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CATEGORIZATION OF PWR ACCIDENT SEQUENCES
AND GUIDELINES FOR FAULT TREES:
SEISMIC INITIATORS

C. Y. Kimura

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MASTER

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EXECUTIVE SUMMARY

This study developed a set of dominant accident sequences that could be applied generically to domestic commercial pressurized water reactors (PWRs) as a standardized basis for a probabilistic seismic risk assessment. This was accomplished by ranking the Zion 1 accident sequences developed in the Seismic Safety Margins Research Program (SSMRP) [1] and by reviewing different PWR safety system interactions. The Zion 1 accident sequences were ranked in order of radioactive release risk as predicted by the SEISM2 computer code.

A set of 18 dominant accident sequences with feed and bleed capability accounting for 92% of the total radioactive release risk was developed. Also developed was another set of 17 accident sequences, assuming no feed and bleed capability, which accounted for 95% of the total radioactive release risk.

The pertinent PWR safety systems were compared on a plant-by-plant basis to determine the applicability of the dominant accident sequences of Zion 1 to other PWR plants. If the PWR safety systems had comparable components, operated similarly, and initiated their operation in the accident sequences in the same approximate time frame as Zion 1, then the applicability of the Zion 1 dominant accident sequences to other PWR plants was demonstrated.

The functional event trees were developed to describe the system functions that must work or not work in order for a certain accident sequence to happen, one for pipe breaks and one for transients. Both of these functional event trees present the basic sequence of functional events that occur following the initiating event. Plant specific systemic event trees should be developed from the functional event trees following the guidelines in Appendix I of the WASH-1400 report [19]. The systemic event tree describes the specific plant safety systems that perform the system functions defined by the functional event tree.

To apply the dominant accident sequences developed, plant-specific systemic event trees should be developed from the functional event tree. A "reactor safety system by reactor safety system" comparison should be made of the plant under analysis with the Zion Unit 1 plant. Significant differences (in the judgement of the analyst) particularly in the timing and order of the reactor safety systems initiation should

be incorporated within the plant specific event trees. If the plant specific systemic event trees do not differ from the SSMRP event trees, then the dominant accident sequences may be applied directly without modification. However, the radioactive release risk of the dominant accident sequences will not apply to the plant under analysis. To calculate the radioactive release risk, fault trees of the plant-specific reactor safety systems should be developed. If the reactor safety systems of the plant under analysis are similar to the Zion Unit 1 reactor safety systems, then a plant-specific fault tree of the particular safety system may not have to be developed. The analyst should be aware, however, that the Zion plant is unique in that many of its reactor safety systems have been designed with additional components and as such, has more system flexibility than most PWR plants. All but the earliest Westinghouse PWR plants were generally designed and built to the same safety philosophy and design standards as the Zion plants but may lack the redundancy and system flexibility of the Zion plants.

The dominant accident sequences should be particularly applicable without modification to the 4-loop 1100-MWe class Westinghouse PWR plants using series 51 SGs and with dry containments. These same dominant accident sequences should also be applicable without modification to the 2-, 3- and 4-loop Westinghouse PWRs that use other SG series and with dry containments. These dominant accident sequences may be applicable to Westinghouse PWR plants with ice containments if the plant specific systemic event tree has been modified to reflect the difference in containment design and construction. The application of these dominant accident sequences to Combustion Engineering PWR plants should be generally possible without modification. The applicability of these dominant accident sequences to Babcock and Wilcox PWR plants may not be possible without considerable modification to the plant-specific systemic event tree because of the much faster response of the B&W plant to plant transients and the differences in reactor safety systems.

PWR GENERIC ACCIDENT SEQUENCES

1) INTRODUCTION

The purpose of this study was to develop a set of dominant accident sequences that could be applied generically to domestic commercial pressurized water reactors (PWRs) as a standardized basis for a probabilistic seismic risk assessment. This was accomplished by ranking the Zion 1 accident sequences developed in the Seismic Safety Margins Research Program (SSMRP) [1] and by reviewing different PWR safety system interactions. The Zion 1 accident sequences were ranked in order of radioactive release risk as predicted by the SEISM2 computer code. The pertinent PWR safety systems were compared on a plant-by-plant basis to determine the applicability of the dominant accident sequences of Zion 1 to other PWR plants. If the PWR safety systems had comparable components, operated similarly, and initiated their operation in the accident sequences in the same approximate time frame as Zion 1, then the applicability of the Zion 1 dominant accident sequences to other PWR plants was demonstrated.

The generic accident sequences were developed and their applicability demonstrated. This report presents a discussion of the following topics: Section 2- a comparison of PWR systems; Section 3- development of the functional event tree; Section 4- development of dominant accident sequences; Section 5- application of dominant accident sequences; and Section 6- conclusions.

2) PWR SYSTEMS COMPARISON

This section describes the comparison of the general plant features, the important reactor safety systems and their interactions for pressurized water reactors (PWRs) designed and built by the three domestic PWR vendors, Westinghouse (W), Combustion Engineering (CE), and Babcock and Wilcox (B&W).

In a typical PWR plant, heat generated in the reactor core is carried away by water (reactor coolant) circulating at sufficient pressure (typically 2250 psia) to prevent the water from flashing to steam, through the primary circuit. The water in the primary loop serves to cool and moderate the reactor. The heat is transferred from the primary to a secondary system containing water at a lower pressure

(typically 1000 psia) to allow steam generation, in a heat exchanger called the steam generator. The major components of a PWR primary loop are: 1) the reactor vessel which contains the uranium oxide fuel, core internals, control rod assemblies, and in-core instrumentation, 2) the pressurizer, 3) the electrically driven reactor coolant pumps, and 4) the steam generators.

Westinghouse PWRs built or being built in the United States are characterized by the number of primary coolant loops connected to the reactor vessel. A Westinghouse PWR may have two, three, or four loops in the primary coolant system. Each loop contains one primary coolant pump and one vertical U-tube and shell steam generator (SG). The components of the primary coolant loop are sized to the thermal power rating of the reactor core. Table 1 lists the domestic Westinghouse PWRs in order of plant size, Westinghouse SG model, and chronological commercial operation.

Two-loop Westinghouse plants typically have 121 fuel assemblies in their reactor cores and use series 93 reactor coolant pumps. Three-loop Westinghouse plants typically have 157 fuel assemblies in their reactor cores and use series 93 or 93A reactor coolant pumps. Four-loop Westinghouse plants typically have 193 fuel assemblies in their reactor cores and use series 93 or 93A reactor coolant pumps. The height difference between the top of the active reactor core and the bottom of the SG tube support plate is approximately 18 feet for the two-loop, three-loop, and the four-loop Westinghouse plants [4].

All Westinghouse PWR plants use zircalloy clad fuel except for San Onofre 1, and Yankee Rowe which uses stainless steel clad fuel [4]. The use of stainless steel clad fuel and its effect on a plant during the course of an accident was not evaluated.

Six Westinghouse PWR plants have ice condenser containment. They are Donald C. Cook 1 and 2, Sequoyah 1 and 2, and Watts Bar 1 and 2. All other Westinghouse PWR plants use the large dry containment concept [8]. The use of the ice condenser containment instead of the usual dry containment concept, allows a smaller containment to be used and less demanding environmental conditions within the containment following a loss-of-coolant-accident (LOCA).

Combustion Engineering PWRs built or being built in the United States are characterized by two loops in the primary coolant system except for the Maine Yankee plant which is the only three-loop CE plant. Each loop contains two primary coolant pumps and one vertical U-tube and shell steam generator (SG) except for Maine Yankee which has one pump and one SG per loop. The components of the primary coolant loop are sized to the thermal power rating of the reactor core. Table 2 lists the domestic CE PWRs in order of plant size and chronological commercial operation.

A typical CE PWR plant uses Byron Jackson primary coolant pumps which are of the same type class (although it may be rated differently) and are generically identical in design. There is typically a 14-ft difference in height between the top of the active reactor core and the bottom tube sheets of CE steam generators [5]. All CE plants use dry containments.

Babcock and Wilcox PWRs built or being built in the United States are characterized by two loops in the primary coolant system. Each loop contains two primary coolant pumps and one vertical once-through tube and shell steam generator (OTSG). The unique feature of B&W PWRs is the use of OTSGs which have a smaller secondary water inventory during normal power operation than Westinghouse or CE SGs. Table 3 lists the domestic B&W PWRs in order of plant size and chronological commercial operation.

Table 4 compares the characteristics of the reactor coolant system (RCS) of as many PWR plants as could be found. All PWR plants have only one pressurizer but different connection points for the surge and spray lines to the RCS. The volume of the pressurizer relative to the total RCS volume varies from 10% to 16%. The volume of the pressurizer relative to the total volume of the RCS is an important indication of the RCS response to pressure transients. As the pressurizer is the only point within the RCS where steam and water exist separately, pressure transients within the RCS increase or decrease the steam volume within the pressurizer. The larger the pressurizer volume relative to the total RCS volume, the more volume available for the steam bubble, and the slower the pressure transient within the RCS because of the greater compressibility of steam to water. The RCS components, i.e. main coolant pumps, steam generators, and pressurizer, of the various PWR plants perform their functions similarly despite differences in design, size, number of components and arrangement within the RCS.

Should a pressure increase occur in the RCS that is beyond the control capabilities of the pressurizer, Pilot Operated Relief Valves (PORVs) and Safety Valves (SVs) actuate to relieve the primary pressure. Table 5 compares the pressure relief system characteristics of as many PWR plants as could be found. The PORVs are spring loaded, air actuated valves designed to prevent the lifting of the pressurizer safety valves and to allow the reactor to respond to small transients without challenging the plant safety limits. Surprisingly, PORVs are not considered as part of the PWR plant's safety systems and thus, are not safety grade items. However, the failure of a PORV led to the March 28, 1979 accident at the Three Mile Island Unit 2 plant. PWR plants attempt to avoid lifting of the pressurizer safety valves since these valves represent the safety pressure limits of the RCS, and thus are seldom used valves which would tend not to reseal properly when actuated. All PWR plants have motor operated block valves located upstream of the PORV, so that a PORV can be isolated if the PORV seat leakage becomes significant or if the PORV fails to reseal. Since the Three Mile Island 2 accident, many PWR plants have isolated their PORVs with their block valves and use their safety valves alone to control severe RCS pressure transients. This has resulted in fewer PWR plant shutdowns due to RCS pressure transients but more approaches to the RCS pressure safety limits. The PORV setpoints listed in Table 5, particularly for the B&W plants, may have been changed as a direct result of the Three Mile Island 2 accident, so this information should be treated with caution.

The Emergency Core Cooling System (ECCS) consists of the High Pressure Injection System (HPIS), the Accumulators (also called the Safety Injection Tanks for CE plants and Core Flood Tanks for B&W plants), and the Low Pressure Injection System (LPIS) (also called the Residual Heat Removal System or Decay Heat Removal System). Some plants, usually Westinghouse, include the charging pumps of the Chemical and Volume Control System (CVCS) as part of the ECCS. Using the CVCS charging pumps to function as part of the ECCS allows water injection into the RCS above the HPIS initiation setpoint. However, the capacity of the charging pumps (generally reciprocating but sometime centrifugal pumps) is usually too small to control the effects of all but the smallest pipe breaks.

Table 6 compares the characteristics of the HPIS of as many PWR plants as could be found. The HPIS injects water into the RCS when the primary pressure drops below approximately 1500 psig. There are differences in piping arrangements and controls,

but a HPIS for a PWR usually consists of several high pressure centrifugal pumps driven by electric motors, drawing its water supply from the refueling water storage tank(s), and activated automatically after a short time delay.

Table 7 compares the characteristics of the LPIS of as many PWR plants as could be found. The LPIS is usually considered to be only the low pressure injection pumps, its piping arrangement, and controls, but since the accumulators are usually initiated at the same approximate pressure as the low pressure injection system, they will be considered together. The accumulators or core flood tanks (as they are called for CE PWRs) or safety injection tanks (as they are called for B&W PWRs) are passive water tanks pressurized by nitrogen gas, that automatically injects water into the RCS when the primary pressure drops below approximately 600 psig. Usually there are two check valves in series, that swing open automatically to allow the accumulator water supply into the RCS. Injection points differ, but the basic purpose of the various accumulator, core flood, and safety injection tank system are the same. The LPIS injects water into the RCS when the primary pressure drops below HPIS initiation pressure and sometimes below the accumulator initiation pressure. Rancho Seco and Arkansas One 1 (both B&W plants) activate their LPIS pumps at the same pressure as their HPIS pumps in order for the LPIS pumps to supply sufficient net positive suction head (NPSH) to the HPIS pumps to operate.

The steam generators (SGs) for PWR plants are the primary heat sink for the RCS. SGs come in two types, vertical shell and U-tube (VS&UT) for Westinghouse and CE plants, and once-through steam generators (OTSGs) for B&W plants. As the SGs are the primary heat sink for the RCS, loss of feedwater supply to the SGs soon results in loss of all cooling capability to the RCS. Tables 8 and 9 present some of the characteristics of the SGs of as many PWR plants as could be found. Because of the larger secondary water inventory of Westinghouse and CE SGs, the dryout time for these SGs with loss of all feedwater supply at hot full power (no reactor scram) is two to four times that of B&W plants with OTSGs. The larger secondary inventory of Westinghouse and CE SGs results in slower RCS pressure and temperature transients due to loss of feedwater supply. The OTSGs of B&W plants are also located such that the bottom of the SG tubes are below the top of the active reactor core, except for the Davis Besse Unit 1 plant and B&W plants designed after it, which use the "raised loop" configuration. The effect of placing the bottom of the OTSG below the active core is to hinder the RCS natural circulation characteristics of B&W plants. The

raised loop design was introduced to improve the thermal-hydraulic performance characteristics of B&W plants subsequent to loss-of-coolant-accidents (LOCA) and to permit exclusion of the vent valves. The vent valves have been retained in this design but reduced in number from 8 to 4 [6]. Another unique feature of B&W PWR plants is the use of an integrated control system (ICS) which is designed to properly coordinate the operation of the reactor, the OSTGs, feedwater control, and the main turbine under normal operating conditions. The ICS is used in B&W PWR plants because the response of the plant to transient conditions (i.e. change in grid demand, change in reactor power, etc.) is much more rapid than in Westinghouse or CE PWR plants.

Table 10 compares the characteristics of the main feedwater (MFW) system of as many PWR plants as could be found. The MFW is the primary source of secondary water supply to the SGs but is not considered a safety grade system. Details such as the inclusion or exclusion of the condensate booster pumps, inclusion or exclusion of full-flow demineralizers, number of feed-water heater stages, etc., differ between the various plants, but the functional performance purpose of the MFW systems of all PWRs is similar.

The auxiliary feedwater system (AFWS) (or emergency feedwater system as they are called in B&W plants) functions as a backup system for the removal of heat from the RCS when the MFW system is not available. It plays an important role in mitigating the effects of small-small LOCAs (pipe breaks less than 1 1/2 inches but greater than 1/2 inch in diameter) and plant transients. The AFWS system is also designed to hold the plant at hot standby, or to cool down the primary system to temperature and pressure levels at which the low pressure decay heat removal system can operate. The AFWS can also be used during normal plant startup and shutdown condition. Tables 11, 12, and 13 present the characteristics of AFWS systems for as many Westinghouse, CE and B&W PWR plants, respectively, as could be found. This information was presented in three separate tables because the detail differences between the various AFWS systems were too great to present in one table. As Tables 11, 12, and 13 show, there is wide disparity between AFWS systems of different manufacturers and even among plants of the same PWR manufacturer. This may be because the AFWS systems are designed by the individual architectural engineering firms that design the conventional portion of the plant, rather than by the nuclear steam supply system (NSSS) vendor. Generally, an AFWS system consists of a

combination of steam-turbine-driven and electric-motor-driven pumps sized so that the system can provide sufficient water to the steam generators for decay heat removal following loss of main feedwater with any one AFW pump out of service.

The containment spray injection system (CSIS) sprays water containing sodium hydroxide (NaOH) into the containment building atmosphere. The CSIS acts to control the containment atmospheric pressure and temperature by condensing steam released into the containment during a reactor vessel rupture or a LOCA. The NaOH scavenges the volatile fission products that are released into the containment during a reactor vessel rupture or LOCA and thus serves to mitigate the radioactivity release. Table 14 compares the characteristics of CSISs of as many PWR plants as could be found. Note that the CSIS of Zion 1 and 2 is unlike the CSIS of any other PWR plant compared. Zion's CSIS has more diverse power sources, more fluid trains, and more system flexibility than other plants.

The containment fan cooler system (CFCS) draws air from the containment atmosphere, cools it, and condenses steam or water vapor contained by the air. The CFCS is an additional system that is used to control the containment atmospheric pressure and temperature and to filter the volatile and gaseous fission products from the containment atmosphere. Table 15 presents the characteristics of CFCS of as many PWR plants as could be found. Generally, the safety design philosophy of a PWR plant is that the CSIS or the CFCS by itself are of sufficient capacity to be able to control the containment atmospheric pressure and temperature during a reactor vessel rupture or a LOCA to below the design limits of the containment. Additionally, half of the CSIS capacity and half of the CFCS capacity acting together are usually sized to control the containment atmospheric pressure and temperature to the same design limits. Note that the Zion plants provides additional redundancy to both the CSIS and the CFCS by providing spare pumps, fans, and fluid trains so that some part of these systems may be down for repair or maintenance and still maintain the required accident mitigation capability. Note also that the CE system 80 PWR plants lack a CFCS entirely because their containment and CSIS have been designed with additional capacity.

Table 16 compares some characteristics of containments of as many PWRs as could be found. The dry containments of most PWRs are steel-lined prestressed reinforced concrete structures. Structural details of the various dry containments differ but

the basic functional performance of these containments is the same. The PWR plants that utilize the ice pressure suppression system, Watts Bar being typical, are designed to much different design standards.

Emergency electrical power is a critical item in the mitigation of a reactor vessel rupture, LOCA, or plant transient consequences. Although emergency electrical power is usually not considered as a component event in the systemic event trees (see Section 3), it is considered within many system fault trees and very often turns out to be contained in the minimal cut sets that lead to system failure. Table 17 compares the characteristics of emergency electrical power sources of as many PWR plants as could be found. Note that the Zion plants are the only plants of the PWR plants compared that share a common diesel (sometimes called the "swing" diesel). This provides additional flexibility to emergency electrical power sources.

Table 1

Westinghouse (W) PWR Plants [2-4]

Plant Name	Commercial Operation (Expected)	Licensed MWth	Maximum Dep. Cap. Net MWe	No. of Primary Loops	S/G Ther- mal Cap. MWth	W S/G Series
Yankee Rowe	7/61	600	175	4	150	27
San Onofre 1	1/68	1347	436	3	449	27
Haddam Neck	1/68	1825	569	4	456.25	27
Robert E. Ginna	7/70	1520	470	2	760	44
Point Beach 1	12/70	1518	495	2	759	44
Point Beach 2	10/72	1518	495	2	759	44
H. B. Robinson 2	3/71	2300	665	3	766.67	44
Turkey Point 3	12/72	2200	666	3	733.33	44
Turkey Point 4	9/73	2200	666	3	733.33	44
Indian Point 2	8/74	2758	849	4	689.50	44
Indian Point 3	8/76	3025	965	4	756.25	44
Prairie Island 1	12/73	1650	503	2	825	51
Prairie Island 2	12/74	1650	500	2	825	51
Kewaunee	6/74	1650	511	2	825	51
Surry 1	12/72	2441	775	3	813.67	51
Surry 2	5/73	2441	775	3	813.67	51
Beaver Valley 1	10/76	2660	810	3	886.67	51
Beaver Valley 2	(5/86)	2660	852*	3	886.67	**
Joseph M. Farley 1	12/77	2652	804	3	884	51
Joseph M. Farley 2	7/81	2652	814	3	884	51
North Anna 1	6/78	2775	877	3	925	51
North Anna 2	12/80	2775	890	3	925	51
Virgil C. Summer 1	(1/84)	2775	900*	3	925	**
Shearon Harris 1	(3/86)	2775	915*	3	925	**
Zion 1	12/73	3250	1040	4	812.50	51
Zion 2	9/74	3250	1040	4	812.50	51
Donald C. Cook 1	8/75	3250	1030	4	812.50	51
Donald C. Cook 2	7/78	3411	1082	4	852.75	51
Trojan	5/76	3411	1080	4	852.75	51
Salem 1	6/77	3338	1079	4	834.50	51
Salem 2	10/81	3411	1106	4	852.75	51
Sequoyah 1	7/81	3411	1148	4	852.75	**
Sequoyah 2	6/82	3411	1128	4	852.75	**
McGuire 1	12/81	3411	1180	4	852.75	**
McGuire 2	(3/84)	3411	1180*	4	852.75	**
Byron 1	(6/84)	3411	1120*	4	852.75	**
Byron 2	(11/85)	3411	1120*	4	852.75	**
Diablo Canyon 1	(6/84)	3338	1084*	4	834.50	**

Table 1 continued

Westinghouse (W) PWR Plants [1-3]

Plant Name	Commercial Operation (Expected)	Licensed MWth	Maximum Dep. Cap. Net MWe	No. of Primary Loops	S/G Ther- mal Cap. MWth	W S/G Series
Diablo Canyon 2	(2/85)	3411	1106*	4	852.75	**
Callaway	(4/84)	3425	1188*	4	856.25	**
Watts Bar 1	(11/84)	3411	1165*	4	852.75	**
Watts Bar 2	(10/86)	3411	1165*	4	852.75	**
Comanche Peak 1	(/84)	3411	1150*	4	852.75	**
Comanche Peak 2	(/86)	3411	1150*	4	852.75	**
Seabrook 1	(12/84)	3411	1198*	4	852.75	**
Seabrook 2	(7/87)	3411	1198*	4	852.75	**
Wolf Creek	(2/85)	3425	1150*	4	856.25	**
Catawba 1	(6/85)	3582	1145*	4	895.50	**
Catawba 2	(6/87)	3582	1145*	4	895.50	**
Braidwood 1	(10/85)	3411	1120*	4	852.75	**
Braidwood 2	(10/86)	3411	1120*	4	852.75	**
Millstone 3	(5/86)	3565	1156*	4	891.25	**
Marble Hill 1	indef.	3415	1130*	4	853.75	**
Marble Hill 2	indef.	3415	1130*	4	853.75	**
Vogtle 1	(3/87)	3565	1100*	4	891.25	**
Vogtle 2	(9/88)	3565	1100*	4	891.25	**
South Texas 1	(6/87)	3800	1250*	4	950	**
South Texas 2	(6/89)	3800	1250*	4	950	**

* - Design Electrical Power (Net MWe).

** - Information unavailable at time of report preparation.

Commercial Operation (Expected) - Date unit was declared by utility owner to be available for the regular production of electricity. Date in parentheses denotes expected date of commercial operation.

Licensed MWth - The maximum thermal power of the reactor authorized by the NRC expressed in megawatts.

Maximum Dep. Cap. Net MWe - Maximum dependable main-unit capacity (gross), winter or summer, whichever is smaller, less the normal station service loads.

S/G Thermal Cap. MWth - Steam generator thermal capacity, expressed in megawatts.

W S/G Series - Westinghouse steam generator series.

Table 2

Combustion Engineering (CE) PWR Plants [2-3, 5]

Plant Name	Commercial Operation (Expected)	Licensed MWth	Maximum Dep. Cap. Net MWe	No. of Primary Loops	S/G Ther- mal Cap. MWth
Fort Calhoun 1	6/74	1500	478	2	750
Maine Yankee	12/72	2630	810	3	876.67
Palisades	12/71	2530	635	2	1265
Calvert Cliffs 1	5/75	2700	825	2	1350
Calvert Cliffs 2	4/77	2700	825	2	1350
Millstone 2	12/75	2700	860	2	1350
St. Lucie 1	12/76	2700	822	2	1350
St. Lucie 2	8/83	2700	805	2	1350
Arkansas One 2	12/78	2815	858	2	1407.50
San Onofre 2	8/83	3410	1087	2	1705
San Onofre 3	(1/84)	3410	1100*	2	1705
Waterford 3	(12/84)	3560	1151*	2	1780
Yellow Creek 1	indef.	3300	1285*	2	1650
Yellow Creek 2	indef.	3300	1285*	2	1650
CE System 80					
Palo Verde 1	(12/84)	3817	1304*	2	1908.5
Palo Verde 2	(9/85)	3817	1304*	2	1908.5
Palo Verde 3	(12/86)	3817	1304*	2	1908.5
WNP 3	indef.	3800	1242*	2	1900

* - Design Electrical Power (Net MWe)

Commercial Operation (Expected) - Date unit was declared by utility owner to be available for the regular production of electricity. Date in parentheses denotes expected date of commercial operation.

Licenses MWth - The maximum thermal power of the reactor authorized by the NRC, expressed in megawatts.

Maximum Dep. Cap. Net MWe - Maximum dependable main-unit capacity (gross), winter or summer, whichever is smaller, less the normal station service loads.

S/G Thermal Cap. MWth - Steam generator thermal capacity, expressed in megawatts.

Table 3

Babcock and Wilcox (B&W) PWR Plants [2-3, 5]

Plant Name	Commercial Operation (Expected)	Licensed MWth	Maximum Dep. Cap. Net MWe	No. of Primary Loops	S/G Ther- mal Cap. MWth
Oconee 1	7/73	2568	860	2	1284
Oconee 2	9/74	2568	860	2	1284
Oconee 3	12/74	2568	860	2	1284
Three Mile Island 1	9/74	2535	776	2	1267.5
Three Mile Island 2	12/78	2772	906*	2	1386
Arkansas One 1	12/74	2568	836	2	1284
Rancho Seco	4/75	2772	873	2	1306
Crystal River 3	3/77	2544	806	2	1272
Davis Besse 1	7/78	2772	874	2	1386
Midland 1	indef.	2452	492*	2	1226
Midland 2	(/86)	2452	818*	2	1226
Bellefonte 1	(4/89)	3760	1235*	2	1880
Bellefonte 2	(4/91)	3760	1235*	2	1880
WNP-1	indef.	3760	1266*	2	1880

* - Design Electrical Power (Net MWe)

Commercial Operation (Expected) - Date unit was declared by utility owner to be available for the regular production of electricity. Date in parentheses denotes expected date of commercial operation.

Licensed MWth - The maximum thermal power of the reactor authorized by the NRC, expressed in megawatts.

Maximum Dep. Cap. Net MWe - Maximum dependable main-unit capacity (gross), winter or summer, whichever is smaller, less the normal station service loads.

S/G Thermal Cap. MWth - Steam generator thermal capacity, expressed in megawatts.

Table 4

Reactor Coolant System (RCS) Characteristics [4-17]

Plant Name	RCS Vol. Cu. Ft.	Pwr. Vol. Cu. Ft.	Pri. Pres. PSIA	Core Tin (F)	Core Tout (F)	Core Flow E6 lbm/hr
	Westinghouse (W)		PWR Plants			
Yankee Rowe	2940	295	2250	*	*	*
San Onofre 1	6450	1000	2250	*	*	*
Haddam Neck	8800	1300	2250	*	*	*
Robert E. Ginna	6245	800	2250	551.9	603.9	64.3
Point Beach 1	6450	1000	2250	552.5	612.5	63.6
Point Beach 2	6450	1000	2250	552.5	612.5	63.6
H. B. Robinson 2	9343	1300	2250	546.2	604.5	97.0
Turkey Point 3	9343	1300	2250	546.2	604.5	97.0
Turkey Point 4	9343	1300	2250	546.2	604.5	97.0
Indian Point 2	12224	1800	2250	543.0	598.5	130.0
Indian Point 3	12032	1800	2250	542.6	602.9	130.1
Prairie Island 1	6450	1000	2250	535.5	601.9	65.2
Prairie Island 2	6450	1000	2250	535.5	601.9	65.2
Kewaunee	6191	1000	2250	535.5	601.9	65.2
Surry 1	9200	1300	2250	*	*	*
Surry 2	9200	1300	2250	*	*	*
Beaver Valley 1	9716	1400	2250	542.5	611.8	96.3
Joseph M. Farley 1	9723	1400	2250	543.4	613.6	96.2
North Anna 1	9957	1400	2250	546.8	614.6	100.5
North Anna 2	9957	1400	2250	546.8	614.6	100.5
Zion 1	12000	1800	2250	530.2	597.0	128.9
Zion 2	12000	1800	2250	530.2	597.0	128.9
Donald C. Cook 1	*	*	2250	536.3	602.0	129.5
Donald C. Cook 2	12612	1800	2250	536.3	602.0	129.5
Trojan	12540	1800	2250	552.5	619.4	126.7
Salem 1	12811	1800	2250	544.4	611.9	126.4
McGuire 1	*	*	2250	558.1	620.8	140.3
McGuire 2	*	*	2250	558.1	620.8	140.3
Byron 1	12074	1800	2250	556.9	620.5	138.6
Byron 2	12074	1800	2250	556.9	620.5	138.6
Watts Bar 1	12145	1800	2250	559.1	617.3	144.8
Watts Bar 2	12145	1800	2250	559.1	617.3	144.8
Braidwood 1	12074	1800	2250	556.9	620.5	138.6
Braidwood 2	12074	1800	2250	556.9	620.5	138.6

Table 4 continued

Reactor Coolant System (RCS) Characteristics [4-17]

Plant Name	RCS Vol. Cu. Ft.	Pzr. Vol. Cu. Ft.	Pri. Pres. PSIA	Core Tin (F)	Core Tout (F)	Core Flow E6 lbm/hr
	Combustion	Engineering	(CE)	PWR	Plants	
Fort Calhoun	7066	900	2100	534.6	590.3	68.5
Maine Yankee	11026	1500	2250	538.9	592.0	117.5
Palisades	12400	1500	2100	545.0	592.0	121.3
Calvert Cliffs 1	11101	1500	2250	543.4	597.4	117.5
Calvert Cliffs 2	11101	1500	2250	543.4	597.4	117.5
Millstone 2	11015	1500	2250	538.9	594.9	117.5
St. Lucie 1	11101	1500	2250	538.9	594.9	117.5
Arkansas One 2	9976	1200	2250	553.5	614	116.2
San Onofre 2	*	1500	2250	553	611.3	148
San Onofre 3	*	1500	2250	553	611.3	148
	CE System 80 PWR Plants					
Palo Verde 1	*	1800	2250	564.5	621.2	164.0
Palo Verde 2	*	1800	2250	564.5	621.2	164.0
Palo Verde 3	*	1800	2250	564.5	621.2	164.0
	Babcock and Wilcox (B&W) PWR Plants					
Oconee 1	11500	1500	2200	554	606.2	131.32
Oconee 2	11500	1500	2200	554	606.2	131.32
Oconee 3	11500	1500	2200	554	606.2	131.32
Three Mile Island 1	11500	1500	2200	554	606.2	131.32
Three Mile Island 2	11500	1500	2200	557	610.6	137.8
Arkansas One 1	11500	1500	*	*	*	*
Rancho Seco	11500	1500	2200	557	610.6	137.8
Crystal River 3	11500	1500	*	*	*	*
Davis Besse 1	11500	1500	2200	555.4	611.7	131.32
WNP-1	12827	*	2250	572.3	631.6	150.5

* - Information unavailable at time of report preparation.

RCS Vol. Cu. Ft. - Total reactor coolant system volume, including the pressurizer, expressed in cubic feet.

Pzr. Vol. Cu. Ft. - Pressurizer volume expressed in cubic feet.

Pri. Pres. PSIA - Primary coolant pressure expressed in psia.

Core Tin (F) - Core inlet temperature expressed in F.

Core Tout (F) - Core outlet temperature expressed in F.

Core Flow E6 lbm/hr - Core coolant flow expressed in million (E6) lbm/hr.

Table 5

Primary Pressure Relief System Characteristics [4-17]

Plant Name	No. of PORVs	PORV SPT., PSIG	PORV Cap., lbm/hr/MWth	No. of SVs	SV SPT., PSIG	SV Cap., lbm/hr/MWth
Yankee Rowe	1	2400	118	2	2485	153
San Onofre 1	2	2190	80	2	2500	178.2
Haddam Neck	2	2270	115.1	3	2485	160.7
Robert E. Ginna	2	2335	117.8	2	2485	189.5
Point Beach 1	2	2335	117.9	2	2485	189.7
Point Beach 2	2	2335	117.9	2	2485	189.7
H. B. Robinson 2	2	2335	95.5	3	2485	130.9
Turkey Point 3	2	2335	95.1	3	2485	132.8
Turkey Point 4	2	2335	95.1	3	2485	132.8
Indian Point 2	2	2335	78.7	3	2485	147.9
Indian Point 3	2	2335	78.7	3	2485	138.9
Prairie Island 1	2	2335	108.5	2	2485	209.1
Prairie Island 2	2	2335	108.5	2	2485	209.1
Kewaunee	2	2335	106	2	2485	209.1
Surry 1	2	2335	86	2	2360	120.5
Surry 2	2	2335	86	2	2360	120.5
Beaver Valley 1	3	2335	79.9	3	2485	129.7
Joseph M. Farley 1	2	2335	79.2	3	2485	130.1
North Anna 1	2	2335	76	3	2485	137
Zion 1	2	2335	64.6	3	2485	129.2
Zion 2	2	2335	64.6	3	2485	129.2
Donald C. Cook 1	3	2335	64.6	3	2485	129.2
Donald C. Cook 2	3	2335	61.8	3	2485	123.5
Trojan	2	2350	61.6	3	2485	123.1
Salem 1	2	2350	63	3	2485	125.8
Byron 1	2	2335	61.6	3	2485	123.1
Byron 2	2	2335	61.6	3	2485	123.1
Watts Bar 1	2	2350	61.6	3	2485	123.1
Watts Bar 2	2	2350	61.6	3	2485	123.1
Braidwood 1	2	2335	61.6	3	2485	123.1
Braidwood 2	2	2335	61.6	3	2485	123.1
Combustion Engineering (CE) PWR Plants						
Yankee Rowe	1	2400	118	2	2485	153
Fort Calhoun	2	2392	69.7	2	2485	281.7
Maine Yankee	2	2385	57.0	3	2485	228.1

Table 5 continued

Primary Pressure Relief System Characteristics [4-17]

Plant name	No. of PORVs	PORV SPT., PSIG	PORV Cap., lbm/hr/MWth	No. of SVs	SV SPT., PSIG	SV Cap., lbm/hr/MWth
Palisades	2	2385	60.5	3	2485	272.7
Calvert Cliffs 1	2	2385	56.7	2	2485	222.2
Calvert Cliffs 2	2	2385	56.7	2	2485	222.2
Millstone 2	2	2380	59.8	2	2485	231.3
St. Lucie 1	2	2385	59.8	3	2485	234.4
Arkansas One 2	0	n/a	n/a	2	2500	280.6
San Onofre 2	*	*	*	*	2500	*
San Onofre 3	*	*	*	*	2500	*
CE System 80 PWR Plants						
Palo Verde 1	0	n/a	n/a	4	2500	*
Palo Verde 2	0	n/a	n/a	4	2500	*
Palo Verde 3	0	n/a	n/a	4	2500	*
Babcock and Wilcox (B&W) PWR Plants						
Oconee 1	1	2255	41.7	2	2500	254
Oconee 2	1	2255	41.7	2	2500	254
Oconee 3	1	2255	41.7	2	2500	254
Three Mile Island 1	1	2255	38.9	2	2435	243
Three Mile Island 2	1	2255	40.4	2	2450	249
Arkansas One 1	1	2255	38.9	2	2500	234
Rancho Seco	1	2255	40.4	2	2500	249
Crystal River 3	1	2255	40.8	2	2500	254
Davis Besse 1	1	2255	40.4	2	2435	242

* - Information unavailable at time of report preparation.

n/a - not applicable.

No. of PORVs - Number of pilot operated relief valves (PORVs).

PORV SPT., PSIG - PORV setpoint expressed in psig.

PORV Cap., lbm/hr/MWth - PORV capacity for each valve expressed in (lbm/hr)/MWth.

No. of SVs - Number of safety valves.

SV SPT., PSIG - safety valves setpoint expressed in psig.

SV Cap., lbm/hr/MWth - Safety valve capacity for each valve expressed in (lbm/hr)/MWth.

Table 6

High Pressure Injection System (HPIS) Characteristics [4-17]

Plant Name	No. of HPIS Pumps	Pump Cap. gpm @ 1000 psig	Pump Cap. gpm @ 1600 psig	HPIS Pump Drv	Act. SPT. psia	Delay Time, sec.
<hr/>						
Westinghouse (W) PWR Plants						
Yankee Rowe	*	99**	0	*	*	*
San Onofre 1	*	570	460	*	1685	*
Haddam Neck	*	800	575	*	1700	*
Robert E. Ginna	*	285	0	*	1715	*
Point Beach 1	*	900	0	*	1735	*
Point Beach 2	*	900	0	*	1735	*
H. B. Robinson 2	*	350	0	*	1715	*
Turkey Point 3	4	410	0	*	1715	25
Turkey Point 4	4	410	0	*	1715	25
Indian Point 2	*	485	0	*	1700	*
Indian Point 3	*	485	0	*	1700	*
Prairie Island 1	*	760	600	*	1815	*
Prairie Island 2	*	760	600	*	1815	*
Kewaunee	2	750	500	*	1815	25
Surry 1	*	520	420	*	1715	*
Surry 2	*	520	420	*	1715	*
Beaver Valley 1	*	495	380	*	1765	*
Joseph M. Farley 1	*	750	550	*	1850	*
North Anna 1	*	650	550	*	1765	*
North Anna 2	*	650	550	*	1765	*
Zion 1	2	490	380	elec.	1815	25
Zion 2	2	490	380	elec.	1815	25
Donald C. Cook 1	*	560	400	*	1815	*
Donald C. Cook 2	*	560	400	*	1900	*
Trojan	2	495	380	elec.	1765	25
Salem 1	*	490	380	*	1765	*
Byron 1	2	650***	*	elec.	*	*
Byron 2	2	650***	*	elec.	*	*
Watts Bar 1	2	650***	*	elec.	*	*
Watts Bar 2	2	650***	*	elec.	*	*
Braidwood 1	2	650***	*	elec.	*	*
Braidwood 2	2	650***	*	elec.	*	*
<hr/>						
Combustion Engineering (CE) PWR Plants						
Fort Calhoun	*	280	0	*	1600	*
Maine Yankee	*	715	550	*	1585	*

Table 6 continued

High Pressure Injection System (HPIS) Characteristics [4-17]

Plant Name	No. of HPIS Pumps	Pump cap. gpm @ 1000 psig	Pump cap. gpm @ 1600 psig	HPIS Pump Drv	Act. Spt. psia	Delay Time, sec.
Palisades	*	400	0	*	1593	*
Calvert Cliffs 1	*	400	0	*	1600	*
Calvert Cliffs 2	*	400	0	*	1600	*
Millstone 2	*	475	0	*	1620	*
St. Lucie 1	*	425	0	*	1600	*
Arkansas One 2	*	500	0	*	1725	*
San Onofre 2	3	415#	*	elec.	1950	*
San Onofre 3	3	415#	*	elec.	1950	*
GE System 80 PWR Plants						
Palo Verde 1	2	1130#	*	elec.	1950	*
Palo Verde 2	2	1130#	*	elec.	1950	*
Palo Verde 3	2	1130#	*	elec.	1950	*
Babcock and Wilcox (B&W) PWR Plants						
Oconee 1	3	500	450	*	1500	10
Oconee 2	3	500	450	*	1500	10
Oconee 3	3	500	450	*	1500	10
Three Mile Island 1	*	300	500	*	1515	*
Three Mile Island 2	*	500	450	*	1615	*
Arkansas One 1	*	500	450	*	*	*
Rancho Seco	*	500	450	*	1615	*
Crystal River 3	*	500	450	*	*	*
Davis Besse 1	2	700	200	*	1615	10
WNP-1	3	700##	*	*	*	*

* - Information unavailable at time of report preparation.

** - Safety injection pump flow in gal./min. at PORV setpoint.

*** - Safety injection pump flow in gal./min. at 700 psig discharge pressure.

- Design safety injection pump flow in gal./min.

- Safety injection pump flow at 1155 psig discharge pressure.

No. of HPIS Pumps - Number of HPIS pumps.

Pump cap. gpm @1000 psig - HPIS pump capacity expressed in gal./min. at 1000 psig discharge pressure.

Pump cap. gpm @1600 psig - HPIS pump capacity expressed in gal./min. at 1600 psig discharge pressure.

HPIS Pump Drv - HPIS pump motive drive, elec. = electric motor.

Table 6 continued

Act. SPt. psia - HPIS actuation setpoint pressure expressed in psia.

Delay Time, sec. - HPIS actuation delay time expressed in sec.

Table 7

Low Pressure Injection System (LPIS) Characteristics [4-17]

Plant Name	No. of Acc.	Min. Wtr. Vol. ea., cu. ft.	Act. Spt. psia	No. of LPIS Pumps	Pump Cap. gpm	LPIS Pump Drv.	LPIS Act. Spt. psia
Westinghouse (W) PWR Plants							
Turkey Point 3	3	720	665	2	3000	*	175
Turkey Point 4	3	720	665	2	3000	*	175
Kewaunee	2	1150	765	2	2000	*	185
Zion 1	4	850	665	2	3000	elec.	170
Zion 2	4	850	665	2	3000	elec.	170
Trojan	4	850	665	2	3000	elec.	170
Byron 1	4	1000	665	2	3000	elec.	*
Byron 2	4	1000	665	2	3000	elec.	*
Watts Bar 1	4	1000	455	2	3000	elec.	*
	2**	1800	1065				
Watts Bar 2	4	1000	455	2	3000	elec.	*
	2**	1800	1065				
Braidwood 1	4	1000	665	2	3000	elec.	*
Braidwood 2	4	1000	665	2	3000	elec.	*
Combustion Engineering (CE) PWR Plants							
San Onofre 2	4	2200	625	2	4150	elec.	665
San Onofre 3	4	2250	625	2	4150	elec.	665
CE System 80 PWR Plants							
Palo Verde 1	4	1790	615	2	5000	elec.	665
Palo Verde 2	4	1790	615	2	5000	elec.	665
Palo Verde 3	4	1790	615	2	5000	elec.	665
Babcock and Wilcox (B&W) PWR Plants							
Oconee 1	2	1010	615	2	3000	*	*
Arkansas One 1	2	905	615	2	3000	elec.	1515
Rancho Seco	2	1010	615	2	3000	elec.	1615
Crystal River 3	2	1020	615	2	3000	elec.	515
Davis Besse 1	2	1010	615	2	3000	*	*
WNP-1	2	1350	615	2	5000	*	*

* - Information unavailable at time of report preparation.

** - Upper head injection accumulators.

No. of Acc. - Number of accumulators or core flood tanks.

Min. Wtr. Vol. ea., cu. ft. - Minimum water volume in accumulators or core flood tanks as per technical specifications expressed in cubic feet.

Act. Spt. psia - Actuation setpoint for accumulators or core flood tanks expressed in psia.

Table 7 continued

No. of LPIS Pumps - Number of LPIS Pumps.

Pump Cap. gpm - LPIS pump capacity expressed in gal./min.

LPIS Pump Drv. LPIS pump motive drive, elec. = electric motor.

LPIS Act. SPt. psia - LPIS pump actuation setpoint expressed in psia.

Table 8

Steam Generator Characteristics I [4-6]

Plant Name	Tot. Pri. Cu. Ft.	Tot. Sec. Vol. Cu. Ft.	Ht. Tran. Area Sq. Ft.	Steam Generator Type	SG Dryout Time, Min.
Westinghouse (W) Series 27					
Yankee Rowe	553	2592	27700	VS&UT	*
San Onofre 1	553	2592	27700	VS&UT	*
Haddam Neck	553	2592	27700	VS&UT	*
Westinghouse (W) Series 44					
Robert E. Ginna	944	4580	44430	VV&UT	*
Point Beach 1	944	4580	44430	VS&UT	*
Point Beach 2	944	4580	44430	VS&UT	*
B. H. Robinson 2	944	4580	44430	VS&UT	*
Turkey Point 3	944	4580	44430	VS&UT	*
Turkey Point 4	944	4580	44430	VS&UT	*
Indian Point 2	944	4580	44430	VS&UT	*
Indian Point 3	944	4580	44430	VS&UT	1.22
Westinghouse (W) Series 51					
Prairie Island 1	1080	5868	51500	VS&UT	*
Prairie Island 2	1080	5868	51500	VS&UT	*
Kewaunee	1080	5868	51500	VS&UT	*
Surry 1	1080	5868	51500	VS&UT	*
Surry 2	1080	5868	51500	VS&UT	*
Beaver Valley 1	1080	5868	51500	VS&UT	*
Beaver Valley 2	1080	5868	51500	VS&UT	*
North Anna 1	1080	5868	51500	VS&UT	*
North Anna 2	1080	5868	51500	VS&UT	*
Zion 1	1080	5868	51500	VS&UT	*
Zion 2	1080	5868	51500	VS&UT	*
Donald G. Cook 1	1080	5868	51500	VS&UT	1.17
Donald G. Cook 2	1080	5868	51500	VS&UT	*
Trojan	1080	5868	51500	VS&UT	*
Salem 1	1080	5868	51500	VS&UT	*
Salem 2	1080	5868	51500	VS&UT	*
Combustion Engineering (CE)					
Fort Calhoun 1	850	4411	47660	VS&UT	*
Maine Yankee	1032	5139	58165	VS&UT	*
Palisades	1572	7074	79310	VS&UT	1.55
Calvert Cliffs 1	1646	8006	90232	VS&UT	*
Calvert Cliffs 2	1646	8006	90232	VS&UT	*
Millstone 2	1527	8006	90232	VS&UT	1.94
St. Lucie 1	1527	8006	90232	VS&UT	*
Arkansas One 2	1588	7957	86116	VS&UT	*

Table 8 continued

Steam Generator Characteristics I [4-6]

Plant Name	Tot. Pri. Vol. Cu. Ft.	Tot. Sec. Vol. Cu. Ft.	Ht. Tran. Area Sq. Ft.	Steam Generator Type	DG Dryout Time, Min.
	Babcock and Wilcox (B&W)				
Oconee 1	*	*	*	otsg	0.48
Oconee 2	*	*	*	otsg	0.48
Oconee 3	*	*	*	otsg	0.48
Three Mile Island 1	*	*	*	otsg	0.48
Three Mile Island 2	*	*	*	otsg	0.45
Arkansas One 1	*	*	*	otsg	0.48
Rancho Seco	*	*	*	otsg	0.45
Crystal River 3	*	*	*	otsg	0.50
Davis Besse 1	*	*	*	otsg	0.45

VS&UT - Vertical shell and U-tube steam generator.

OTSG - Vertical once-through shell and tube steam generator.

* - Information unavailable at time of report preparation.

Tot. Pri. Vol. Cu. Ft. - Total primary side volume, expressed in cubic feet.

Tot. Sec. Vol. Cu. Ft. - Total secondary side volume, expressed in cubic feet.

Ht. Tran. Area Sq. Ft. - Unfouled heat transfer area, expressed in square feet.

SG Dryout Time, Min. - Steam generator secondary inventory dryout time with loss of all feedwater supply at hot full power (no reactor scram), expressed in minutes.

Table 9

Steam Generator (S/G) Characteristics II [7-18]

Plant Name	Oper Pres. psi. Prim./Sec.	Oper Temp. F Prim./Sec.	Cool. Flw. E6 lbm/hr Prim./Sec.	No. of S/G Tubes	Ht. Trn. Area Sq. Ft.
Westinghouse (W) PWR Plants					
Point Beach 1	2250/1035	*	33.4/*	*	*
Point Beach 2	2250/1035	*	33.4/*	*	*
H. B. Robinson 2	2250/1035	*	33.9/*	*	*
Indian Point 2	2250/1120	*	34.1/*	*	*
Zion 1	2250/1100	600/506.3	33.8/14.0	3388	51500
Zion 2	2250/1100	600/506.3	33.8/14.0	3388	51500
Trojan	*	*	33.8/*	3388	51500
Byron 1	*	*	*	4578	48300
Byron 2	*	*	*	4578	48300
Braidwood 1	*	*	*	4578	48300
Braidwood 2	*	*	*	4578	48300
Combustion Engineering (CE) PWR Plants					
San Onofre 2	2250/1000	560/532	74/7.57	*	*
San Onofre 3	2250/1000	560/532	74/7.57	*	*
CE System 80 PWR Plants					
Palo Verde 1	2250/1070	*	82.0/8.59	*	*
Palo Verde 2	2250/1070	*	82.0/8.59	*	*
Palo Verde 3	2250/1070	*	82.0/8.59	*	*
Babcock and Wilcox (B&W) PWR Plants					
Oconee 1	2185/910	*/570	65.7/5.6	15530	*
Rancho Seco	2185/910	*/570	68.9/6.12	*	*
Davis Besse	2185/910	*	65.7/*	*	*
WNP-1	2235/1060	*/603	75.2/7.43	16000	*

* - Information unavailable at time of report preparation.

Oper Pres. psia Prim./Sec. - Steam generator operating pressure expressed in psia for primary (tube) and for secondary (shell) sides.

Oper Temp. F Prim./Sec. - Steam generator operating temperature expressed in F for primary (tube) and for secondary (shell) sides.

Cool Flw. E6 lbm/hr Prim./Sec. - Steam generator coolant flow rate expressed in million (E6) lbm/hr for primary (tube) and for secondary (shell) sides.

No. of S/G Tubes - Number of steam generator heat transfer tubes.

Ht. Trn. Area Sq. Ft. - Unfouled heat transfer area expressed in square feet.

Table 10

Main Feedwater (MFW) System Characteristics [4-16]

Plant Name	No. of MFW Pumps	Pump Cap., gpm @ hd. ft.	% Cap.	Shtf hd ft.	MFW Pump Drv	No. of Con. Pumps	No. of Bstr Pumps
Westinghouse (W) PWR Plants							
Yankee Rowe	3	21600@1280	*	2110	elec.	3	0
San Onofre 1	2	14000@2000	50	2720	elec.	4	0
Baddam Neck	2	9600@*	50	2570	elec.	2	0
Robert E. Ginna	2	14000@2000	50	2720	elec.	3	3
Point Beach 1	2	7800@2170	50	2450	elec.	2	0
Point Beach 2	2	7800@2170	50	2450	elec.	2	0
H. B. Robinson 2	2	12690@*	*	2400	elec.	2	0
Turkey Point 3	2	13000@1880	60	2650	elec.	2	3
Turkey Point 4	2	13000@1880	60	2650	elec.	2	3
Indian Point 2	2	15300@2270	*	1830	stm.	3	0
Indian Point 3	2	15300@2270	*	1830	stm.	3	0
Prairie Island 1	2	7.1 E6**	65	2450	elec.	3	0
Prairie Island 2	2	7.1 E6**	65	2450	elec.	3	0
Kewaunee	2	10000@*	60	2520	elec.	2	0
Surry 1	2	13800@*	50	1700	elec.	3	0
Surry 2	2	13800@*	50	1700	elec.	3	0
Beaver Valley 1	2	15200@1900	*	2450	elec.	2	0
Joseph M. Farley 1	2	15000@*	*	3400	stm.	3	0
North Anna 1	3	16250@*	50	2775	elec.	3	0
Zion 1	3	15800@2710	50	3720	2 stm. 1 elec.	4	4
Zion 2	3	15800@2710	50	3720	2 stm. 1 elect	4	4
Donald C. Cook 1	2	16750@*	70	2625	stm.	2	0
Donald C. Cook 2	2	16750@*	70	2625	stm.	2	0
Trojan	2	19800@2020	70	3250	stm.	2	0
Salem 1	2	18613@2070	50	2930	stm.	3	0
Byron 1	2	*	*	*	stm.	4	4
	1	*	*	*	elec.		
Byron 2	2	*	*	*	stm.	4	4
	1	*	*	*	elec.		
Watts Bar 1	2	17630@*	*	2012	stm.	3	3
	1	6100@*	*	1890	elec.		
Watts Bar 2	2	17630@*	*	2012	stm.	3	3
	1	6100@*	*	1890	elec.		
Braidwood 1	2	*	*	*	stm.	4	4
	1	*	*	*	elec.		

Table 10 continued

Main Feedwater (MFW) System Characteristics [4-16]

Plant Name	No. of MFW Pumps	Pump Cap., gpm @ hd. ft.	% Cap.	Shft hd ft.	MFW Pump Drv	No. of Con. Pumps	No. of Bstr Pumps
Braidwood 2	2 1	* *	* *	* *	stm. elec.	4	4
Combustion Engineering (CE) PWR Plants							
Fort Calhoun	3	8000@1740	*	2370	elec.	3	0
Maine Yankee	2	14000@2410	60	3590	elec.	3	0
Palisades	2	135000	*	2640	stm.	2	0
Calvert Cliffs 1	2	15000@*	50	2710	stm.	3	3
Calvert Cliffs 2	2	15000@*	50	2710	stm.	3	3
Millstone 2	2	15000@*	55	2100	stm.	3	0
St. Lucie 1	2	14100@*	60	1780	elec.	2	0
Arkansas One 2	2	14500@*	*	2700	stm.	4	0
San Onofre 2	2	18800@*	50	*	stm.	4	0
San Onofre 3	2	18800@*	50	*	stm.	4	0
CE System 80 PWR Plants							
Palo Verde 1	*	*	*	*	stm.	3	*
Palo Verde 2	*	*	*	*	stm.	3	*
Palo Verde 3	*	*	*	*	stm.	3	*
Babcock and Wilcox (B&W) PWR Plants							
Oconee 1	2	25000@2440	75	2890	stm.	3	3
Oconee 2	2	25000@2440	75	2890	stm.	3	3
Oconee 3	2	25000@2440	75	2890	stm.	3	3
Three Mile Island 2	*	15500@2440	*	*	stm.	3	3
Arkansas One 1	2	14750@1901	*	2540	stm.	3	0
Rancho Seco	2	6 E6 lb/hr	*	3200	stm.	3	0
Crystal River 3	2	13300@2280	*	2550	stm.	2	2
Davis Besse 1	2	15000@2150	*	2560	stm.	3	2***

* - Information unavailable at time of report preparation.

** - Pump capacity given in lbm./hr.

*** - Same shaft as main feedwater pumps but geared down to 1780 rpm (500 ft. hd.).

No. of MFW Pumps - Number of main feedwater pumps.

Pump Cap., gpm @hd. ft. - Main feedwater pump capacity expressed in gal./min., at discharge head expressed in feet.

% Cap. - Main feedwater pump capacity expressed in percent of total requirement.

Table 10 continued

Shtf Hd Ft. - Shutoff head expressed in feet.

MFW Pump Drv - Mainfeedwater pump motive drive, elec. = electric motor, stm. = steam turbine.

No. of Con Pumps - No. of condensate pumps.

No. of Bstr Pumps - No. of condensate booster pumps.

Table 11

**Auxiliary Feedwater (AFW) System Characteristics
Westinghouse (W) PWR Plants [4, 7-10]**

Plant Name	No. of Stm. Dr. Pumps	Stm. Dr. Pump Cap. gpm @ hd. ft.	No. of Motor Dr. Pumps	Motor Dr. Pump Cap. gpm @ hd. ft.	AFW Int. Mode
Yankee Rowe	1*	90 @ 2770	0	n/a	man.
San Onofre 1	1	300 @ 2560	1	235 @ 2380	man.
Haddam Neck	2	450 @ 2300	0	n/a	man.
Robert E. Ginna	1	400 @ 2640	2	200 @ 2600	auto.
			2	200 @ **	man.
Point Beach 1	1***	400 @ 2780	1***	200 @ 2780	auto.
Point Beach 2	1***	400 @ 2780	1***	200 @ 2780	auto.
H. B. Robinson 2	1	600 @ 3000	2	300 @ 3000	auto.
Turkey Point 3	3#	600 @ 2775	0	n/a	auto.
Turkey Point 4	3#	600 @ 2775	0	n/a	auto.
Indian Point 2	1	800 @ 3140	2	400 @ 3140	auto.
Indian Point 3	1	800 @ 3140	2	400 @ 3140	auto.
Prairie Island 1	1##	220 @ 2800	1##	220 @ 2800	auto.
Prairie Island 2	1##	220 @ 2800	1##	220 @ 2800	auto.
Kewaunee	1	240 @ 2850	2	240 @ 2850	auto.
Surry 1	1###	700 @ 2730	2###	350 @ 2730	auto.
Surry 2	1###	700 @ 2730	2###	350 @ 2730	auto.
Beaver Valley 1	1	700 @ 2696	2	350 @ 2696	auto.
Joseph M. Farley 1	1	700 @ 2960	2	350 @ 2960	auto.
North Anna 1	1	700 @ 2800	2	350 @ 2800	auto.
Zion 1	1	900 @ 3099	2	450 @ 3099	auto.
Zion 2	1	900 @ 3099	2	450 @ 3099	auto.
Donald C. Cook 1	1***	900 @ 2714	1***	450 @ 2714	auto.
Donald C. Cook 2	1***	900 @ 2714	1***	450 @ 2714	auto.
Table 11 continued					
Trojan	1	960 @ 3400	1 diesel	960 @ 3400	auto.
Salem 1	1	380 @ 3570	2	440 @ 3000	auto.
Byron 1	1 diesel	890 @ 3350	1	890 @ 3350	auto.
Byron 2	1 diesel	890 @ 3350	1	890 @ 3350	auto.
Watts Bar 1	1	940 @ 2950	2	470 @ 2950	auto.
Watts Bar 2	1	940 @ 2950	2	470 @ 2950	auto.
Braidwood 1	1 diesel	890 @ 3350	1	890 @ 3350	auto.
Braidwood 2	1 diesel	890 @ 3350	1	890 @ 3350	auto.

Table 11 continued

n/a - not applicable.

* - Charging and safety injection systems serve as backup.

** - Information unavailable at time of report preparation.

*** - Per unit motor pump supplies both units.

- One pump normally supplies each unit - 3rd pump is backup for either unit.

- Per unit motor pump normally feeds opposite unit steam generator.

- One pump each AFW system can feed opposite unit.

No. Stm. Dr. Pumps - Number of steam turbine driven AFW pumps.

Stm. Dr. Pump Cap. gpm @ hd. ft. - Steam driven AFW pump capacity expressed in gal./min. at discharge head expressed in feet.

No. of Motor Dr. Pumps - Number of electric motor driven AFW pumps.

Motor Dr. Pump Cap. gpm @ hd. ft. - Electric motor driven AFW pump capacity expressed in gal./min. at discharge head expressed in feet.

AFW Int. Mode - AFW initiation mode, manual or automatic.

Table 12

Auxiliary Feedwater (AFW) System Characteristics
Combustion Engineering (CE) PWR Plants [5, 11-13]

Plant Name	No. of Stm. Dr. Pumps	Stm. Dr. Pump Cap. gpm @ hd. ft.	No. of Motor Dr. Pumps	Motor Dr. Pump Cap. gpm @ hd. ft.	AFW Int. Mode
Fort Calhoun	1	260 @ 2400	1	260 @ 2400	Semi*
Maine Yankee	1	500 @ 2570	2	1500 @ 2570	man.
Palisades	1	415 @ 2730	1	415 @ 2730	man.
Calvert Cliffs 1	2	700 @ 2530	0	n/a	man.
Calvert Cliffs 2	2	700 @ 2530	0	n/a	man.
Millstone 2	1	600 @ 2437	2	300 @ 2437	man.
St. Lucie 1	1	500 @ 2760	2	250 @ 2760	man.
Arkansas One 2	1	575 @ 2800	1	575 @ 2800	auto.
San Onofre 2	1	860 @ **	2	860 @ **	**
San Onofre 3	1	860 @ **	2	860 @ **	**
CE System 80					
Palo Verde 1	1	875 @ **	2	875 @ **	**
Palo Verde 2	1	875 @ **	2	875 @ **	**
Palo Verde 3	1	875 @ **	2	875 @ **	**

n/a - not applicable.

* - Semiautomatic motor-driven pump manually connected to diesel generator.

** - Information unavailable at time of report preparation.

No. Stm. Dr. Pumps - Number of steam turbine driven AFW pumps.

Stm. Dr. Pump Cap. gpm @ hd. ft. - Steam driven AFW pump capacity expressed in gal./min. at discharge head expressed in feet.

No. of Motor Dr. Pumps - Number of electric motor driven AFW pumps.

Motor Dr. Pump Cap. gpm @ hd. ft. - Electric motor driven AFW pump capacity expressed in gal./min. at discharge head expressed in feet.

AFW Int. Mode - AFW Initiation mode, manual or automatic.

Table 13

Emergency Feedwater (EFW) System Characteristics
Babcock and Wilcox (B&W) PWR Plants [6, 14-16]

Plant Name	No. of Stm. Dr. Pumps	Stm. Dr. Pump Cap. gpm @ hd. ft.	No. of Motor Dr. Pumps	Motor Dr. Pump Cap. gpm @ hd. ft.	EFW Int. Mode
Oconee 1	1	1080 @ 2450	0	n/a	auto.
Oconee 2	1	1080 @ 2450	0	n/a	auto.
Oconee 3	1	1080 @ 2450	0	n/a	auto.
Three Mile Island 2	1	940 @ 2600	2	470 @ 2560	auto.
Arkansas One 1	1	780 @ 2600	1	780 @ 2600	auto.
			1*	1150 @ 1100	
Rancho Seco	1**	840 @ 2650	1	840 @ 2650	auto.
Crystal River 3	1	740 @ 3000	1	740 @ 3000	auto.
Davis Besse 1	2	1050 @ 2500	0	n/a	auto.

n/a - not applicable.

* - Auxiliary (nonsafety grade) feedwater pump used for startup operations up to 5% power. Also backup to main feedwater system.

** - Steam turbine-motor driven tandem.

No. of Stm. Dr. Pumps - Number of steam turbine driven EFW pumps.

Stm. Dr. Pump Cap. gpm @ hd. ft. - Steam driven EFW pump capacity expressed in gal./min. at discharge head expressed in feet.

No. of Motor Dr. Pumps - Number of electric motor driven EFW pumps.

Motor Dr. Pump Cap. gpm @ hd. ft. - Electric motor driven EFW pump capacity expressed in gal./min. at discharge head expressed in feet.

EFW Int. Mode - EFW initiation mode, manual or automatic.

Table 14

Containment Spray Injection System (CSIS) Characteristics [7-17]

Plant Name	No. of CSIS Trains	No. of Spray Headers	No. of Spray Pumps	CSIS Pump Cap. gal/min	CSIS Pump Drive
Westinghouse (W) PWR Plants					
Zion 1	3	6	3	3000	2-elec., 1-dsl.
Zion 2	3	6	3	3000	2-elec., 1 dsl.
Trojan	2	2	2	2800	elec.
Byron 1	2	6	1	3550	elec.
			1	4060	elec.
Byron 2	2	6	1	3550	elec.
			1	4060	elec.
Watts Bar 1	2	2	2	4000	elec.
Watts Bar 2	2	2	2	4000	elec.
Braidwood 1	2	6	1	3550	elec.
			1	4060	elec.
Braidwood 2	2	6	1	3550	elec.
			1	4060	elec.
Combustion Engineering (CE) PWR Plants					
San Onofre 2	2	2	2	1750	elec.
San Onofre 3	2	2	2	1750	elec.
GE System 80 PWR Plants					
Palo Verde 1	2	2	2	3890	elec.
Palo Verde 2	2	2	2	3890	elec.
Palo Verde 3	2	2	2	3890	elec.
Babcock and Wilcox (B&W) PWR Plants					
Oconee 1	*	*	2	1500	elec.
Arkansas One 1	2	2	2	1500	elec.
Rancho Seco	2	2	2	1500	elec.
Crystal River 3	2	2	2	1500	elec.
Davis Besse 1	*	*	2	1500	elec.
WNP-1	*	*	2	2500	elec.

*Information unavailable at time of report preparation.

No. of CSIS Trains - Number of CSIS trains.

No. of Spray Headers - Total number of CSIS spray headers.

No. of Spray Pumps - Total number of CSIS pumps.

CSIS Pump Cap. gal/min - CSIS pump capacity expressed in gal./min.

CSIS Pump Drive - CSIS pump motive drive, elec. = electric motor,
dsl. = diesel motor.

Table 15

Containment Fan Cooler System (CFCS) Characteristics [7-17]

Plant Name	No. of Circ. Trns.	No. of Circ. Fans	Circ. Fan Cap.		Heat Removal Cap.	
			Normal (cfm)	/Accident (cfm)	Normal (btu./hr)	/Accident (btu/hr)
Westinghouse (W) PWR Plants						
Zion 1	5	5	85000	53000	3.15 E6	81 E6
Zion 2	5	5	85000	53000	3.15 E6	81 E6
Trojan	2	8	100000	86000	2.085 E6	76.2 E6
Byron 1	2	4	93600	*	1.94 E6	132 E6
Byron 2	2	4	93600	*	1.94 E6	132 E6
Watts Bar 1	2	*	14000	*	*	*
Watts Bar 2	2	*	14000	*	*	*
Braidwood 1	2	4	93600	*	1.94 E6	132 E6
Braidwood 2	2	4	93600	*	1.94 E6	132 E6
Combustion Engineering (CE) PWR Plants						
San Onofre 2	2	4	*	31000	*	70 E6
San Onofre 3	2	4	*	31000	*	70 E6
CE System 80 PWR Plants						
Palo Verde 1	0	0	n/a	n/a	n/a	n/a
Palo Verde 2	0	0	n/a	n/a	n/a	n/a
Palo Verde 3	0	0	n/a	n/a	n/a	n/a
Babcock and Wilcox (B&W) PWR Plants						
Oconee 1	3	*	*	*	*	80 E6
Arkansas One 1	4	4	*	*	*	*
Rancho Seco	4	4	*	40000	*	60 E6
Crystal River 3	3	3	108000	54000	2.15 E6	80 E6
Davis Besse 1	3	*	*	*	*	75 E6
WNP-1	4	*	*	*	*	*

* - Information unavailable at time of report preparation.

No. of Circ. Trns. - Number of CFCS trains.

No. of Circ. Fans - total number of CFCS fans.

Circ. Fan Cap. Normal (cfm)/Accident (cfm) - CFCS fan capacity for normal and accident conditions, expressed in cubic feet/min.

Heat Removal Cap. Normal (btu/hr)/Accident (btu/hr) - CFCS heat removal capacity for normal and accident conditions, expressed in btu/hr.

Table 16

Containment Characteristics [7-17]

Plant Name	Cntnmt. Type	Tot. Int. Volume cu. ft.	Containment Pressure Normal (psig)	Accdnt (psig)	Design (psig)	Cntnmt. Temp F	Cntnmt. Spprsn. Type
Westinghouse (W) PWR Plants							
Zion 1	dry	*	0	47	47	271	spray
Zion 2	dry	*	0	47	47	271	spray
Trojan	dry	2.222 E6	0	52.9	60	288	spray
Byron 1	dry	2.809 E6	0	43.6	50	*	spray
Byron 2	dry	2.809 E6	0	43.6	50	*	spray
Watts Bar 1	ice	1.191 E6	0.3	12.3	15	250	i & s
Watts Bar 2	ice	1.191 E6	0.3	12.3	15	250	i & s
Braidwood 1	dry	2.809 E6	0	43.6	50	*	spray
Braidwood 2	dry	2.809 E6	0	43.6	50	*	spray
Combustion Engineering (CE) PWR Plants							
San Onofre 2	dry	2.366 E6	0	60	60	300	spray
San Onofre 3	dry	2.366 E6	0	60	60	300	spray
CE System 80 PWR Plants							
Palo Verde 1	dry	2.6 E6	0	60	60	300	spray
Palo Verde 2	dry	2.6 E6	0	60	60	300	spray
Palo Verde 3	dry	2.6 E6	0	60	60	300	spray
Babcock and Wilcox (B&W) PWR Plants							
Oconee 1	dry	1.9 E6	*	*	59	*	spray
Arkansas One 1	dry	*	*	*	*	*	spray
Rancho Seco	dry	1.98 E6	0	59	59	286	spray
Crystal River 3	dry	*	0	53.9	55	*	spray
Davis Besse 1	dry	2.8 E6	*	*	40	*	spray
WNP-1	dry	3.09 E6	*	*	52.0	*	spray

* - Information unavailable at time of report preparation.

Cntnmt. Type - Containment type.

Tot. Int. Volume cu. ft. - Total internal volume, expressed in cubic feet.

Containment Pressure, Normal, Accdnt, Design (psig) - Containment pressure for normal, accident and design conditions, expressed in psig.

Cntnmt. Temp F - Containment temperature expressed in F.

Cntnmt. Spprsn. Type - Containment suppression type, i & s = ice and spray suppression.

Table 17

Emergency Electrical Power Source Characteristics [1-17]

Plant Name	Type	Number of Units		Capacity each
		Dedicated	Shared	
Westinghouse (W) PWR Plants				
Zion 1	diesels	2	1	*
Zion 2	diesels	2	1	*
Trojan	diesels	2	n/a	4418 kW @ 0.8 pf.
Byron 1	diesels	2	0	*
Byron 2	diesels	2	0	*
Watts Bar 1	diesels	2	0	*
Watts Bar 2	diesels	2	0	*
Braidwood 1	diesels	2	0	*
Braidwood 2	diesels	2	0	*
Combustion Engineering (CE) PWR Plants				
San Onofre 2	diesels	2	0	*
San Onofre 3	diesels	2	0	*
CE System 80 PWR Plants				
Palo Verde 1	diesels	2	0	*
Palo Verde 2	diesels	2	0	*
Palo Verde 3	diesels	2	0	*
Babcock and Wilcox (B&W) PWR Plants				
Oconee 1	hydroelectric	2	0	87.5 MVA
Rancho Seco	diesels	2	n/a	2750 kW @ 0.8 pf.
Davis Besse 1	diesels	2	0	2600 kW
WNP-1	diesels	2	0	2600 kW

* - information unavailable at time of report preparation.

3) FUNCTIONAL EVENT TREE DEVELOPMENT

The Reactor Safety Study [19] defined two types of event trees to describe reactor accidents, the functional event tree and the systemic event tree.

The functional event tree is used to describe the system functions that must work or not work in order for a certain accident sequence to happen. Two functional event trees have been defined for this study: one for pipe breaks (all loss of coolant accidents or LOCAs), and one for transients. The following functions have been defined for the functional event trees: 1) Reactor trip (RT), 2) Power conversion system (PCS), 3) Secondary heat removal system (SHRS), 4) Emergency core cooling system (ECCS), 5) Post accident radioactivity removal, 6) Post accident heat removal, and 7) Emergency coolant recirculation. Figure 1 shows the functional event tree defined for pipe breaks (LOCAs), and Figure 2 shows the functional event tree defined for transients. Both of these functional event trees present the basic sequence of functional events that occur following the initiating event. The plant-specific systemic event tree should be developed from the functional event trees following the guidelines given in Appendix I of the WASH-1400 report [18].

The systemic event tree describes the specific plant safety systems that perform the system functions defined by the functional event tree. For example, the emergency core cooling (ECC) function can be successfully achieved by the high pressure injection system (HPIS), the low pressure injection system (LPIS), by the accumulators, or by a combination of these systems depending on the specific PWR design and the initiating event. Table 18 presents the relationship between the functional event categories and the specific reactor safety systems. This table was developed from Appendix I of the WASH-1400 report [19] and from the SSMRP report [1].

Table 18

Functional Event - Reactor Safety System Relationship [1, 19]

Initiating Event	RT	PCS	SHRS	ECCS	PARR	PAHR	ECR
Reactor Vessel Rupture (RVR)	n/a	n/a	n/a	n/a	CSIS, CFCS(I),	CFCS(R), CSRS,	RHRS,
Large Loca (LLOCA) p.b.> 6" I.D.	n/a	n/a	n/a	ACC, LPIS,	CSIS, CFCS(I),	CFCS(R), CSRS,	RHRS,
Medium LOCA (MLOCA) 6" > p.b. > 3" I.D.	RPS	n/a	n/a	ACC, LPIS, CVCS-CP,	CSIS, CFCS(I),	CFCS(R), CSRS,	RHRS, HPIS, CVCS-CP,
Small LOCA (SLOCA) 3" > p.b. > 1.5" I.D.	RPS	n/a	n/a	HPIS, CVCS-CP,	CSIS, CFCS(I),	CFCS(R), CSRS,	RHRS, HPIS, CVCS-CP,
Small-Small LOCA (SSLOCA) 1.5" > p.b. > 0.5" I.D.	RPS	n/a	AFWS SSR F&B	HPIS, CVCS-CP,	CSIS, CFCS(I),	CFCS(R), CSRS,	RHRS, HPIS, CVCS-CP,
Transients (T1 or T2)	RPS	MT MFWS	AFWS SSR PR-PORVs F&B	CVCS-CP,	n/a	n/a	RHRS,

Functional Event Categories

Specific Reactor Safety Systems

RT - Reactor Trip,

RPS - Reactor Protection System,

PCS - Power Conversion System,

MT - Main Turbine (including main condenser steam dump),

SHRS - Secondary heat removal system,

MFWS - Main feedwater system,

ECCS - Emergency core cooling system,

AFWS - Auxiliary and/or emergency feedwater system,

PARR - Post accident radioactivity removal,

SSR - Secondary steam relief system,

PAHR - Post accident heat removal,

PR-PORVs - Pressurizer PORVs,

ECR - Emergency coolant recirculation,

F&B - Feed and bleed operation,

ACC - Accumulators, safety injection or core flood tanks,

LPIS - Low pressure injection system,

Table 18 continued

HPIS - High pressure injection system,

CVCS-CP - Chemical and volume control system-charging pumps,

CSIS - Containment spray injection system,

CFCS(I) - Containment fan cooler system (injection mode),

CFCS(R) - Containment fan cooler system (recirculation mode),

RHRS - Residual heat removal system (usually includes LPIS and RHR heat exchangers),

n/a - not applicable.

Figure 1
LOCA Functional Event Tree

Figure 2
Transient Functional Event Tree

4) DOMINANT ACCIDENT SEQUENCES DEVELOPMENT

This section describes the method used to identify the dominant accident sequences. For the development of the dominant accident sequences, the following terms are defined. A dominant event sequence is defined as any event sequence whose probability of radioactive release is greater than or equal to $1 \text{ E-}10/\text{yr}$ at any earthquake level. A dominant accident sequence is defined as those dominant event sequences summed over all containment failure modes. The event sequence probabilities for radioactive release were generated from SEISIM2 computer code runs from the probabilistic seismic risk assessment of the Zion Unit 1 PWR plant for event sequences with and without feed and bleed (F&B) operation [1].

When a probability of radioactive release of $1 \text{ E-}10/\text{yr}$ is used as the basis of a dominant event sequence at each earthquake level, SEISIM predicts a radioactive release risk of $1.95 \text{ E-}3 \text{ manrem/yr}$. If this radioactive release risk is assumed to occur at each of the six earthquake levels defined by SEISIM, a total radioactive release risk for a dominant event sequence due to all earthquake levels is $1.17 \text{ E-}2 \text{ manrem/yr}$. This figure can be considered the maximum amount of radioactive release risk that would not be considered if an event sequence with a probability of radioactive release of $1 \text{ E-}10/\text{yr}$ or less were to be disregarded. Based on the permissible levels of radiation of 0.5 manrem/yr in unrestricted areas as specified in 10CFR20.105 [20], the total radioactive risk of $1.17 \text{ E-}2 \text{ manrem/yr}$ from a dominant event sequence can indeed be considered small enough to be disregarded.

The probabilities of the dominant event sequences at each earthquake level were summed over all earthquake levels to obtain the probability of radioactive release due to any seismic event. Table 19 shows the dominant event sequences radioactive release risk summed over all earthquake levels with feed and bleed (F&B) operation. The total radioactive release risk from the 35 dominant event sequences with F&B capability is 5.26 manrem/yr . This is equivalent to 87% of the total radioactive release risk when all event sequences involving F&B capability are considered.

Table 20 shows the dominant event sequences radioactive release risk summed over all earthquake levels without feed and bleed (F&B) operation. The total radioactive release risk from the 36 dominant event sequences without F&B capability considered

is 6.94 manrem/yr. This is equivalent to 92% of the total radioactive release risk when all event sequences not involving F&B capability are considered.

In compiling these lists of dominant event sequences, it was noted that many dominant event sequences were similar, differing only by the containment failure mode. The dominant event sequences were then summed over all containment failure modes to obtain the radioactive release risk from a particular accident sequence. Table 21 shows the dominant accident sequences involving F&B capability and not involving F&B capability. These dominant accident sequences considered more event sequences than those listed in Tables 19 and 20 because event sequences similar to those in Tables 19 and 20 but with a different containment failure mode and a probability of radioactive release less than $1 \text{ E-}10/\text{yr}$ are now included. The total radioactive release risk from the 18 dominant accident sequences with F&B capability considered is 5.28 manrem/yr. This is equivalent to 92% of the total radioactive release risk when all of the accident sequences involving F&B capability are considered. The total radioactive release risk from the 17 dominant accident sequences not involving F&B capability is 6.97 manrem/yr. This is equivalent to 95% of the total radioactive release risk when all of the accident sequences not involving F&B are considered.

Each event sequence consists of several component events which involve the successful operation or failure of a reactor safety system or an operator initiated action. These component events are shown in Table 22 for dominant accident sequences involving F&B operation, and in Table 23 for dominant accident sequences not involving F&B operation. Those events with bars over them denote successful system operation or operator action. Note that for all dominant accident sequences of Tables 22 and 23, event K (operation of the reactor protection system), event J (emergency core cooling functionability), events P and Q (opening and reclosing of the pressurizer's PORVs), and event H (emergency coolant recirculation operation) consider only the success of these events. Since the probability of an event sequence is the union of its component events probability, increasing the component event probability will increase the event sequence probability. By assuming the probability of events K, J, P, Q, and H successes to be equal to unity, the probability of the dominant accident sequences of Tables 22 and 23 is increased, while the complexity of the analysis needed to determine the probabilities of the dominant accident sequences is decreased.

Figures 3 through the 10 show the simplified event trees developed from the dominant accident sequences defined for reactor vessel rupture (Fig. 3), large LOCA (Fig. 4), Medium LOCA (Fig. 5), Small LOCA (Fig. 6), Small-Small LOCA with F&B capability (Fig. 7), Small-Small LOCA without F&B capability (Fig. 8), transients with F&B capability (Fig. 9) and transients without F&B capability (Fig. 10). Because of their similarity, the dominant accident sequences defined for the Class 1 Transient (T1) and the Class 2 Transients (T2) were combined into one common event tree for transients with F&B capability and another common event tree for transients without F&B capability. These simplified event trees were developed from full event trees taken from the SSMRP report [1] by "pruning" accident sequence "branches" that did not fulfill the definition of being dominant.

Table 19

Dominant Event Sequences (with Feed and Bleed) Ranking

Rank	Event Sequence Description			Radioactive Release Risk (manrem/yr)
	Initiating Event	Accident Sequence No.	Containment Failure Mode	
1	SLOCA	21	delta	1.82
2	T2	4	delta	7.52 E-1
3	RVR	7	gamma	7.20 E-1
4	SSLOCA	21	delta	6.61 E-1
5	T2	4	gamma	3.22 E-1
6	SSLOCA	35	gamma	2.37 E-1
7	MLOCA	21	delta	1.70 E-1
8	SSLOCA	35	delta	7.91 E-2
9	RVR	6	gamma	7.37 E-2
10	T2	4	alpha	7.06 E-2
11	RVR	7	alpha	5.34 E-2
12	LLOCA	27	gamma	3.80 E-2
13	SSLOCA	34	gamma	3.53 E-2
14	T2	4	epsilon	3.14 E-2
15	RVR	7	delta	2.37 E-2
16	SSLOCA	35	alpha	2.23 E-2
17	SLOCA	21	alpha	1.84 E-2
18	SLOCA	28	alpha	1.71 E-2
19	LLOCA	27	delta	1.27 E-2
20	LLOCA	13	alpha	1.22 E-2
21	SSLOCA	34	delta	1.10 E-2
22	LLOCA	27	epsilon	8.21 E-3
23	SSLOCA	35	epsilon	7.87 E-3
24	LLOCA	28	alpha	7.20 E-3
25	SSLOCA	34	epsilon	7.13 E-3
26	SSLOCA	21	alpha	6.67 E-3
27	RVR	6	alpha	5.47 E-3
28	MLOCA	27	gamma	5.22 E-3
29	LLOCA	13	epsilon	5.15 E-3
30	RVR	1	alpha	5.14 E-3
31	SSLOCA	D	gamma	4.58 E-3
32	T1	7	alpha	4.07 E-3
33	MLOCA	13	alpha	3.92 E-3
34	SSLOCA	34	alpha	3.09 E-3
35	RVR	1	epsilon	2.17 E-3
Total				5.26 manrem/yr

See Table 18 for initiating event definition.

See Figures 3-10 for accident sequence number definition.

alpha = containment failure by vessel steam explosion.

beta = containment failure by leakage.

gamma = containment failure by hydrogen explosion.

delta = containment failure by overpressurization.

epsilon = containment failure by core melt-through.

Table 20

Dominant Event Sequences (without Feed and Bleed) Ranking

Rank	Event Sequence Description	Radioactive Release Risk (manrem/yr)
Initiating Event	Accident Sequence No. Failure Mode	
1	SLOCA 21 delta	1.82
2	T2 4 delta	1.39
3	SSLOCA 21 delta	8.11 E-1
4	RVR 7 gamma	7.20 E-1
5	T2 4 gamma	5.96 E-1
6	T2 4 alpha	4.35 E-1
7	SSLOCA 35 gamma	2.30 E-1
8	T2 4 epsilon	1.86 E-1
9	MLOCA 21 delta	1.71 E-1
10	SSLOCA 34 gamma	8.28 E-2
11	SSLOCA 35 delta	7.67 E-2
12	RVR 6 gamma	7.37 E-2
13	RVR 7 alpha	5.34 E-2
14	LLOCA 27 gamma	3.80 E-2
15	SSLOCA 34 delta	2.76 E-2
16	SSLOCA 29 alpha	2.57 E-2
17	RVR 7 delta	2.37 E-2
18	SSLOCA 35 alpha	2.16 E-2
19	SLOCA 21 alpha	1.84 E-2
20	SSLOCA 34 epsilon	1.79 E-2
21	SLOCA 28 alpha	1.71 E-2
22	LLOCA 27 delta	1.27 E-2
23	LLOCA 13 alpha	1.22 E-2
24	SSLOCA 29 epsilon	1.08 E-2
25	LLOCA 27 epsilon	8.21 E-3
26	SSLOCA 21 alpha	8.19 E-3
27	SSLOCA 34 alpha	7.77 E-3
28	SSLOCA 35 epsilon	7.63 E-3
29	LLOCA 28 alpha	7.20 E-3
30	RVR 6 alpha	5.47 E-3
31	MLOCA 27 gamma	5.21 E-3
32	LLOCA 13 epsilon	5.15 E-3
33	RVR 1 alpha	5.14 E-3
34	TI 7 alpha	4.07 E-3
35	MLOCA 13 epsilon	3.92 E-3
36	RVR 1 epsilon	2.17 E-3

Total

6.94 manrem/yr

See Table 18 for initiating event definition.

See Figures 3-10 for accident sequence number definition.

alpha = containment failure by vessel steam explosion.

beta = containment failure by leakage.

gamma = containment failure by hydrogen explosion.

delta = containment failure by overpressurization.

epsilon = containment failure by core melt-through.

Table 21

Dominant Accident Sequences
Radioactive Release Risk

Initiating Event	Accident Sequence Number	Radioactive Release Risk with Feed and Bleed (manrem/yr.)	Radioactive Release Risk without Feed and Bleed (manrem/yr.)
RVR	1	7.31 E-3	7.31 E-3
RVR	6	8.16 E-2	8.16 E-2
RVR	7	7.97 E-1	7.97 E-1
LLOCA	13	1.74 E-2	1.74 E-2
LLOCA	27	6.24 E-2	6.24 E-2
LLOCA	28	1.02 E-2	1.02 E-2
MLOCA	13	5.58 E-3	5.58 E-3
MLOCA	21	1.72 E-1	1.73 E-1
MLOCA	27	8.58 E-3	8.56 E-3
SLOCA	21	1.84	1.84
SLOCA	28	2.43 E-2	2.43 E-2
SSLOCA	21	6.67 E-1	8.19 E-1
SSLOCA	29	4.48 E-4	3.65 E-2
SSLOCA	34	5.42 E-2	1.36 E-1
SSLOCA	35	3.47 E-1	3.36 E-1
SSLOCA	3	7.53 E-3	n/a
T1	7	5.79 E-3	5.80 E-3
T2	4	1.18	2.61
Total		5.28 manrem/yr	6.97 manrem/yr

n/a - not applicable

See Table 18 for initiating event definition.

See Figure 3 for RVR accident sequence number definition.

See Figure 4 for LLOCA accident sequence number definition.

See Figure 5 for MLOCA accident sequence number definition.

See Figure 6 for SLOCA accident sequence number definition.

See Figures 7 and 8 for SSLOCA accident sequence number definition.

See Figures 9 and 10 for T1 and T2 accident sequence number definition.

Table 22

Dominant Accident Sequences
with Feed and Bleed Capability

Accident Sequence Descrip.	Component Events												
	K RPS	M PCS	L AFWS	C CSIS & &SSR CFCS(I)	B Feed& Bleed	D ECI	J ECF	P SRV -O	Q SRV -R	E CFCS (R)	F RHRS	G CSRE	H ECR
RVR-ACC01				$\bar{C}$						$\bar{E}$			
RVR-ACC06				C							$\bar{F}$	G	
RVR-ACC07				C							F		
LLOCA-ACC13				$\bar{C}$		D				$\bar{E}$			
LLOCA-ACC27				C		D					$\bar{F}$	G	
LLOCA-ACC28				C		D					F		
MLOCA-ACC13	$\bar{K}$			$\bar{C}$		D				$\bar{E}$			
MLOCA-ACC21	$\bar{K}$			C		$\bar{D}$	$\bar{J}$				F		$\bar{H}$
MLOCA-ACC27	$\bar{K}$			C		D					$\bar{F}$	H	
SLOCA-ACC21	$\bar{K}$			C		$\bar{D}$	$\bar{J}$				F		$\bar{H}$
SLOCA-ACC28	$\bar{K}$			C		D					F		
SSLOCA-ACC21	$\bar{K}$		$\bar{L}$	C		$\bar{D}$	$\bar{J}$				F		$\bar{H}$
SSLOCA-ACC29	$\bar{K}$		L	$\bar{C}$	B					$\bar{E}$			
SSLOCA-ACC34	$\bar{K}$		L	C							$\bar{F}$	G	
SSLOCA-ACC35	$\bar{K}$		L	C							F		
SSLOCA-ACCD	$\bar{K}$		L	$\bar{C}$	$\bar{B}$					E	$\bar{F}$	G	
T1-ACC07	$\bar{K}$	M	L	*	B			$\bar{P}$	$\bar{Q}$	$\bar{C}^*$			
T2-ACC04	$\bar{K}$		L	*	B			$\bar{P}$	$\bar{Q}$	$\bar{C}^*$			

*Event C (CSIS and CFCS(I)) occurs after Event P (SRV-O) and Event Q (SRV-Q) for T1 and T2 transients.

Event K - operation of the reactor protection system (RPS).

Event M - operation of the power conversion system (PCS).

Table 22 continued

- Event L - operation of the auxiliary feedwater system (AFWS) and secondary steam relief system (SSR).
- Event C - operation of the containment spray injection system (CSIS) and containment fan cooler system (injection mode.) (CFCS(I)).
- Event B - feed and bleed operation (F&B).
- Event D - emergency coolant injection (ECI) operation. Same as emergency core cooling system (ECCS) operation. See Table 18 for specific reactor safety system that fulfills ECI operational requirements for each initiating event.
- Event J - emergency core functionability (ECF). Not a system. Included to account for effectiveness of ECI to cool the core.
- Event P - opening of pressurizer's PORVs.
- Event Q - reclosing of pressurizer's PORVs.
- Event E - operation of containment fan cooler system (recirculation mode) (CFCS(R)).
- Event F - operation of residual heat removal system (RHRS).
- Event G - operation of containment spray recirculation system (CSRS).
- Event H - emergency coolant recirculation (ECR). See Table 18 for specific reactor safety system that fulfills ECR operational requirements for each initiating event.

Table 23

Dominant Accident Sequences
without Feed and Bleed Capability

Accident Sequence Descrip.	Component Events											
	K RPS	M PCS	L AFWS & SSR	C CSIS & CFCS (I)	D ECI	J ECF	P SRV -O	Q SRV -R	E CFCS (R)	F RHRS	G CSRS	H ECR
RVR-ACC01				$\bar{C}$					$\bar{E}$			
RVR-ACC06				C						$\bar{F}$	G	
RVR-ACC07				C						F		
LLOCA-ACC13				$\bar{C}$	D				$\bar{E}$			
LLOCA-ACC27				C	D					$\bar{F}$	G	
LLOCA-ACC28				C	D					F		
MLOCA-ACC13	$\bar{K}$			$\bar{C}$	D				$\bar{E}$			
MLOCA-ACC21	$\bar{K}$			C	$\bar{D}$	$\bar{J}$				F		$\bar{H}$
MLOCA-ACC27	$\bar{K}$			C	D					$\bar{F}$	G	
SLOCA-ACC21	$\bar{K}$			C	$\bar{D}$	$\bar{J}$				F		$\bar{H}$
SLOCA-ACC28	$\bar{K}$			C	D					F		
SSLOCA-ACC21	$\bar{K}$		$\bar{L}$	C	$\bar{D}$	$\bar{J}$				F		$\bar{H}$
SSLOCA-ACC29	$\bar{K}$		L	$\bar{C}$					$\bar{E}$			
SSLOCA-ACC34	$\bar{K}$		L	C						$\bar{F}$	G	
SSLOCA-ACC35	$\bar{K}$		L	C						F		
T1-ACC07	$\bar{K}$	M	L				$\bar{P}$	$\bar{Q}$				
T2-ACC04	$\bar{K}$		L				$\bar{P}$	$\bar{Q}$				

Event K - operation of the reactor protection system (RPS).

Event M - operation of the power conversion system (PCS).

Event L - operation of the auxiliary feedwater system (AFWS) and secondary steam relief system (SSR).

Event C - operation of the containment spray injection system (CSIS) and containment fan cooler system (injection mode) (CFCS(I)).

Table 23 continued

- Event D - emergency coolant injection (ECI) operation. Same as emergency core cooling system (ECCS) operation. See Table 18 for specific reactor safety system that fulfills ECI operational requirements for each initiating event.
- Event J - emergency core functionability (ECF). Not a system. Included to account for effectiveness of ECI to cool the core.
- Event P - opening of pressurizer's PORVs.
- Event Q - reclosing of pressurizer's PORVs.
- Event E - operation of containment fan cooler system (recirculation mode) (CFCS(R)).
- Event F - operation of residual heat removal system (RHRS).
- Event G - operation of containment spray recirculation system (CSRS).
- Event H - emergency coolant recirculation (ECR). See Table 18 for specific reactor safety system that fulfills ECR operational requirements for each initiating event.

Figure 3
 Reactor Vessel Rupture Dominant Accident Sequences
 Simplified Event Tree

Figure 4

Large LOCA Dominant Accident Sequences

Simplified Event Tree

Figure 5
Medium LOCA Dominant Accident Sequences
Simplified Event Tree

Figure 6
Small LOCA Dominant Accident Sequences
Simplified Event Tree

Figure 7
Small-Small LOCA Dominant Accident Sequences
with Feed and Bleed Capability
Simplified Event Tree

Figure 8

Small-Small LOCA Dominant Accident Sequences
without Feed and Bleed Capability
Simplified Event Tree

T-K	M	L	B	P-Q	C		
Transient -RPS	PCS	AFWS & SSR	Feed & Bleed	S/RU-0 S/RU-R	CSIS & CFCs(1)	Seq. No.	Sequence
<div style="display: flex; align-items: center; justify-content: center;"> <div style="text-align: center; margin-right: 10px;"> Success ↑ ↓ Failure </div> <div style="border: 1px solid black; padding: 10px; text-align: center;"> <div style="display: flex; justify-content: space-around; width: 100%;"> (M)LBP-QC </div> <div style="height: 150px; border: 1px solid black; margin: 10px 0;"></div> <div style="display: flex; justify-content: space-around; width: 100%;"> MLBP-QC </div> </div> </div>						T2-4	T2KLB $\overline{P}$ QC
						T1-7	T1RMLB $\overline{P}$ QC

Figure 9
Transient Dominant Event Sequences
with Feed and Bleed Capability
Simplified Event Tree

T-K	M	L	P-Q		
Transient -RPS	PCS	AFWS & SSR	S/RV-0 S/RV-R	Seq. No.	Sequence
Success ↑ ↓ Failure	(M)	L	P-Q	T2-4	T2KLPQ
	M	L	P-Q	T1-7	T1KMLPQ

Figure 10
 Transient Dominant Event Sequences
 without Feed and Bleed Capability
 Simplified Event Tree

5) DOMINANT ACCIDENT SEQUENCES APPLICATION

To apply the dominant accident sequences developed in the previous section (Section 4), it is suggested that plant-specific systemic event trees be developed from the functional event tree developed in Section 3. A "reactor safety system by reactor safety system" comparison should be made of the plant under analysis with the Zion Unit 1 plant (analyzed in the SSMRP report [1]). Significant differences (in the judgement of the analyst) particularly in the timing and order of the reactor safety systems initiation should be incorporated within the plant-specific event trees. If the plant-specific systemic event trees do not differ from the SSMRP event trees, then the dominant accident sequences may be applied directly without modification. However, the radioactive release risk of the dominant accident sequences will not apply to the plant under analysis. To calculate the radioactive release risk, fault trees of the plant-specific reactor safety systems should be developed. If the reactor safety systems of the plant under analysis are similar to the Zion Unit 1 reactor safety systems, then a plant-specific fault tree of that particular safety system may not have to be developed. The analyst should be aware, however, that the Zion plant is unique in that many of its reactor safety systems have been designed with additional components and as such, has more system flexibility than most PWR plants. All but the earliest Westinghouse PWR plants were generally designed and built to the same safety philosophy and design standards as the Zion plants but may lack the redundancy and system flexibility of the Zion plants.

The dominant accident sequences should be particularly applicable without modification to the 4-loop 1100-MWe class Westinghouse PWR plants using series 51 SGs and with dry containments. These same dominant accident sequences should also be applicable without modification to the 2-, 3- and 4- loop Westinghouse PWRs that use other SG series and with dry containments. These dominant accident sequences may be applicable to Westinghouse PWR plants with ice containments if plant-specific systemic event tree has been modified to reflect the difference in containment design and construction. The application of these dominant accident sequences to Combustion Engineering PWR plants should be generally possible without modification. The applicability of these dominant accident sequences to Babcock and Wilcox PWR plants may not be possible without considerable modification to the plant-specific systemic event tree because of the much faster response of the B&W plant to plant transients and the differences in reactor safety systems.

6) CONCLUSIONS

This study developed from the SSMRP Report [1], a set of 18 dominant accident sequences with feed and bleed capability that considered 92% of the total radioactive release risk. Another set of 17 dominant accident sequences without feed and bleed capability which considered 95% of the total radioactive release risk was also developed.

These dominant accident sequence sets can be applied to other PWR plants, if the plant-specific systemic event trees developed from the functional event tree are similar to the systemic event trees of the SSMRP Report [1]. If they are similar, then the dominant accident sequences may be applied to the other PWR plants without modification. However, if the systemic event trees differ significantly (in the judgement of the analyst), the dominant accident sequence sets may be applied but with some modifications. The degree of modifications required will depend on the magnitude of the differences between the reactor safety systems of the plants being compared.

To estimate the radioactive release risk using these dominant accident sequence sets applied to another PWR plant, plant-specific fault trees should be developed in order to assign probabilities to the component events comprising the dominant accident sequences.

Westinghouse PWR plants, particularly in the 4-loop 1100-MWe class with dry containments have the most similarities to the Zion Unit 1 plant since they are of the same class. The dominant accident sequence sets developed can probably be applied to this class of PWR plants without modification.

7) REFERENCES

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