15.0 Accident and Analysis

General

In this chapter the effects of anticipated process disturbances and postulated component failures are examined to determine their consequences and to evaluate the capability built into the plant to control or accommodate such failures and events. The system response analysis is based upon the core loading shown in Figure 4.3-1 and is used to identify the limiting events for the ABWR. Other fuel designs and core loading patterns, including loading patterns similar to Figure 4.3-2, will not affect the sensitivities demonstrated by this study. Evaluation of these limiting events for each plant cycle will assure that the criteria in Appendix 4B are met.

A systematic approach to plant safety has been developed. The key to this approach to plant safety is the Nuclear Safety Operational Analysis (NSOA). Key inputs into the NSOA are derived from the applicable regulations and through industry codes and standards. The generic NSOA for ABWR is presented in Appendix 15A.

The entire spectrum of events in the NSOA has been evaluated to establish the most limiting or design basis events in a meaningful manner. It is the design basis events that are quantified in this chapter.

The scope of the situations analyzed includes anticipated (expected) operational occurrences (AOOs) (e.g., loss of electrical load), off-design abnormal (unexpected) transients that induce system operations condition disturbances, postulated accidents of low probability (e.g., the sudden loss of integrity of a major component), and finally, hypothetical events of extremely low probability (e.g., an anticipated transient without the operation of the entire Control Rod Drive System).

15.0.1 Nuclear Safety Operational Analysis

In Appendix 15A, Nuclear Safety Operational Analyses, all unacceptable safety results and all required safety actions are identified. In addition, an evaluation of the entire spectrum of events is consistently carried out for all plant designs to demonstrate that a consistent level of safety has been attained.

The NSOA acceptance criteria are based on event probability, which means that events more likely to occur are tested against more restrictive limits. This is consistent with industry practice and the applicable regulatory requirements.

The starting point for the NSOA is the establishment of unacceptable safety results. This concept enables the results of any safety analysis to be compared to applicable criteria. Unacceptable safety results represent an extension of the nuclear design criteria for plant systems and components which are used as the basis for system design. The unacceptable safety results have been selected so that they are consistent with applicable regulations and industry codes and standards.

The focal point of the NSOA is the event analysis, in which all essential protection sequences are evaluated until all required safety actions are successfully completed. The event analysis identifies all required front-line safety systems and their essential auxiliaries.

The full spectrum of initial conditions limited by the constraints placed on planned operation for AOOs, accidents, and plant capability demonstrations are evaluated. All events are analyzed until a stable condition is obtained. This assures that the event being evaluated does not have a characteristic for long-term consideration which is important.

In the event analysis all essential systems, operator actions, and limits to satisfy the required safety actions are identified. Limits are derived only for those parameters continuously available to the operator. Credit for operator action is taken only when an operator can be reasonably expected to perform the required action based on the information available to him.

In the NSOA, a complete and consistent set of safety actions (i.e., those required to prevent unacceptable results) has been developed. For transients and accidents, a single-failure-proof path to plant shutdown must be shown. The application of a single-failure criterion to these events is imposed as an additional measure of conservatism in the NSOA process.

15.0.2 Event Analytical Objective

The spectrum of postulated initiating events developed from the NSOA was divided into categories based upon the type of disturbance and the expected frequency of the initiating occurrence. The limiting events in each combination of category and frequency were evaluated using the core loading in Figure 4.3-1 to determine the limiting events. The plant safety analysis evaluates the ability of the plant to operate without unacceptable safety results within regulatory guidelines. This objective is met by satisfying the criteria in Appendix 4B.

15.0.3 Analytical Categories

Each event analyzed is assigned to one of eight categories listed in Chapter 15 of Regulatory Guide 1.70.

15.0.4 Event Evaluation

15.0.4.1 Identification of Causes and Frequency Classification

Situations and causes which lead to the initiating event analyzed in the sensitivity study are described within the categories designated in Subsection 15.0.3. The frequency of occurrence of each event is summarized based upon the NSOA and currently available operating plant history for the transient event. Events for which inconclusive data exist are discussed separately within each event section.

Each initiating event within the major groups is assigned to one of three frequency groups defined in Regulatory Guide 1.70.

15.0.4.2 Identified Results

Events analyzed for each plant must meet the criteria in Appendix 4B.

15.0.4.3 Sequence of Events and Systems Operations

Each transient or accident evaluated in the sensitivity study is discussed and evaluated in terms of:

(1) A step-by-step sequence of events from initiation to final stabilized condition

(2) The extent to which normally operating plant instrumentation controls are assumed to function

(3) The extent to which the plant and reactor protection systems are required to function

(4) The credit taken for the functioning of normally operating plant systems

(5) The operation of engineered safety systems that is required

This sequence of events is supported by the NSOA for the transient or accident. The effect of a single equipment failure or malfunction or an operator error on the event is shown in the NSOA.

15.0.4.4 Analysis Basis

The sensitivity study results given in this chapter are based upon the core loading given in Figure 4.3-1. These sensitivities are valid for other fuel designs and core loadings.

15.0.4.4.1 Evaluation Models

The computer codes used in the analysis of the transients and accidents in this chapter are shown in Table 15.0-1a. These models have been approved by the USNRC.

POLCA is a three-dimensional code for simulating the neutronic, thermal, and hydraulic behavior of a reactor core under steady-state conditions. The code solves the two-group neutron diffusion equation and couples it to thermal-hydraulic equations. Detailed pin-by-pin nuclear data are calculated by PHOENIX, which is a two-dimensional multi-group transport theory code, and then averaged to obtain lattice physics constants for use in POLCA. Further information on POLCA and PHOENIX can be found in reference 15.0-1.

BISON is a one-dimensional thermal-hydraulic model for the reactor cooling system, a one-dimensional (axial) neutron kinetics model with time-dependent 2 group theory and 6 groups of delayed neutron as well as a finite difference model for the fuel heat conduction and heat transfer to the coolant. The hydrodynamic model, which features a homologous model for coolant pump dynamics, can represent external and internal pumps, as well as a jet pump loop. The effect of steam separators is modeled. The code also includes a one-dimensional model for the steam line flow dynamics, and models for the turbine and feedwater system and the various

control systems. The safety and relief valve systems are represented as well as the hydraulic control rod insertion system. A slave channel model for the reactor core or individual coolant channels is used for calculating transient dryout margin in the hot channel. Further information on BISON can be found in references 15.0-2 through 15.0-7.

RAMONA is a systems transient code used for simulation of the dynamic behavior of a BWR. A 1-1/2 energy group, coarse mesh diffusion model in a three-dimensional rectangular coordinate system is used to predict transient three-dimensional fission power distributions in the core. Six delayed neutron groups are accounted for. Also decay heat from fission products is computed in RAMONA-3. Thermal energy storage and conduction in fuel elements (pellet, gas gap and fuel cladding), each one representing all the fuel in a computational cell of the three-dimensional mesh for neutron kinetics calculations is computed using spatial discretization in the radial direction in a finite difference form. The RAMONA-3 models allow two phase flow with unequal phase velocities described by a slip correlation and treat subcooled or superheated liquid phases. Four conservation equations treating vapor mass, mixture mass, and momentum, and energy conservation describe the coolant dynamics in the vessel. More information about the RAMONA code can be found in reference 15.0-8.

The POLCA-T code performs the 3D coupled thermal hydraulic and neutron kinetic calculation of an entire event simulated by the code. The simulated cases can be transients, stability, reactivity events etc. The reactor is divided into a user-specified number of volume cells and flow paths. The flow paths represent momentum control volumes between adjacent mass/energy control volume cells.

The POLCA-T code models can be divided into four main sections:

- The hydraulic model solves the mass, energy, and momentum conservation equations for each phase and for each control volume.
- The system models contain detailed models of the various reactor components.
- The thermal model calculates the heat conduction and heat transfer from the heat structures (fuel rods, pressure vessel, and internals (slabs)) to the coolant.
- The power generation models calculate the heat generation due to fission in the fuel, direct heat released in the coolant, and decay heat. A two group 3D kinetics model determines fission power.

More information about the POLCA-T code can be found in reference 15.0-9.

15.0.4.4.2 Input Parameters and Initial Conditions for Analyzed Events

In general, the events analyzed for the sensitivity analysis documented within this section have values for input parameters and initial conditions as specified in Table 15.0-1. Analyses which

assume data inputs different than these values are designated accordingly in the appropriate event discussion.

The normal maximum allowable reactor operating condition is the 100%-power/111%-flow condition. The maximum power measurement uncertainty is usually ~2%. Therefore, the sensitivity analyses are based on 102% power level.

The analytical values for some system characteristics, like SRV delay/stroke time, reactor internal pump coastdown time constant, etc., bound the design specification for that system. These values will be checked during startup tests.

All setpoints for the protection system assumed in the analyses are conservative, which includes instrument uncertainty, calibration error and instrument drift. The nominal and allowable values for these setpoints, (see Technical Specifications) assume that the setpoints will not exceed what are assumed in the analyses.

In conclusion, the input parameters and initial conditions (including uncertainties) used in the sensitivity study are conservative values and bound the operating band.

15.0.4.4.3 Initial Power/Flow Operating Constraints

The power/flow map used for the system response analysis is shown in Figure 15.0-1. The analyses basis for most of the sensitivity analyses is 102% thermal power at reduced core flow (90%). Rated core flow can be achieved with either nine or ten pumps in operation. This operating point is the apex of the operating power/flow map which, in response to any classified abnormal operational transients, will usually yield the minimum pressure and thermal margins. Referring to Figure 15.0-1, the apex of the bounded power/flow map is point A, the upper bound is the design flow control line (102% rod line A-D), the lower bound is the zero power line H'-J, the right bound is the maximum flow line A'-H', and the left bound is the natural circulation line D-J.

The power/flow map (A-D-J-H'-A') represents the operational region covered by abnormal operational transient evaluations.

Any other constraint which may truncate the bounded power/flow map (e.g., the moisture carryover protection region, the licensed power limit and other restrictions based on pressure and thermal margin criteria) must be observed. See Subsection 4.4.3.3 for power/flow map operating restrictions. The upper operating power/flow limit of a reactor is predicated on the operating basis of the analysis and the corresponding constant rod pattern line. This boundary may be truncated by the licensed power and the MCPR operating limit.

Certain localized events are evaluated at other than the above-mentioned conditions. These conditions are discussed pertinent to the appropriate event.

The power/flow operating map for a plant may differ from that used in the system response analysis given in this chapter. Differences in the map will not change the designation of limiting

events. The operating map used at a plant will be provided by the COL applicant to the USNRC for information (Subsection 4.4.7.1).

15.0.4.5 Evaluation of Results

The results of the system response analyses are presented in Table 15.0-2. Based on these results, the limiting events have been identified. Reasons why the other events are not limiting are given in the event documentation. The limiting events which establish CPR operating limit include:

(1) **Limiting Pressurization Events:** Inadvertent closure of one turbine control valve and generator load rejection with all bypass valve failure and turbine trip with all bypass valve failure*.

(2) **Limiting Decrease in Core Coolant Temperature Events:** Feedwater Controller Failure—Maximum Demand

For the core loading in Figure 4.3-1, the resulting equilibrium core MCPR operating limit is 1.37, based on a reference Safety Limit MCPR (SLMCPR) of 1.11. The operating limit based on the plant loading pattern and SLMCPR will be provided by the COL applicant to the USNRC for information (Subsection 15.0.5.2 for COL license information requirement).

Results of the transient analyses for individual plant reference core loading patterns will differ from the results shown in this chapter. However, the relative results between core associated events do not change. Therefore, only the results of the identified limiting events given in Table 15.0-4 will be provided by the COL applicant to the USNRC for information (Subsection 15.0.5.1).

15.0.4.5.1 Effect of Single Failures and Operator Errors

The effect of a single equipment failure or malfunction or operator error is provided in Appendix 15A.

15.0.4.5.2 Analysis Uncertainties

The analysis uncertainties meet the criteria in Appendix 4B.

15.0.4.5.3 Barrier Performance

The significant areas of interest for internal pressure damage are the high-pressure portions of the reactor coolant pressure boundary (i.e., the reactor vessel and the high pressure pipelines attached to the reactor vessel). The plant shall meet the criteria in Appendix 4B.

* If turbine control is partial-arc

15.0.4.5.4 Radiological Consequences

This chapter describes the consequences of radioactivity release for the core loading in Figure 4.3-1 during three types of events: (1) incidents of moderate frequency (anticipated operational occurrences); (2) infrequent incidents (abnormal operational occurrences); and (3) limiting faults (design basis accidents). For all events whose consequences are limiting, a detailed quantitative evaluation is presented. For nonlimiting events, a qualitative evaluation is presented or results are referenced from a more limiting or enveloping case or event.

A summary of applicable accidents is provided in Table 15.0-3, which compares the calculated amount of failed fuel to that used in worst-case radiological calculations for the core shown in Figure 4.3-1. Radiological calculations for a plant initial core will be provided by the utility to the USNRC for information (see Subsection 15.0.5 for COL license information requirements).

15.0.5 COL License Information

15.0.5.1 Anticipated Operational Occurrences (AOO)

The results of the events identified in Subsection 15.0.4.5 for plant core loading will be provided by the COL applicant referencing the ABWR design to the USNRC for information.

15.0.5.2 Operating Limits

The operating limit resulting from the analyses normally provided in this subsection will be provided by the COL applicant referencing the ABWR design to the USNRC for information.

15.0.5.3 Design Basis Accidents

Results of the design basis accidents, including radiological consequences, will be provided by the COL applicant referencing the ABWR design to the USNRC for information.

15.0.6 References

15.0-1 "The Advanced PHOENIX and POLCA Codes for Nuclear Design of Boiling Water Reactors", CENPD-390-P-A, December 2000.

15.0-2 "BISON – A One Dimensional Dynamic Analysis Code for Boiling Water Reactors", RPA 90-90-P-A, December 1991.

15.0-3 "BISON – One Dimensional Dynamic Analysis Code for Boiling Water Reactors: Supplement 1 to Code Description and Qualification", CENPD-292-P-A, July 1996.

15.0-4 "Supplement 2 to BISON Topical Report RPA 90-90-P-A", WCAP-16606-P-A, Revision 1, August 2009.

15.0-5 "Supplement 3 to BISON Topical Report RPA 90-90-P-A – SAFIR Control System Simulator", WCAP-17079, Revision 0, October 2009.

15.0-6 "Supplement 4 to BISON Topical Report RPA 90-90-P-A – Extended Qualification of BISON", WCAP-17202, June 2010.

15.0-7 "Supplement 5 to BISON Topical Report RPA 90-90-P-A – Fast Transient and ATWS Methodology", WCAP-17203, June 2010.

15.0-8 "ABB Atom Control Rod Drop Accident Analysis Methodology for Boiling Water Reactors. The RAMONA-3B Computer Code", RPA 89-112, November 1989.

15.0-9 "POLCA-T: System Analysis Code with Three-Dimensional Core Model", WCAP-16747-P, Revision 0, March 2007.

Table 15.0-1 Input Parameters and Initial Conditions for System Response Analysis Transients

1.	Thermal Power Level (MWt)	
	Warranted Value	3926
	Analysis Value	4005
2.	Steam Flow (kg/h)	
	Warranted Value	7.65×10^6
	Analysis Value	7.84×10^6
3.	Core Flow (kg/h)	
	Rated	52.2×10^6
	Maximum	58.0×10^6
	Analysis Value	47.0×10^6
4.	Feedwater Flow Rate (kg/s)	
	Warranted Value	2124
	Analysis Value	2179
5.	Feedwater Temperature (^oC)	217
6.	Vessel Dome Pressure (MPaG)	7.17
7.	Not Used	
8.	Turbine Bypass Capacity (% NBR)	33
9.	Core Coolant Inlet Enthalpy (kJ/g)	1.22
10.	Turbine Inlet Pressure (MPaG)	6.77
11.	Fuel Lattice	N
12.	Core Leakage Flow (%)	16.52
13.	Required MCPR Operating Limit	1.37
14.	MCPR Safety Limit	1.11
15.	Not Used	
16.	Not Used	
17.	Not Used	
18.	Not Used	
19.	Control Rod Drive Position versus time	Table 15.0-6
20.	Nuclear characteristics used in BISON simulations	EOEC*
21.	Number of Reactor Internal Pumps	10
22.	Safety/Relief Valve Capacity (%NBR) at 7.89 MPaG	89.7
	Quantity Installed	18
23.	Relief Function Delay (s)	0.55

Table 15.0-1 Input Parameters and Initial Conditions for System Response Analysis Transients (Continued)

24.	Relief Function Opening Time (s)	0.15
25.	Safety Function Delay (s)	0.0 †
26.	Safety Function Opening Time (s)	0.3
27.	Setpoints for Safety/Relief Valves Safety Function (MPaG) Relief Function (MPaG)	 8.12, 8.19, 8.26, 8.33, 8.39 7.89, 7.96, 8.03, 8.10, 8.17, 8.24
28.	Number of Valve Groupings Simulated Safety Function (No.) Relief Function (No.)	 5 6
29.	S/R Valve Reclosure Setpoint — Both Modes (% of setpoint) — Maximum Safety Limit (used in analysis) — Minimum Operational Limit	 98 93
30.	High Flux Trip (% NBR) Analysis Setpoint (125 x 1.02)	 127.5
31.	High Pressure Scram Setpoint (MPaG)	7.62
32.	Vessel level Trips (m above bottom of separator skirt bottom) Level 8—(L8) (m) Level 4—(L4) (m) Level 3—(L3) (m) Level 2—(L2) (m)	 1.73 1.08 0.57 –0.75
33.	APRM Simulated Thermal Power Trip Scram % NBR Analysis Setpoint (115 x 1.02) Time Constant (s)	 117.3 7
34.	Reactor Internal Pump Trip Delay (s) — RPV High Pressure — Low Water Level — TCV/TSV Closure	 0.3 0.2 0.16
35.	Recirculation Pump Inertia for Analysis (kg-m^2) — ATWS — Trip of RIPs for mitigation — Flow Decrease Event	 21.5 19.5 17.5
36.	Total Steamline Volume (m^3)	113.3
37.	Set pressure of Recirculation pump trip (MPaG)	7.76

* EOEC = End of Equilibrium Cycle

† This is a programming convenience number.

Figure 15.0-1a Computer Codes Used in the Analysis of Transients and Accidents

SubSection I.D.	Figure I.D.	Event	Analysis Code
15.1		**Decrease in Reactor Coolant Temperature**	
15.1.1	15.1-1	Loss of Feedwater Heating	POLCA
15.1.2	15.1-2	Runout of One Feedwater Pump	BISON
15.1.2	15.1-3	Feedwater Controller Failure-Maximum Demand	BISON
15.1.3	15.1-4	Opening of One Bypass Valve	BISON
15.1.3	15.1-5	Opening of all Control and Bypass Valves	BISON
15.2		**Increase in Reactor Pressure**	
15.2.1	15.2-1	Closure of One Turbine Control Valve	BISON
15.2.1	15.2-2	Pressure Regulator Downscale Failure	BISON
15.2.2	15.2-3	Generator Load Rejection, Bypass on	BISON
15.2.2	15.2-4	Generator Load Rejection, Failure of One Bypass Valve	BISON
15.2.2	15.2-5	Generator Load Rejection with Bypass Off	BISON
15.2.3	15.2-6	Turbine Trip, Bypass On	BISON
15.2.3	15.2-7	Turbine Trip, Failure of One Bypass Valve	BISON
15.2.3	15.2-8	Turbine Trip with Bypass Off	BISON
15.2.4	15.2-9	Inadvertent MSIV Closure	BISON
15.2.5	15.2-10	Loss of Condenser Vacuum	BISON
15.2.6	15.2-11	Loss of Aux. Power Transformer	BISON
15.2.6	15.2-12	Loss of Grid Connections	BISON
15.2.7	15.2-13	Loss of All Feedwater Flow	BISON
15.3		**Decrease in Reactor Coolant System Flow Rate**	
15.3.1	15.3-1	Trip of Three Reactor Internal Pumps	BISON
15.3.1	15.3-2	Trip of All Reactor Internal Pumps	BISON
15.3.2	15.3-3	Fast Runback of One Reactor Internal Pump	BISON
15.3.2	15.3-4	Fast Runback of All Reactor Internal Pumps	BISON
15.3.3	15.3-5	Seizure of One Reactor Internal Pump	BISON
15.4		**Reactivity and Power Distribution Anomalies**	
15.4.1	15.4-1	Rod Withdrawal Error–Low Power	RAMONA
15.4.5	15.4-2	Fast Runout of One Reactor Internal Pump	BISON

Figure 15.0-1a Computer Codes Used in the Analysis of Transients and Accidents (Continued)

SubSection I.D.	Figure I.D.	Event	Analysis Code
15.4.5	15.4-3	Fast Runout of All Reactor Internal Pumps	BISON
15.5		**Increase in Reactor Coolant Inventory**	
15.5.1	15.5-1	Inadvertent HPCF Startup	BISON
15.6		**Decrease in Reactor Coolant Inventory**	
15.6.5		Steam System Pipe Break Outside Containment	See Section 6.3
15.6.5		LOCA Within RCPB	See Section 6.3
15.6.6		Feedwater Line Break	See Section 6.3
15.8		**Anticipated Transients Without Scram**	See Section 15E
15E.6.1		Main Steam Isolation Valve Closure	BISON
15E.6.2		Loss of AC Power	BISON
15E.6.3		Loss of Feedwater	BISON
15E.6.4		Loss of Feedwater Heating	BISON
15E.6.5		Turbine Trip with Bypass	BISON
15E.6.6		Loss of Condenser Vacuum	BISON
15E.6.7		Feedwater Controller Failure	BISON
15.9		**Stability**	POLCA-T

Table 15.0-2 Results Summary of System Response Analysis Transient Events

Sub Section I.D.	Figure I.D.	Description	Max. Neutron Flux % NBR	Max. Dome Pressure (MPaG)	Max.Vessel Bottom Pressure (MPaG)	Max. Steamline Pressure (MPaG)	Max. Core Average Surface Heat Flux (% of Initial)	Δ in CPR	Freq. Category*
15.1		Decrease in core coolant temperature							
15.1.1		Loss of Feedwater Heating	113	7.17	7.41	7.10	112.8	0.09	†§
15.1.2	15.1-2	Runout of One Feedwater Pump	105	7.18	7.42	7.11	102	0.04	†
15.1.2	15.1-3	Feedwater Controller Failure—Maximum Demand	122	8.05	8.18	8.04	105	0.24	†‡
15.1.3	15.1-4	Opening of One Bypass Valve	102	7.17	7.41	7.10	100.0	0.03	†
15.1.3	15.1-5	Opening of All Control and Bypass Valves	102	7.91	8.00	7.94	100.0	0.03	†‡
15.1.4		Inadvertent Opening of One SRV				SEE	TEXT		†§
15.1.6		Inadvertent RHR Shutdown Cooling				SEE	TEXT		†‡
15.2		Increase in Reactor Pressure							
15.2.1	15.2-1A	Fast Closure of One Turbine Control Valve	115	7.41	7.65	7.34	105	0.10	†
15.2.1	15.2-1B	Slow Closure of One Turbine Control Valve	106	7.35	7.59	7.28	103	0.06	†
15.2.1	15.2-2	Pressure Regulator Downscale Failure	139	8.46	8.62	8.28	110	**	‡
15.2.2	15.2-3	Generator Load Rejection, Bypass On	121	8.04	8.17	8.04	101	0.22	†
15.2.2	15.2-4	Generator Load Rejection, Failure of One Bypass Valve	121	8.18	8.31	8.16	101	0.23	†§
15.2.2	15.2-5	Generator Load Rejection, Failure of All Bypass Valves	123	8.37	8.51	8.47	102	0.26	†‡
15.2.3	15.2-6	Turbine Trip, Bypass On	115	8.03	8.16	8.02	100.0	0.21	†
15.2.3	15.2-7	Turbine Trip w/Failure of One Bypass Valve	116	8.17	8.30	8.15	100	0.22	†§

Table 15.0-2 Results Summary of System Response Analysis Transient Events (Continued)

Sub Section I.D.	Figure I.D.	Description	Max. Neutron Flux % NBR	Max. Dome Pressure (MPaG)	Max.Vessel Bottom Pressure (MPaG)	Max. Steamline Pressure (MPaG)	Max. Core Average Surface Heat Flux (% of Initial)	Δ in CPR	Freq. Category*
15.2.3	15.2-8	Turbine Trip with Failure of All Bypass Valves	120	8.37	8.51	8.33	101	0.25	†‡
15.2.4	15.2-9	Inadvertent MSIV Closure	102	8.33	8.49	8.33	100	0.03	†
15.2.5	15.2-10	Loss of Condenser Vacuum	115	8.03	8.16	8.02	100	0.21	†
15.2.6	15.2-11	Loss of AC Power	103	8.00	8.12	8.05	100	0.18	†
15.2.7	15.2-12	Loss of All Feedwater Flow	103	7.17	7.41	7.10	100	0.07	†
15.2.8		Feedwater Piping Break				SEE	TEXT		
15.2.9		Failure of RHR Shutdown Cooling				SEE	TEXT		
15.3		Decrease in Reactor Coolant System Flow Rate							
15.3.1	15.3-1	Trip of Three Reactor Internal Pumps	102	7.17	7.42	7.11	100	0.11	†
15.3.1	15.3-2	Trip of All Reactor Internal Pumps	102	7.18	7.48	7.11	100	***	***
15.3.2	15.3-3	Fast Runback of One Reactor Internal Pump	104	7.17	7.42	7.10	100	0.03	†
15.3.2	15.3-4	Fast Runback of All Reactor Internal Pumps	102	7.17	7.41	7.10	100	0.03	†‡
15.3.3	15.3-5	Seizure of One Reactor Internal Pump	105	7.18	7.42	7.11	101	0.06	†‡
15.3.4		One Pump Shaft Break				SEE	TEXT		‡
15.4		Reactivity and Power Distribution Anomalies							
15.4.1.1		RWE-Refueling				SEE	TEXT		
15.4.1.2	15.4-1	RWE-Startup				SEE	TEXT		§
15.4.2		RWE at Power				SEE	TEXT		
15.4.3		Control Rod Misoperation				SEE	TEXT		

Table 15.0-2 Results Summary of System Response Analysis Transient Events (Continued)

Sub Section I.D.	Figure I.D.	Description	Max. Neutron Flux % NBR	Max. Dome Pressure (MPaG)	Max.Vessel Bottom Pressure (MPaG)	Max. Steamline Pressure (MPaG)	Max. Core Average Surface Heat Flux (% of Initial)	Δ in CPR	Freq. Category*
15.4.4		Abnormal Startup of One Reactor Internal Pump				SEE	TEXT		
15.4.5	15.4-2	Fast Runout of One Reactor Internal Pump	76	6.95	7.07	6.93	107	0.06^{+}	†
15.4.5	15.4-3	Fast Runout of All Reactor Internal Pumps	130	7.17	7.40	7.17	171	0.23^{+}	†‡
15.4.7		Mislocated Bundle Accident				SEE	TEXT		§
15.4.8		Misoriented Fuel Bundle Accident				SEE	TEXT		§
15.4.9		Rod Ejection Accident				SEE	TEXT		
15.4.10		Control Rod Drop Accident				SEE	TEXT		
15.5		Increase in Reactor Coolant Inventory							
15.5.1	15.5-1	Inadvertent HPCF Startup	102	7.17	7.41	7.10	100	0.03	†

* Frequency definition is discussed in Subsection 15.0.4.1.
† Moderate Frequency Incident.
†§ This event should be classified as an Infrequent Incident. However, criteria for moderate frequency events are conservatively applied.
§ Infrequent Incident .
†‡ This event should be classified as a Limiting Fault. However, criteria for moderate frequent events are conservatively applied.
‡ Limiting Fault.
** CPR criterion does not apply – see subsection 15.2.1. 10% of 10CFR100 met.
*** CPR criterion does not apply – see Subsection 15.3.1. PCT 487.2ºC.
\+ Transients initiated from low power.

Table 15.0-3 Summary of Accidents

		Failed Fuel Rods	
Subsection I.D.	Title	Calculated Value	NRC Worst-Case Assumption
15.2.1	Pressure Regulator Downscale Failure	None	<0.2%
15.3.1	Trip of All Reactor Internal Pumps	None	None
15.3.3	Seizure of one Reactor Internal Pump	None	None
15.3.4	Reactor Internal Pump Shaft Break	None	None
15.6.2	Instrument Line Break	None	None
15.6.4	Steam System Pipe Break Outside Containment	None	None
15.6.5	LOCA Within RCPB	None	100%
15.6.6	Feedwater Line Break	None	None
15.7.1.1	Main Condenser Gas Treatment System Failure	N/A	N/A
15.7.3	Liquid Radwaste Tank Failure	N/A	N/A
15.7.4	Fuel-Handling Accident	<145	178
15.7.5	Cask Drop Accident	None	All Rods in Cask

Table 15.0-4
Core-Wide Transient Analysis Results To Be Provided for Different Core Design

Transient	Max. Neutron Flux (%NBR)	Max. Core Average Surface Heat Flux (%NBR)	Δ CPR	Figure
Closure of One Turbine Control Valve	X	X	X	X
Load Rejection with all Bypass Valves Failure	X	X	X	X
Turbine Trip with all Bypass Valves Failure*	X	X	X	X
Feedwater Controller Failure—Maximum Demand	X	X	X	X
Fast Runout of all RIPs	X	X	X	X

* If turbine control is partial-arc

Table 15.0-5 Not Used

Table 15.0-6 ABWR FMCRD Scram Time

Rod Insertion (%)	Scram Time (seconds) (Including Solenoid De-energization) Used in Analysis
10	0.46
40	1.208
60	1.727
100	3.719

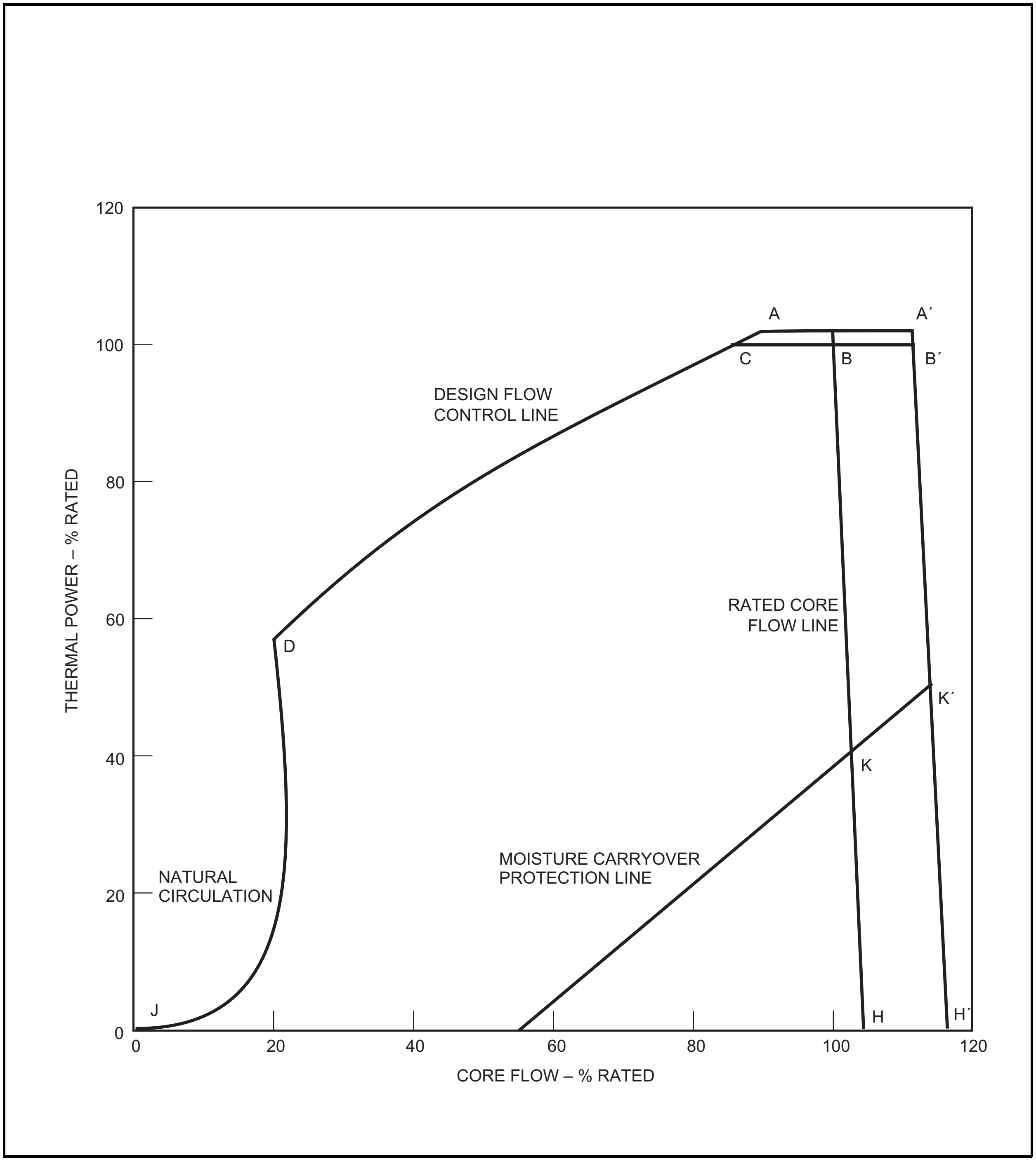

Figure 15.0-1 System Response Analysis Power/Flow Map