with these phenomena preclude inclusion of detailed models in severe accident analysis codes. Reactor vessel blowdown into the drywell can be accommodated with drywell mass and energy input tables and drywell failure due to missile penetration can be appropriately simulated by allowing the user to input containment failure times and hole size data. The remaining aspects of primary and secondary containment response during ATWS accidents can be adequately simulated by a code having the capabilities previously described in this report.

Chapter 10 References

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10.4. *RAMONA Analysis of the Peach Bottom-2 Turbine Trip Transients*, EPRI NP-1869, Electric Power Research Institute (June 1981).

10.5. Ibid, Sect. 7.

10.6. D. H. Cook, *Pressure Suppression Pool Thermal Mixing*, ORNL/TM-8906, Oak Ridge National Laboratory (to be published).

Table 10.1. Case I ATWS degraded core scenario

1. Auto scram failure
2. Manual scram failure
3. Manual rod insertion failure
4. SLC failure
5. CST inventory depletion
6. HPCI/RCIC recirculation mode unavailable
7. ADS/SRVs unavailable for depressurization
8. Core melt

Table 10.2. Case II ATWS degraded core scenarios

1. Auto scram failure
2. Manual scram failure
3. Manual rod insertion failure
4. SLC failure
5. HPECCS insufficient
6. Depressurize

II A	II B	II C	II D	II E
7a. Reduced level maintained with LPECCS	7b. Reduced level maintained with LPECCS	7c. LPECCS injection pressure limited (core maintained partially covered)	7d. LPECCS fills vessel	7e. LPECCS begins
8a. LPECCS fails	8b. RHR pool cooling fails	8c. Partial core melt?	8d. Core power pulse	8e. Fuel shatters
9a. Core melts	9b. Containment fails			
			9d. Core melts/ vessel fails	9e. Core melts
	10b. Injection fails			
	11b. Core melts			

11. SUMMARY

11.1 Introduction

The purpose of this report is to identify the BWR plant hardware, systems, and phenomena which must ultimately be modeled to permit realistic simulation of severe accidents in boiling water reactors. Such an undertaking is a formidable task since our current understanding of many severe accident phenomena is quite limited and the definition of "realistic simulation" is, therefore, debatable. Due to the time restrictions associated with completion of the assessment, this report constitutes a basic description of what should be modeled rather than a guide to how phenomena should be modeled. The latter is a necessary and logical extension of the present assessment.

The approach employed in this assessment is based on consideration of five major factors; (1) actual plant hardware and operating procedures, (2) best estimate visualization of severe accident phenomena, (3) previous BWR severe accident analysis experience, (4) present modeling capabilities, and (5) current and future code applications. Only after appropriate consideration of each of these five factors can one specify modeling priorities. This report has attempted to address each of these areas. Based on this assessment, the following section presents the author's views regarding current BWR severe accident model development priorities.

11.2 BWR Model Development Priorities

Tables 11.1 and 11.2 present the author's views regarding current BWR severe accident model development priorities. Many additional component and system models must be developed or improved, but only after the items in Tables 11.1 and 11.2 are addressed.

One of the most fundamental components of the in-vessel severe accident model is the thermohydraulic (TH) model. The TH model provides necessary boundary conditions for heat and fission product transfer calculations and therefore forms the basic foundation for all in-vessel severe accident phenomenological modeling. The existing MARCH TH model is inappropriate for BWR applications and unnecessarily limits the degree to which the code can be improved for BWR applications.

Future BWR severe accident analysis codes should be formulated around an in-vessel TH model which explicitly accounts for multiple subchannel flows, bypass zones, upper and lower plenum structural heat transfer, natural circulation and fission product heating. With the possible exception of fission product heating, the current body of knowledge regarding these phenomena is such that significant improvements in existing BWR severe accident TH modeling is achievable within the existing state of the art.

The development of credible BWR structural heating, deformation and relocation models ranks second in priority only to development of the TH model. In many cases no appropriate structural models exist (control rod guide tube and vessel head melt penetration), while in other cases (core structures) existing models are in such embryonic states that significant improvements are necessary and achievable prior to pursuit of other model development goals.

The three major containment modeling priorities are listed in Table 11.2. As in the case for reactor modeling, the top severe accident containment modeling priority is the development of a generalized thermodynamic modeling approach which accommodates both series and parallel flow paths in both the primary and secondary containment. The modeling approach must also be capable of accommodating MK II primary containment phenomena.* Such a model is necessary to provide a framework for later implementation of containment ESF models.

The second containment modeling priority is the development and integration of a practical core/concrete interaction model into the thermodynamic model discussed above. This is particularly necessary for analysis of the small MK I and MK II containments. The development of improved pressure suppression pool models is of equal importance to the core/concrete interaction model development since pressure suppression pool and core/concrete interaction phenomena are the major sources of mass and energy inputs to the containment.

Following development of improved modeling approaches for the three areas listed in Table 11.2, models for the remaining items in Tables 8.1 through 8.7 should be incorporated. Great care should be taken to ensure that newly developed TH models can accommodate the systems and structures shown in Figs. 8.1 through 8.5.

In some cases model developers may be able to draw upon experience and approaches developed previously. However, in many situations, existing models are either too unwieldy for application in integrated severe accident analysis codes (i.e., TRAC-BD1, CONTAIN, RAMONA), or too simplistic (MARCH). It should be emphasized that no models exist at all for some BWR severe accident phenomena.

Future BWR severe accident analysis code developers should strive to formulate models which embody a reasonable balance between mechanistic modeling, computer resource demands, and user practicality.

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*The core/concrete mixture would dump directly into the suppression pool following drywell floor failure.

Chapter 11 References

11.1 R. M. Harrington and L. J. Ott, *The Effect of Small Capacity, High-Pressure Injection Systems on TQUV Sequences at Browns Ferry Unit One*, NUREG/CR-3179, ORNL/TM-8635, Oak Ridge National Laboratory (September 1983).

Table 11.1. BWR in-vessel model development priorities

Model	Priority[a]
1. In-vessel thermohydraulics	1
incorporating: (a) multiple channels (b) bypass zone (c) upper and lower plenum structural heat transfer (d) natural circulation (e) fission product heating of structures, gases (f) multiple injection points	
2. Structural heatup/deformation/relocation	2
for: (a) fuel (b) clad (c) canisters (d) control blades (e) upper vessel internals (f) control rod guide tubes and drive mechanisms (g) lower vessel head	

[a]1 = highest priority.

Table 11.2. BWR containment model development priorities

Model	Priority[a]
1. Generalized primary and secondary containment model	1
incorporating: (a) series and parallel flow paths (b) MK II capability	
2. Core/concrete interaction	2
3. Pressure suppression pool	2

[a]1 = highest priority.

Appendix A

ACRONYMS AND SYMBOLS

ADS	Automatic Depressurization System
AHU	Air Handling Unit
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
BOP	Balance of Plant
BWR	Boiling Water Reactor
CAD	Containment Atmospheric Dilution
CPU	Central Processor Unit
CRD	Central Rod Drive
CRDS	Central Rod Drive System
CRDHS	Control Rod Drive Hydraulic System
CST	Condensate Storage Tank
ECCS	Emergency Core Cooling System
ESF	Engineered Safety Feature
FCU	Fan Coil Unit
FPCC	Fuel Pool Cooling and Cleanup
HCU	Hydraulic Control Unit
HEPA	High Efficiency Particulate Air
HPCI	High Pressure Coolant Injection
HPCS	High Pressure Core Spray
HVAC	Heating Ventilating and Air Conditioning
IRM	Intermediate Range Monitor
LOCA	Loss of Coolant Accident
LPCI	Low Pressure Coolant Injection
LPCS	Low Pressure Core Spray
LPRM	Local Power Range Monitor
MSIV	Main Steam Isolation Valve
MSIV-LCS	Main Steam Isolation Valve Leakage Control System
NRC	Nuclear Regulatory Commission
NPSH	Net Positive Suction Head
NSSSS	Nuclear Steam Supply Shutoff System

ORNL	Oak Ridge National Laboratory
PCIS	Primary Containment Isolation System
PWR	Pressurized Water Reactor
RCIC	Reactor Core Isolation Cooling
RCIS	Rod Control and Information System
RCPB	Reactor Coolant Pressure Boundary
RERS	Reactor Enclosure Recirculation System
RES	Nuclear Regulatory Commission Office of Nuclear Regulatory Research
RHR	Residual Heat Removal
RPS	Reactor Protection System
RWCU	Reactor Water Cleanup Unit
SASA	Severe Accident Sequence Analysis
SDIV	Scram Discharge Instrument Volume
SGTS	Standby Gas Treatment System
SJAE	Steam Jet Air Ejector
SLC	Standby Liquid Control
SPMS	Suppression Pool Makeup System
SRM	Source Range Monitor
SRV	Safety/Relief Valve

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16. ABSTRACT (200 words or less)

This report documents the results of an assessment performed at Oak Ridge National Laboratory to determine the reactor and containment hardware, systems, and phenomena which must be modeled in realistic boiling water reactor severe accident analysis computer codes. The scope of the assessment is limited to BWR-4, 5, and 6 reactors and Mark I, II, and III containment systems. The report presents a concise review of the subject reactor and containment designs, together with a description of the reactor and containment systems which have the capacity to impact the outcome of severe accidents. The results of recent BWR probabilistic risk assessments are briefly discussed, and a detailed visualization of a BWR core melt accident is presented. Recommendations are made regarding the type of phenomena which should be modeled and the level of modeling sophistication required for various stages of the core melt accident. Finally, the current availability of the necessary models is discussed along with the associated model development priorities.

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