GE Nuclear Energy

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February 16, 1993

Docket No. STN 52-001

Chet Posiusny, Senior Project Manager
Standardization Project Directorate
Associate Directorate for Advanced Reactors
and License Renewal
Office of the Nuclear Reactor Regulation

Subject: **Submittal Supporting Accelerated ABWR Review Schedule - Reactor Systems and Mechanical Engineering Branches Outstanding Items**

Dear Chet:

Enclosed are SSAR markups addressing: Open Items 5.3.2-1 and 15.3-1; and Confirmatory Items 4.2-3 and 4.4-1.

It should be noted that in addressing Open Item 15.3-1, it was necessary to markup proprietary Page 4B-7. This markup is an addition to an earlier submittal and the proprietary affidavit under which it was originally issued is applicable.

Please provide copies of this transmittal to George Thomas and Dave Terao.

Sincerely,

Jack Fox

Jack Fox
Advanced Reactor Programs

cc: Hal Careway (GE)
Norman Fletcher (DOE)
Bob Huang (GE)
Caroline Smith (GE)

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4.2 FUEL SYSTEM DESIGN

4.2.1 Discussion

The fuel to be loaded in an ABWR is any fuel design approved by the USNRC or that meets the criteria documented in Appendix 4B. Using these designs will assure that all fuel system design requirements are met.

To demonstrate ABWR system response in this SSAR, a reference core of BP8x8R fuel is used. This core is shown in Section 4.3; information for this fuel design is provided in Reference 1. Each utility referencing the ABWR design may have different fuel and core designs which will be provided by the COL applicant to the USNRC for information. See Subsection 4.2.2.1 for COL license information.

The control rods perform the dual function of power shaping and reactivity control. A discussion of the rod control system components is presented in Section 4.6.

The control rod design to be used in an ABWR is any design approved or that meets the criteria documented in Appendix 4C. To demonstrate the ABWR system response in this report, a control rod design of sheathed cruciform array of stainless steel tubes filled with boron-carbide was used. This design is documented in Reference 2 and shown in Figure 4.2-1. The control blade design to be used at the plant will be provided by the COL applicant to the USNRC for information. See Subsection 4.2.2.2. for COL license information.

INSERT A

4.2.2 COL License Information

4.2.2.1 Fuel Design

The fuel bundle name and a reference to documentation of the fuel design will be provided by the utility referencing the ABWR design to the USNRC for information. (See Subsection 4.2.1).

4.2.2.2 Control Blade Design

The control blade model and reference to documentation of the control blade design will be provided by the utility referencing the ABWR design to the USNRC for information. (See Subsection 4.2.1).

4.2.3 References

1. *GE Fuel Bundle Designs,* NEDE-31152P.

2. *GE Control Rod Designs,* (To be issued).

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The reference ABWR control rod design consists of a sheathed cruciform array of stainless steel tubes filled with boron carbide (B_4C) powder. Figure 4.2-1 is an illustration of the reference design. The main structural members of the reference design are made of stainless steel and consist of a top handle, a lower transition piece with a control rod drive coupling, a vertical cruciform center post, and four U-shaped absorber tube sheaths. The top handle, lower transition piece and center post are welded into a single skeletal structure. The U-shaped sheaths are welded to the center post, handle, and lower transition piece to form the housing for the absorber rods filled with B_4C. Rollers at the top and bottom of the control rod guide the control rod as it is inserted and withdrawn from the core. The B_4C powder in the absorber tubes is compacted to approximately 70 percent of its theoretical density. The B_4C is sealed into the absorber tubes by plugs welded into each end, and is longitudinally separated into individual compartments by stainless steel balls. Typical parameters of the reference ABWR control rod design are provided in Table 4.2-1.

CI 4.2-3

TABLE 4.2-1

Typical Parameters - Reference ABWR Control Rod Design

Control Rod Weight (lb)	180
Absorber Rod - B_4C	
Number per control rod	72
Length (in.)	143
Diameter (in.)	.166
Density (% Theoretical)	70
Abssorber Tube - B_4C	
Cladding material	304SS
Outside diameter (in.)	.220
Wall thickness (in.)	.027
Sheath Thickness (in.)	.045
Pin Material	PH13-8MØ
Roller Material	INCONEL x-750

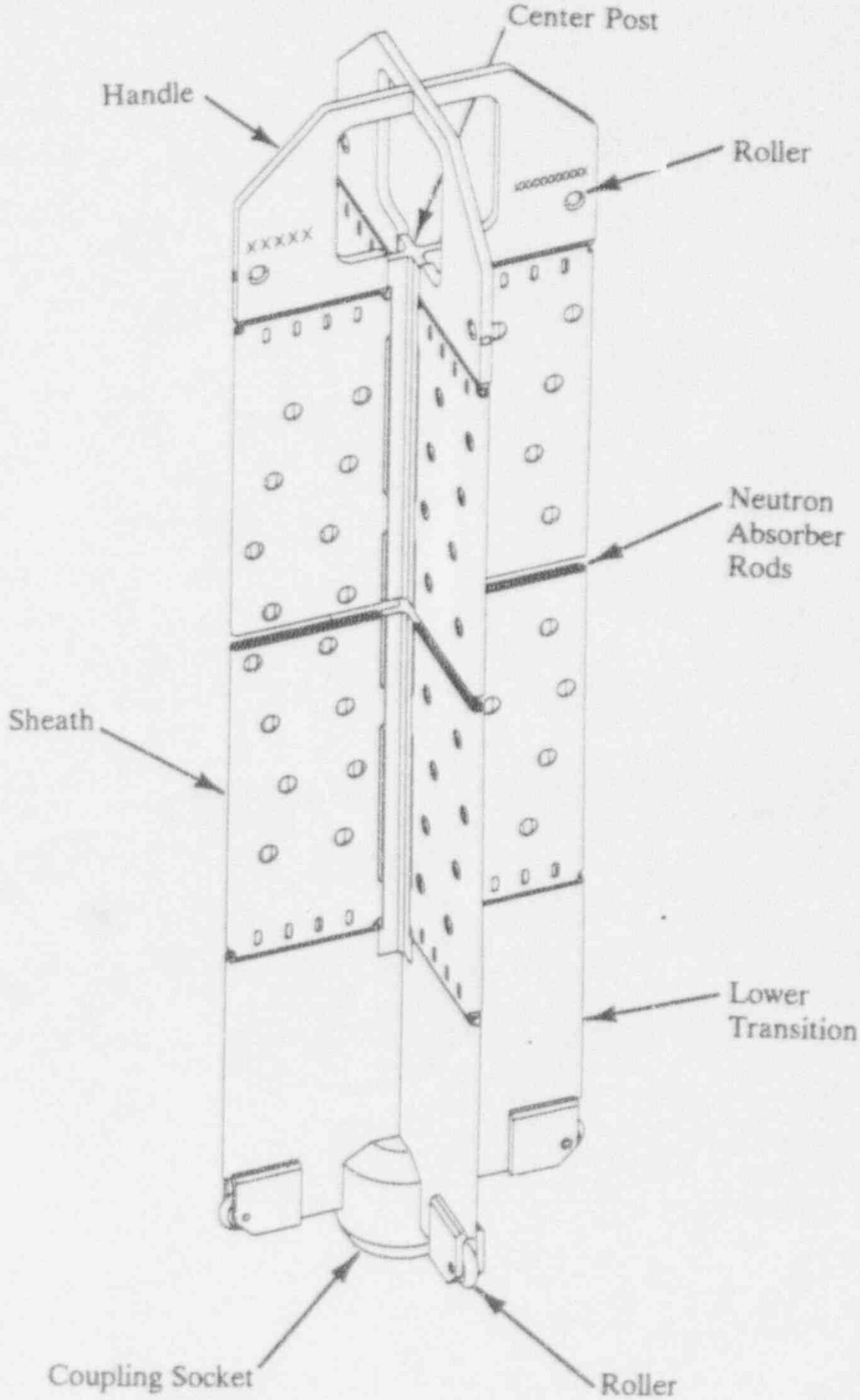

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FIGURE 4.2-1

Figure 4.2-1 Reference ABWR Control Rod Assembly

CI 4.4-1

The power-flow map for 10 RIP operation is shown in Figure 4.4-1, and for 9 RIP operation in Figure 4.4-2.

4.4.2 Description of the Thermal and Hydraulic Design of the Reactor Coolant System

4.4.2.1 Plant Configuration Data

4.4.2.1.1 Reactor Coolant System Configuration

The reactor coolant system is described in Section 5.4.

4.4.2.1.2 Reactor Coolant System Thermal Hydraulic Data

The steady-state distribution of temperature, pressure and flow rate for each flow path in the reactor coolant system is shown in Figure 5.1-1.

4.4.2.1.3 Reactor Coolant System Geometric Data

Volumes of regions and components within the reactor vessel are shown in Figure 5.1-2.

Table 4.4-5 provides the flow path length, height, liquid level, minimum elevations, and minimum flow areas for each major flow path volume within the reactor vessel and recirculation loops of the reactor coolant systems.

4.4.2.2 Operating Restrictions on Pumps

Expected recirculation pump performance curves are shown in Figure 5.4-3. These curves are valid for all conditions with a normal operation range varying from approximately 20 percent to 115 percent of rated pump flow.

Subsection 4.4.2.3 gives the operating limits imposed on the recirculation pumps by cavitation, pump load, bearing design flow starvation, pump speed, and steam separator performance.

4.4.2.3 Power-Flow Operating Map

4.4.2.3.1 Limits for Normal Operation

A BWR must operate with certain restrictions because of pump net positive suction head (NPSH), overall plant control characteristics, core thermal power limits, etc. ~~The~~ Those power-flow maps for the power range of operation shown ~~in Figure 4.4-1 was~~ were used in the system response analyses documented in Section 6.3 and Chapter 15. The specific power-flow operating map for each plant will be provided to the USNRC for information. See Subsections 4.4.4.1 and 4.4.4.2 for COL license information. The nuclear system equipment, nuclear instrumentation, and the reactor protection system, in conjunction with operating procedures, maintain operations within the area of the operating map for normal operating conditions. The boundaries on this map are as follows:

Natural Circulation Line, 0: The operating state of the reactor moves along this line for the normal control rod withdrawal sequence in the absence of recirculation pump operation.

102% Power Rod Line or Rated Power (Whichever Is Less): The 102% power rod line passes through 102% power at 100% flow. Some plants may establish an operating map in which 102% power can be achieved at lower flow. The operating state for the reactor follows this rod line (or similar ones) during recirculation flow changes with a fixed control rod pattern; however, rated power may not be exceeded.

Steam Separator Limit Line: This line results from the requirements to have acceptable moisture carryover fraction from steam separator.

4.4.2.3.2 Other Performance Characteristics

Other performance characteristics shown on the power-flow operating map are:

Constant Rod Lines A, B, C, D, E, F: These lines show the change in flow associated with power change while maintaining constant control rod position.

Constant Pump Speed Lines 1, 2, 3, 4, 5, 6, 7, 8: These lines show the change in flow associated with power changes while maintaining RIP speeds at a constant speed.

¶ It is required that at least 9 out of 10 RIPs are operating for normal operation. For operation with less than 9 RIPs in operation, the COL applicant will provide the necessary supporting analyses.

CI 4.4-1

4.4.2.3.3 Regions of the Power Flow Map

Region I This region defines the system operational capability with the reactor internal pumps running at their minimum speed (30%). Power changes, during normal startup and shutdown, will be in this region. The normal operating procedure is to start up along curve 1.

Region II This is the low power area of the operating map where the carryover through steam separators is expected to exceed the acceptable value. Operation within this region is precluded by system interlocks.

Region III This is the high power/low flow area of the operating map which the system is the least damped. Operation within this region is precluded by SCRRI (Selected Control Rods Run-In).

Region IV This represents the normal operating zone of the map where power changes can be made, by either control rod movement or by core flow changes, through the change of the pump speeds.

4.4.2.3.4 Design Features for Power-Flow Control

The following limits and design features are employed to maintain power-flow conditions shown in Figure 4.4-1:

(1) Minimum Power Limits at Intermediate and High Core Flows: To prevent unacceptable separator performance, the recirculation system is provided with an interlock to reduce the RIP speed.

(2) Pump Minimum Speed Limit: The Reactor Internal Pumps (RIPs) are equipped with Anti-Rotation Devices (ARD) which prevent a tripped RIP from rotating backwards. The ARD begins operating at 300 rpm decreasing speed. In order to prevent mechanical wear in the ARD, minimum speed is specified at 300 rpm. However, to provide a stable operation, the minimum pump speed is set at 450 rpm (30% of required).

4.4.2.3.5 Flow Control

The normal plant startup procedure requires the startup of all RIPs first and maintain at their minimum pump speed (30% of rated), at which point reactor heatup and pressurization can commence. When operating pressure has been established, reactor power can be increased. This power-flow increase will follow a line within Region I of the flow control map shown in Figure 4.4-1. The system is then brought to the desired power-flow level within the normal operating area of the map (Region IV) by increasing the RIP speeds and by withdrawing control rods.

Control rod withdrawal with constant pump speed will result in power/flow changes along lines of constant pump speed (Curves 1 through 8). Change of pump speeds with constant control rod position will result in power/flow changes along, or nearly parallel to, the rated flow control line (curves A through F).

4.4.2.4 Thermal and Hydraulic Characteristics Summary Table

The thermal-hydraulic characteristics are provided in Table 4.4-1 for the core and tables of Section 5.4 for other portions of the reactor coolant system.

← INSERT A (NEXT 2 PAGES)

CI 4.4-1

ATTACHMENT A

4.4.2.5 Thermal Hydraulic Stability Performance

As discussed in the Response to Question 100.1, the ABWR design assures the stability performance in the normal operating region is more stable than current operating BWRs by incorporating the following design features:

(1) Smaller inlet orifices, which increase the inlet single-phase pressure drop, and, consequently, improve the core and channel stability.

(2) Wider control rod pitch, which increases flow area, and, consequently, reduces the void reactivity coefficient and improves both core and channel stability, and

(3) More steam separators, which reduce the two-phase pressure drop, and improve the stability.

In order to reconfirm this conclusion, a stability analysis based on the procedures developed by the BWROG committee on thermal hydraulic stability (Reference 1) was performed for the ABWR. In this analysis, a conservative nuclear conditions, taking into consideration of future core design, were assumed. The results at the most limiting conditions in the normal operating region (i.e.; the intercept of 102 % rod line with all operating RIPs at their minimum speeds, assuming only 9 out of 10 RIPs are in operation) are as follows:

– Core Decay Ratio 0.72,

– Channel Decay ratio 0.36.

These results are also shown in Figure ~~1~~ 4.4-3 together with the criteria. From Figure 1, it is confirmed that that ABWR is stable in the normal operating region.

Furthermore, automatic logics (Figure ~~2~~ 4.4-4) which prevent plant operation in the region with the least stability margin is also implemented. This design is similar to Option I-A, one of long-term solutions considered by the BWROG. In addition, in order to meet the stability design requirements specified in the ALWR Utility Requirements Document, Option III, LPRM based Oscillation Power Range Monitor (OPRM), which is also one of long-term solutions

considered by the BWROG, will be implemented in the ABWR design, when the OPRM design is approved by the NRC.

As for issues relates to ATWS stability, they are of no concerns to the ABWR design, since the ABWR design has logic to automatically initiate the SLCS, including automatic initiation of feedwater run back. Furthermore, the ABWR EPG will incorporate any changes recommended by the BWROG.

In summary, the ABWR stability design is consistent with the licensing methodology proposed by the BWROG committee on thermal hydraulic stability. The ABWR will be stable in the normal operating region.

CI 4.4-1

conditioning equipment, alarming monitor, signal analysis and data acquisition equipment, and calibration equipment.

4.4.4 COL License Information

4.4.4.1 Power Flow Operating Map

The specific power flow operating map to be used at the plant will be provided by the COL applicant to the USNRC for information. (See Subsection 4.4.2.3.1).

4.4.4.2 Thermal Limits

The thermal limits for the core loading at the plant will be provided by the COL applicant to the USNRC for information. (See Subsection 4.4.2.3.1).

4.4.5 References

1. NEDO-31960, "BWR Owners' Group Long-Term Stability Solutions Licensing Methodology", June 1991.

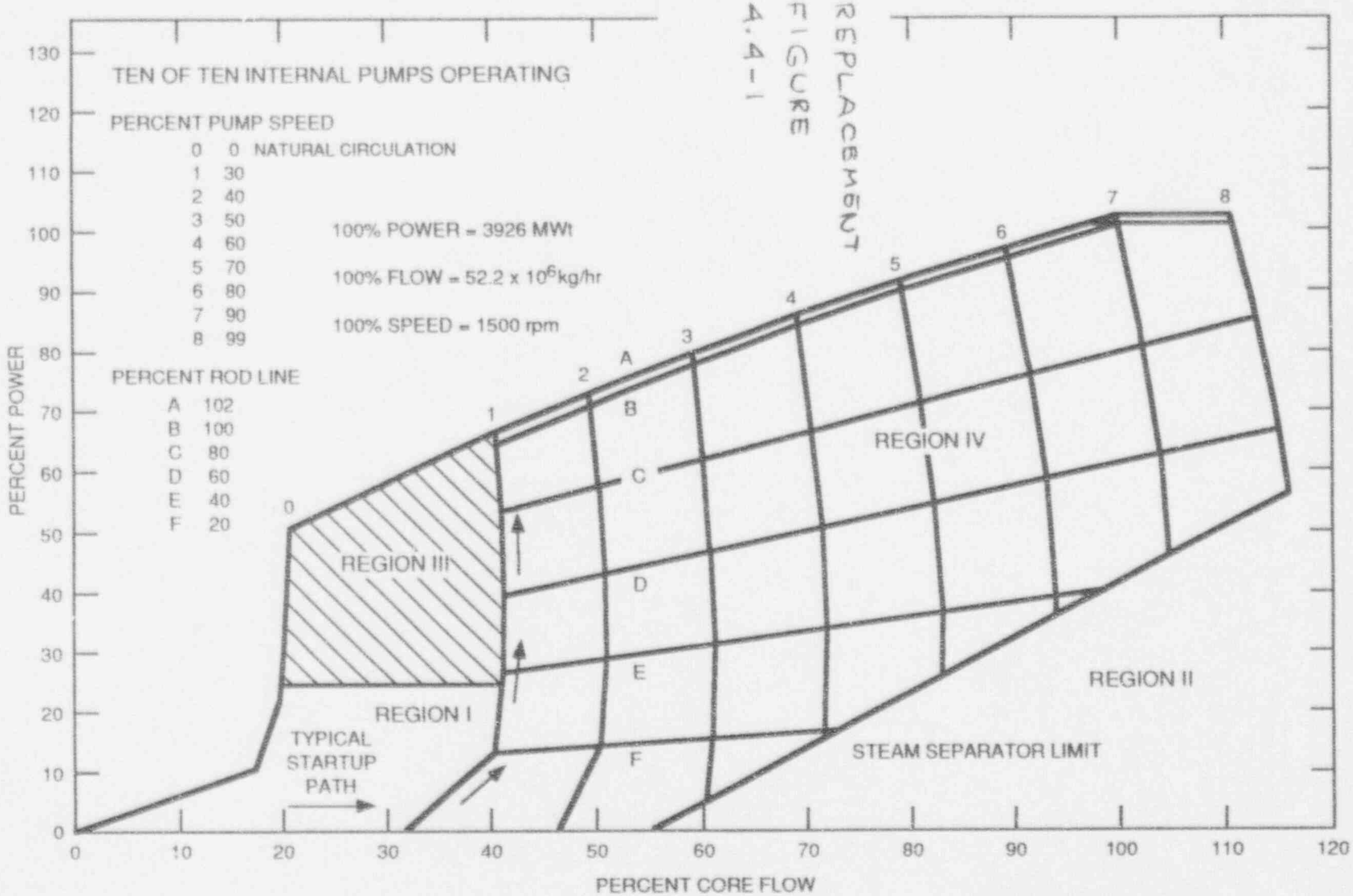

Figure 4.4-1 POWER-FLOW OPERATING MAP USED FOR SYSTEM RESPONSE STUDY (10 RIPS OPERATION)

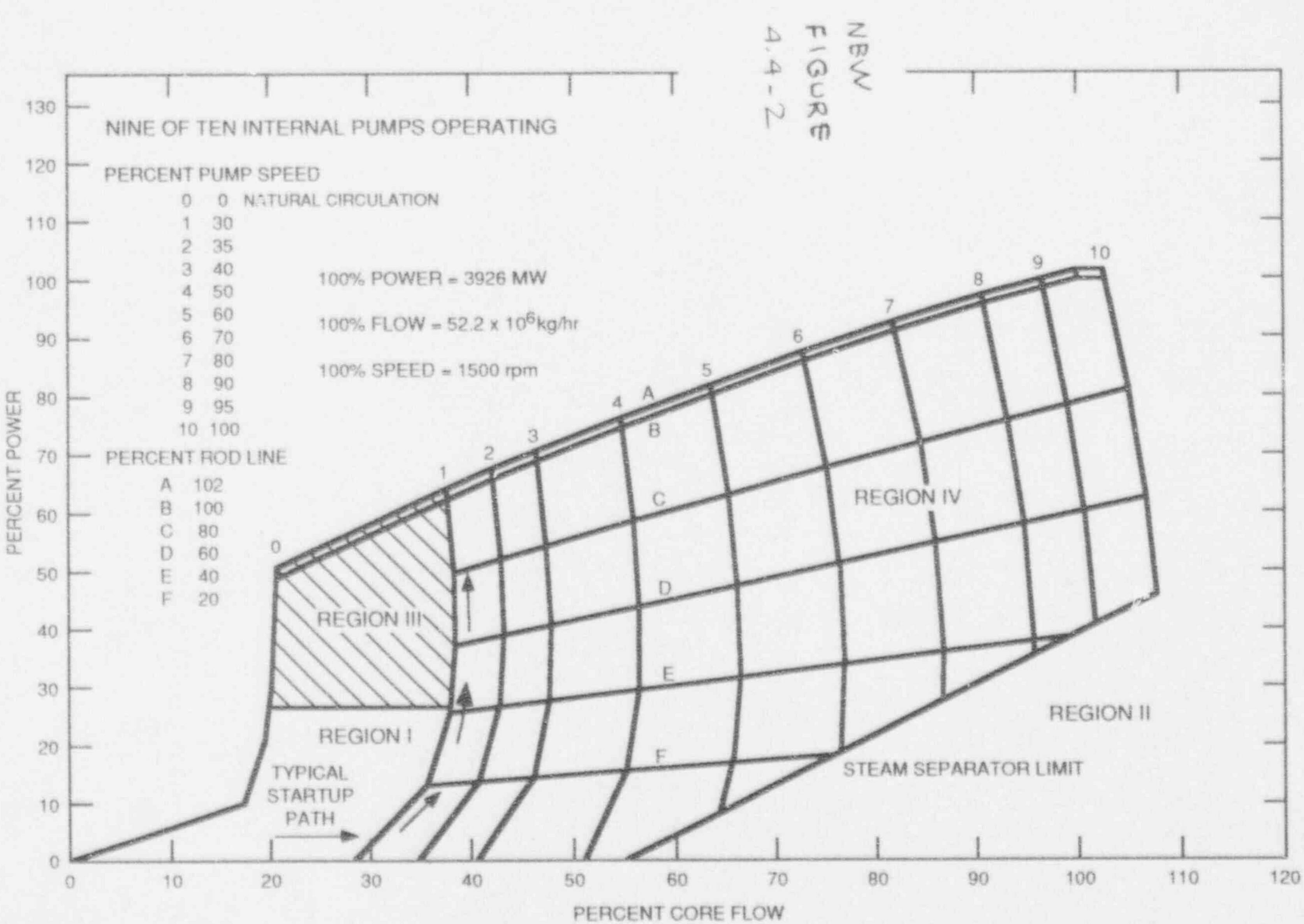

Figure 4.4-2 POWER-FLOW OPERATING MAP USED FOR SYSTEM RESPONSE STUDY (9 RIPS OPERATION)

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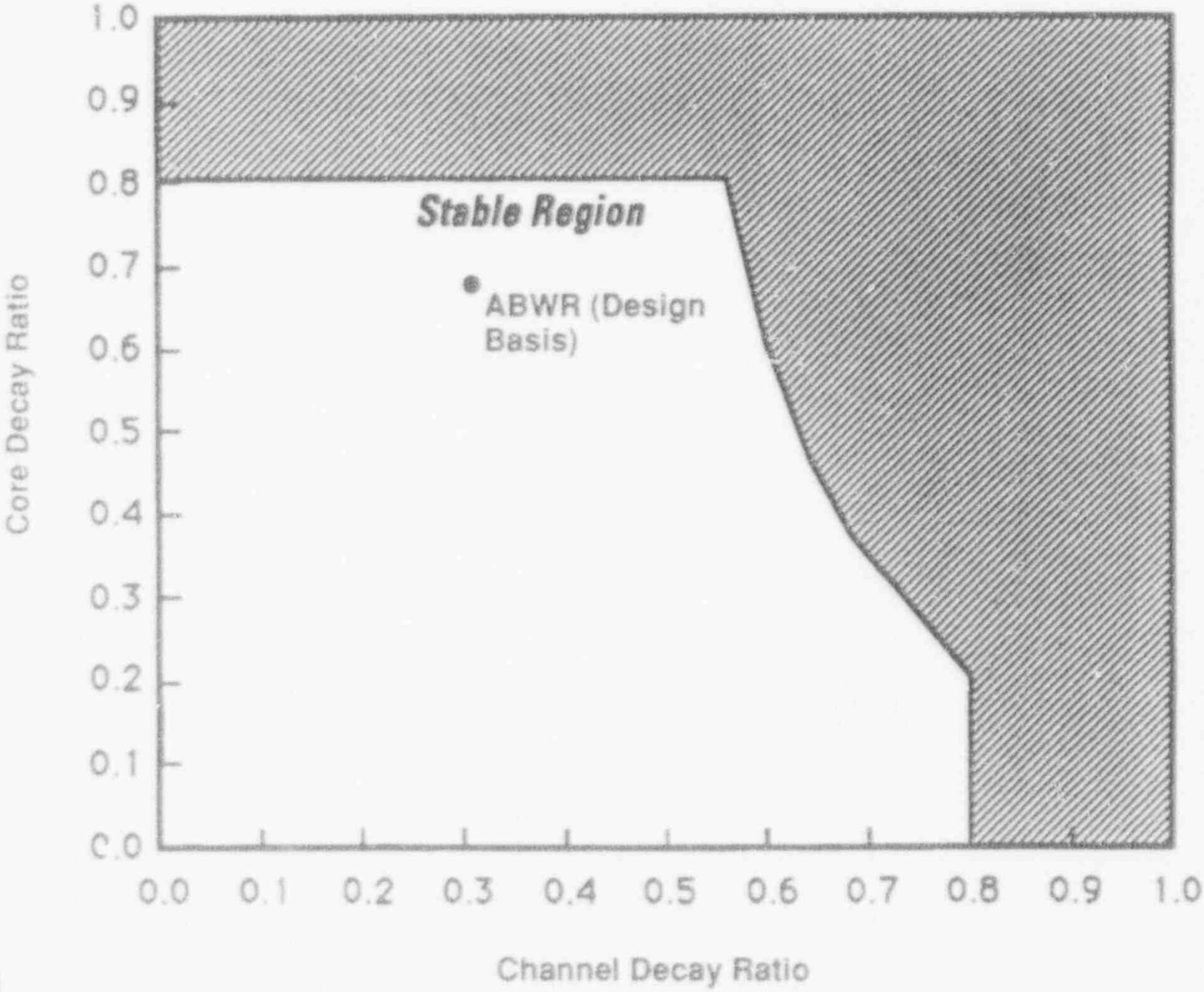
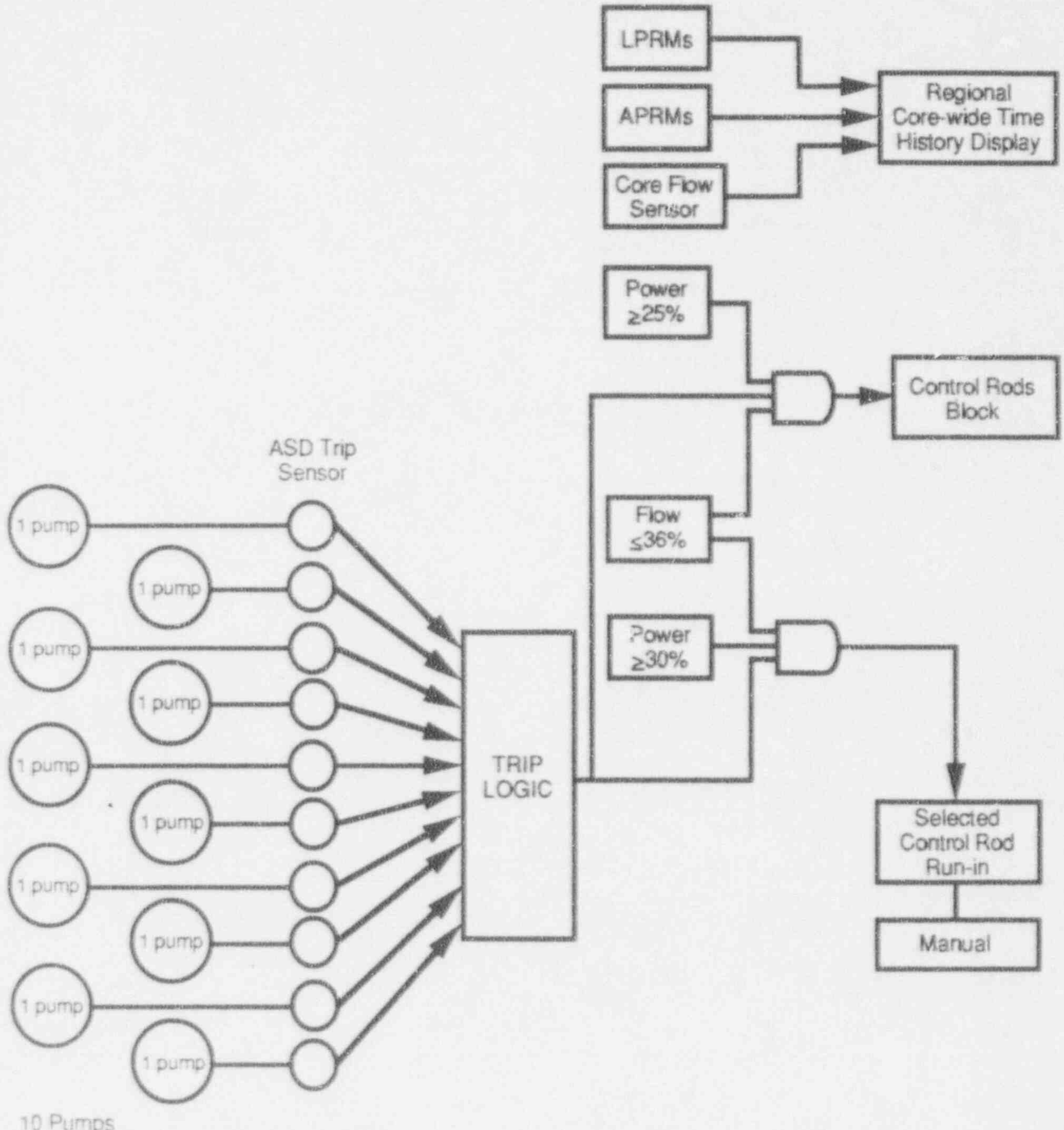
Figure ~~2~~ 4.4-3 ABWR Stability



ABWR is stable in normal operating domain

NEW FIGURE 4.4-3



Notes:

1. Power ≥ 30%: To assure power level below 80% rod line at natural circulation.
2. Flow ≤ 36%: To assure flow rate is higher than that of eight RIPs operations with minimum pump speed

Figure ~~2~~ 4.4-4 Stability Controls and Protection Logic

NEW FIGURE 4.4-4

5.3.2 Pressure/Temperature Limits

5.3.2.1 Limit Curves

The pressure/temperature limit curves in Figure 5.3-1 are based on the requirements of 10CFR50, Appendix G. The prressure/temperature limits look different than SRP Section 5.3.2 because the ABWR temperature limits are based on a more recent revision of Regulatory Guide 1.99.

All the vessel shell and head areas remote from discontinuities plus the feedwater nozzles were evaluated, and the operating limit curves are based on the limiting location. The boltup limits for the flange and adjacent shell region are based on a minimum metal temperature of RT_{NDT} plus 33°C. The maximum throughwall temperature gradient from continuous heating or cooling at 55.5°C per hour was considered. The safety factors applied were as specified in ASME Code, Appendix G, and Reference 2.

The material for the vessel will be provided with the following requirements of RT_{NDT} as determined in accordance with Branch Technical Position MTEB 5-2: shell and flanges -229°C; nozzles - 29°C and welds - 29°C.

See Subsection 5.3.4.3 for COL license information.

5.3.2.1.1 Temperature Limits for Boltup

Minimum closure flange and fastener temperatures of RT_{NDT} 33°C are required for tensioning at preload condition and during detensioning. Thus, the limit is -29°C + 33°C = + 4°C.

5.3.2.1.2 Temperature Limits for ISI Hydrostatic and Leak Pressure Tests

Pressure (measured in the top head) versus temperature (minimum vessel shell and head metal temperature) limits to be observed for the test and operating conditions are specified in Figure 5.3-1. Temperature limits for preservice and inservice tests are shown in Curve A of Figure 5.3-1.

5.3.2.1.3 Operating Limits During Heatup, Cooldown, and Core Operation

Heatup and Cooldown.

Curve B in Figure 5.3-1 specifies limits for non-nuclear heatup and cooldown following a nuclear shutdown.

Reactor Operation

Curve C in Figure 5.3-1 specifies limits applicable for operation whenever the core is critical except for low-level physics tests.

5.3.2.1.4 Reactor Vessel Annealing

Inplace annealing of the reactor vessel, because of radiation embrittlement, is not anticipated to be necessary.

5.3.2.1.5 Predicted Shift in RT_{NDT} and Drop in Upper-Shelf Energy (Appendix G-IV B)

For design purposes the adjusted reference nil ductility temperature and drop in the upper-shelf energy for BWR vessels is predicted using the procedures in Regulatory Guide 1.99.

The calculations (see response to Question 251.5) are based on the specified limits on Phosphorous (0.020%), Vanadium (0.05%), Copper (0.08%) and Nickel (1.2%) in the weld material. In plate material, the limits are Copper (0.05%) and Nickel (0.73%). Forgings will have the same chemitry as plate but the nickel limit is 1%.

The ABWR neutron fluences are low when compared with the past reactors because of the fact that the incorporation of internal pumps increased the annulus between the shourd and the vessel wall.

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A surveillance program in accordance with ASTM E 185 will be used. The surveillance program will include samples of base metal, weld metal and heat affected zone material. Subsection 5.3.1.6 provides added detail on the surveillance program.

5.3.2.2 Operating Procedures

A comparison of the pressure versus temperature limit in Subsection 5.3.2.1 with intended normal operation procedures of the most severe service level B transient shows that those limits will not be exceeded during any foreseeable upset condition. Reactor operating procedures have been established so that actual transients will not be more severe than those

OI 5.3.2-1

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The evaluation of fast neutron fluence for the ABWR vessel was done using the Oak Ridge National Lab code DOT-4 on a CRAY X-MP Super Computer using an eighth core symmetry fixed source model. The neutron source was based upon a three dimensional nodal fuel model of ABWR for an integrated equilibrium core with a 26 group neutron spectrum. The results shown below are reasonable in comparison to the BWR/6 calculations which were performed with an older version of DOT. In this comparison, the BWR/6 40 year quarter thickness evaluations for the 218-624 plant were compared to the 40 year BWR/6 238-748 plant and the 40 year ABWR values which are shown on line three of ~~the table below~~ Table 5.3-1. In evaluating the relative fluence, the power level and shroud to vessel water thickness were taken into account. In the case of the water thickness, the neutron reduction factor was interpolated from ~~the attached~~ figure 5.3-3 which shows the calculated fast neutron flux for an annular region as a function of water thickness. The incorporation of internal pumps increased the annulus between the shroud and the vessel wall for ABWR. This leads to an order of magnitude reduction in the expected fast fluence.

OI 5.3.2-1

TABLE 5.3-1

	BWR/6		ABWR
	218-624	238-748	
Peak Fluence (40yr) (1/4 t)	5.5E+18	4.3E+18	2.2E+17
Power (MWt)	2894	3579	3926
Bundles	624	748	872
Power Lvl (kw/l)	52.8	54.5	51.3
Vessel IR	276.86	302.26	353.06
Shroud OR (cm)	234.95	256.54	280.35
Water Gap (cm)	41.9	45.7	72.7
Neutron Reduction Factor for Water	0.007	0.0044	0.00042
Expected Fast Fluence based upon 218-624	5.5E+18	3.6E+18	3.4E+17

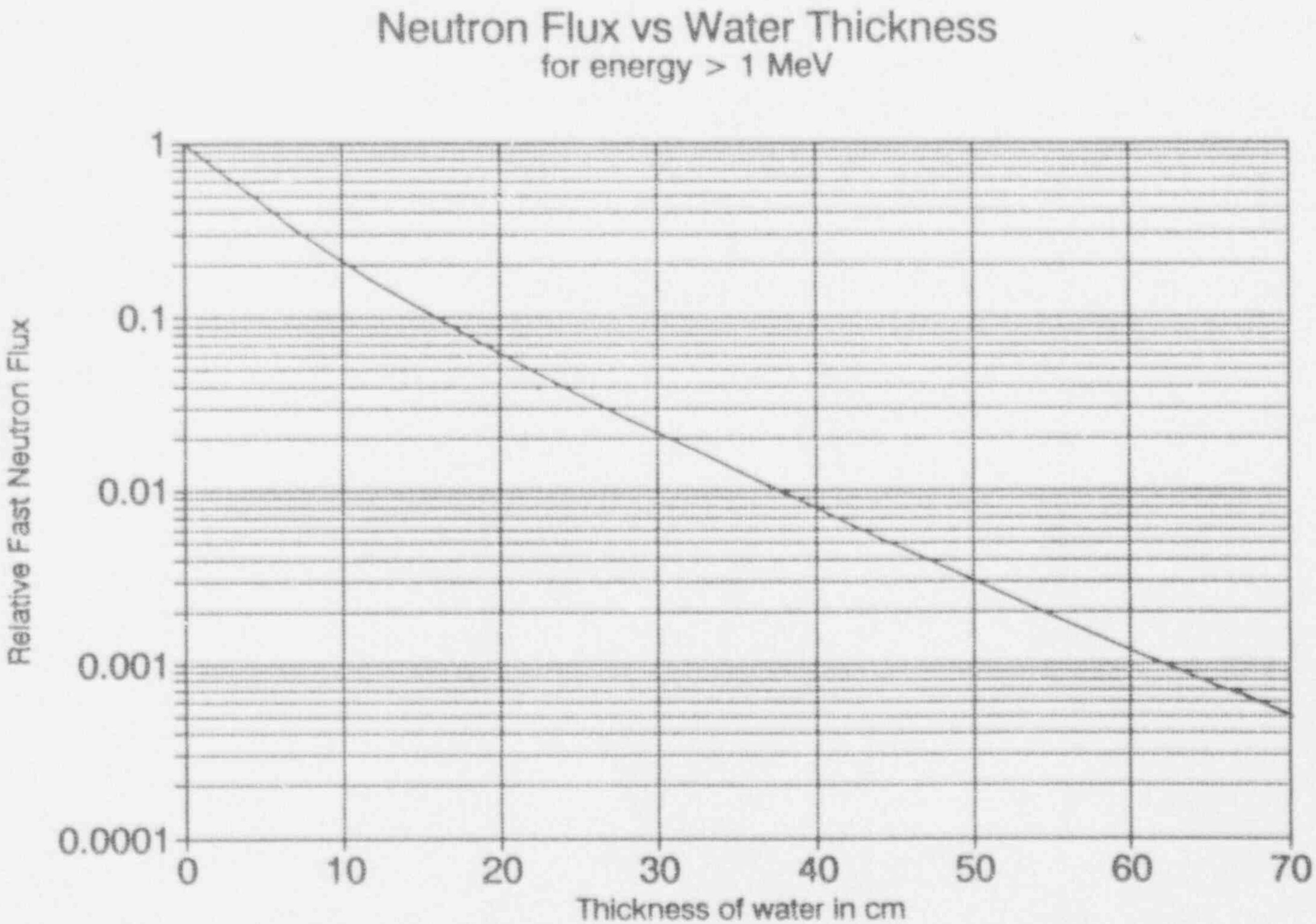

Figure ~~3.4.2.2~~ 5.3-3 Fast Neutron Flux as a Function of Water Thickness

OI 15.3-1

vendor fuel. Mixed core effects are included in the calculation.

When loading GE fuel into a core with other vendor fuel, GE will use the nuclear libraries calculated fro the other vendor fuel and the actual fuel bundle parameters of the other vendor fuel in a mixed core analysis to demonstrate the adequacy of the plant overpressure system.

4B.11 LOSS-OF-COOLANT ACCIDENT ANALYSIS METHODS

(1) The criteria in 10CFR50.46 is met by either plant-specific or bounding analyses.

The criteria is currently met by plant exposure dependent, bundle/lattice specific MAPLHGR values which must be met during plant operation. In the future, other criteria or bounding analyses may be approved by the NRC.

(2) MAPLHGR adjustment factors are utilized for each design if required for operating flexibility options.

Plant MAPLHGR adjustment factors for operation in a configuration or region requiring revised MAPLHGR values such as single recirculation loop operation must be confirmed for each new fuel design. This will be done for each plant prior to the cycle of operation of the new fuel design in that plant.

(3) When GE fuel is loaded into a core with other vendor fuel, and, when no new system analysis is performed, MAPLHGR values for other vendor fuel will be those calculated by that vendor.

The LOCA analysis used to calculate fuel MAPLHGR values is independent of the other fuel in the core. Therefore, insertion of GE fuel into a core with other vendor fuel will not impact the results of the LOCA analyses previously performed for the other vendor fuel if no new system analysis is performed.

4B.12 ROD DROP ACCIDENT ANALYSIS

(1) Plant cycle specific analysis results, if analysis is required, shall not exceed the licensing limit.

(2) Applicability of the bounding BPWS analysis must be confirmed if analysis is required.

4B.13 REFUELING ACCIDENT ANALYSIS

The consequences of refueling accidents are confirmed as bounding or a new analysis shall be performed and documented when a new fuel design is introduced.

The consequences of the refueling accident are primarily dependent upon the number of fuel rods in a bundle. When the number of fuel rods changes, the effect on the refueling accident must be determined based on the fuel design information report.

The fuel misorientation event shall be analyzed and documented. Its consequences should meet all criteria approved by the NRC.

4B.14 ANTICIPATED TRANSIENT WITHOUT SCRAM

The fuel must meet either criteria (1) or (2) below:

(1) A negative core moderator void reactivity coefficient, consistent with the analyzed range of void coefficients provided in Appendix 15E, shall be maintained for any operating conditions above the startup critical condition.

(2) If criterion (1) above is not satisfied, the limiting events (as described in Appendix 15E) will be evaluated to demonstrate that the plant response in within the ATWS criteria specified in Appendix 15E.

4B.15 UNACCEPTABLE RESULTS FOR INFREQUENT INCIDENTS (UNEXPECTED OPERATION OCCURRENCES)

The following are considered to be unacceptable safety results for infrequent incidents (unexpected operational occurrences):

(1) release of radioactivity which results in dose consequences that exceed small fraction (10 percent) of 10CFR100.

(2) failure of fuel cladding which could cause changes in core geometry such that core cooling would be inhibited.

OI 15.3-1

worst possible location, and the plant is operated with the mislocated bundle. This event is categorized as a limiting fault based on the following data:

Expected Frequency: 0.002 events/operating cycle.

This number is based upon past experience.

15.4.7.2 Sequence of Events and Systems Operation

15.4.7.2.1 Sequence of Events

The postulated sequence of events for the misplaced bundle accident (MBA) is presented in Table 15.4-6.

15.4.7.2.2 Systems Operation

A fuel loading error, undetected by in-core instrumentation following fueling operations, may result in an undetected reduction in thermal margin during power operations. For the analysis reported herein, no credit for detection is taken and, therefore, no corrective operator action or automatic protection system functioning is assumed to occur.

15.4.7.3 Core and System Performance

This event is presented in Subsection S.2.5.4 of Reference 1.

Mislocated bundle analyses are not performed for reload cores because, based on analysis of data available from past reloads, the probability that a mislocated fuel bundle loading error will result in a CPR less than the safety limit is sufficiently small (see Reference S.2-58 of Reference 1).

For ABWR initial core, the mismatch of exposures and integrated bundle power between mislocated bundles are less severe than the equilibrium cycle. Therefore, the consequence of a postulated misplaced bundle accident for the initial core is less severe than that for the equilibrium cycle. Consequently, the conclusion drawn from the reload core analysis as previously presented is applicable to the ABWR initial core. Hence, no specific analysis is required.

15.4.7.4 Barrier Performance

An evaluation of the barrier performance is not made for this event, because it is a mild and highly localized event. No perceptible change in the core pressure is observed.

15.4.7.5 Radiological Consequences

An evaluation of the radiological consequences is not required for this event, because no radioactive material is released for the fuel.

15.4.8 Rod Ejection Accident

15.4.8.1 Identification of Causes and Frequency Classification

The rod ejection accident is caused by a major break on the FMCRD housing, outer tube or associated CRD pipe lines. Due to a break of this type, the reactor pressure exerted on the CRD spud pushes down the hollow piston and the ballnut with a large force. The shaft screw and the motor are forced to unwind. A passive brake mechanism is installed in the FMCRD system to prevent the control rod from moving. The design of the brake is presented in Section 4.6.1. The probability of the initial causes, i.e., a CRD pipe line break or housing break, is considered low enough to warrant its being categorized as a limiting fault. Even if this accident does happen, the brake prevents the control rod from ejection. Should the brake fail, the check valve will serve as a backup brake to prevent the rod ejection.

15.4.8.2 Sequence of Events and Systems Operation

If a major break occurs on the FMCRD housing, the reactor pressure will provide forces that could cause the shaft screw to unwind. The FMCRD brake mechanism prevents the rod from moving. Therefore, no rod ejection can occur.

The COL applicant will provide an analysis to confirm that the consequences of a fuel misorientation event meet all requirements approved by the NRC. See subsection 15.4.10.1 for COL license information.

OI 15.3-1

control rod from the hollow piston of the FMCRD. If the control rod is stuck, the separation-detection devices will detect the separation of the control rod and hollow piston from the ballnut of the FMCRD and rod block interlock will prevent further rod withdrawal. The operator will be alarmed for this separation.

There is no basis for the control rod drop event to occur.

15.4.9.3.2 Identification of Operator Actions

No operator actions are required to preclude this event. However, the operator will be notified by the separation-detection alarm if separation is detected.

15.4.9.4 Core and System Performance

The performance of the separation-detection devices and the rod block interlocks virtually preclude the cause of a rod drop accident.

15.4.9.5 Barrier Performance

An evaluation of the barrier performance is not made for this accident since there is no circumstance for which this event could occur.

15.4.9.6 Radiological Consequences

The radiological analysis is not required.

15.4.~~10~~ 11 References

1. *General Electric Standard Application for Reactor Fuel--United States Supplement*, NEDE-24011-P-A-US, (Latest approved revision).

2. C. J. Paone and J. A. Woolley, *Rod Drop Accident Analysis for Large Boiling Water Reactors*, Licensing Topical Report, March 1972 (NEDO-10527, Supplements 1 and 2).

15.4.10 COL License Information

15.4.10.1 Fuel Misorientation Event Analysis

COL applicants will provide an analysis to confirm that the consequences of a fuel misorientation event meet all requirements approved by the NRC. (See Subsection 15.4.7.3).