The B&W Owners Group

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SAFETY CRITERIA AND METHODOLOGY FOR ACCEPTABLE CYCLE RELOAD ANALYSES

THE B&W *OWNERS GROUP*

CORE PERFORMANCE COMMITTEE

AREVA

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BAW-10179

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ABSTRACT

Subsequent to the issuance of NRC Generic Letter 88-16, the B&W Owners Group (BWOG) authorized the preparation of topical report BAW-10179P-A entitled "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses." BAW-10179P-A was reviewed and approved by the NRC and is referenced in the Technical Specification reference for Core Operating Limit Report parameters for B&W-designed nuclear plants.

Since the original approval, five revisions to the report have been issued in the form of multiple appendices to incorporate additional NRC-approved codes and methods. In addition, the mechanical design code COPERNIC has received NRC approval, and a modified zero power physics testing program for B&W 177-FA plants has been incorporated. The purpose of Revision 7 is to:

1. Incorporate the appendices from Revisions 1 through 6 into the main body of the report,
2. Update the methodology to incorporate the new NRC-approved design code COPERNIC,
3. Provide a summary of the modified zero power physics testing program, and
4. Update the methodology to incorporate the new NRC-approved statistical fuel assembly hold down methodology,
5. Add clarification where needed and remove unnecessary information.
6. Provide generic guidelines on the use of limited scope high burnup lead test assemblies (LTAs) and satisfy the requirement to incorporate WCAP-15604-NP, Revision 2-A ("Limited Scope High Burnup Lead Test Assemblies) explicitly into the licensee's Technical Specifications by virtue of it being referenced in BAW-10179P-A.

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

LIST OF ACRONYMS

AMSAC	ATWS Mitigation System Actuation Circuitry
ANSI	American National Standards Institute
APSR	Axial Power Shaping Rod
AROCBC	All Rods Out Critical Boron Concentration
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
B_{eff}	Effective Delayed Neutron Fraction
B&W	The Babcock and Wilcox Company
BOC	Beginning-of-Cycle
BPRA	Burnable Poison Rod Assembly
BWOG	B&W Owners Group
CFM	Centerline Fuel Melt
CFR	Code of Federal Regulations
CHF	Critical Heat Flux
COLR	Core Operating Limits Report
CRA	Control Rod Assembly
CRAFT2	CRAFT2 Transient Analysis Computer Code
CRDM	Control Rod Drive Mechanism
CRG	Control Rod Group
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
DSS	Diverse SCRAM System
ECCS	Emergency Core Cooling System
EFPD	Effective Full Power Days
EM	Evaluation Model
EOC	End-of-Cycle
EPRI	Electric Power Research Institute
FANP	Framatome ANP, Inc.
FFCD	Final Fuel Cycle Design
FP	Full Power (Full Rated Thermal Power)
FSAR	Final Safety Analysis Report
GDC	General Design Criteria
GWd/mtU	Gigawatt Days per Metric Ton of Uranium
HFP	Hot Full Power
HPI	High Pressure Injection
HZP	Hot Zero Power
ID	Inner Diameter

LIST OF ACRONYMS

ITC	Isothermal Temperature Coefficient
LBB	Leak-Before-Break
LBLOCA	Large Break Loss-of-Coolant Accident
LCO	Limiting Condition for Operation
LHR	Linear Heat Rate
LOCA	Loss-of-Coolant Accident
LOCF	Loss-of-Coolant-Flow
LTA	Lead Test Assembly
MAP	Maximum Allowable Peaking
MTC	Moderator Temperature Coefficient
MWd/mtU	Megawatt Days per Metric Ton of Uranium
NRC	Nuclear Regulatory Commission
NSAC	Nuclear Safety Analysis Center
NSSS	Nuclear Steam Supply System
OBE	Operational Basis Earthquake
OD	Outside Diameter
pcm	Percent millirho (a reactivity measurement unit)
PCT	Peak Cladding Temperature
PIE	Post-Irradiation Examination
ppm	Parts Per Million
PSC	Preliminary Safety Concern
P-T	Pressure-Temperature
PV	Pressure-Velocity
RC	Reactor Coolant
RCS	Reactor Coolant System
RELAP5	RELAP5/MOD2-B&W Transient Analysis Computer Code
RIL	Rod Insertion Limit
RPD	Relative Power Density
RPS	Reactor Protection System
RTP	Rated Thermal Power
SAFDL	Specified Acceptable Fuel Design Limit
SAR	Safety Analysis Report
SBLOCA	Small Break Loss-of-Coolant Accident
SCD	Statistical Core Design
SDL	Statistical Design Limit
SDM	Shutdown Margin
SER	Safety Evaluation Report

LIST OF ACRONYMS

SGTR	Steam Generator Tube Rupture
SRSS	Square-Root-of-Sum-of-Squares
SSE	Safe Shutdown Earthquake
T_{avg}	Average Reactor Coolant Temperature
TCS	Transient Cladding Strain
TD	Theoretical Density
TDL	Thermal Design Limit
TIL	Time-in-Life (refers to burnup between 0 GWd/mtU and maximum analyzed burnup)
VLPT	Variable Low Reactor Coolant Pressure Trip
VMFT	Vessel Model Flow Test
ZPPT	Zero Power Physics Testing
4PSUT	Fourth Pump Startup Temperature
177-FA	177 Fuel Assembly

SECTION 1 INTRODUCTION

Nuclear Regulatory Commission (NRC) Generic Letter 88-16, issued in October 1988, allows the removal of cycle-dependent variables from technical specifications provided the values of these variables are determined with NRC-approved methodology and are included in a Core Operating Limits Report (COLR). In reference 1 the NRC agreed that this philosophy can be extended to the cycle-dependent protective and maximum allowable setpoint limits. Framatome ANP, Inc. (FANP) designs and fabricates fuel, and prepares reload safety evaluations for a number of B&W 177 fuel assembly (177-FA) nuclear power plants. All of these plants use Mark-B fuel assemblies. The methodology for performing reload design evaluations for this class of plants operating with Mark-B fuel is presented in topical report BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses." The utility owners for whom FANP performs reload safety evaluations reference BAW-10179P-A in the Administrative Controls sections of the plant technical specifications. The technical specifications identify BAW-10179P-A as the NRC-approved FANP methodology for determining the limits contained in the COLR. The technical specifications also state that the latest approved revision of BAW-10179P-A shall be specified in the COLR.

The COLR for a B&W-designed plant typically contains the following parameters:

- Control rod physical insertion, sequence, and overlap limits
- Control rod program
- Axial power shaping rod (APSR) insertion limits
- Axial power imbalance operating limits
- Quadrant power tilt limits
- End-of-cycle moderator temperature coefficient (MTC)
- Nuclear heat flux hot channel factor limit, F_Q^N
- Nuclear enthalpy rise hot channel factor limit, $F_{\Delta H}^N$
- Refueling boron concentration
- Axial power imbalance protective limits
- Trip setpoint for nuclear overpower based on reactor coolant system (RCS) flow

These parameters were identified by the NRC in the original approval of BAW-10179P-A. Individual utility owners of the B&W-designed plants have successfully negotiated with the NRC to include additional parameters in their COLRs.

Since the original approval, five revisions to the report have been issued in the form of multiple appendices to incorporate additional NRC-approved codes and methods. In addition, the mechanical design code COPERNIC has received NRC approval, and a modified zero power physics testing program for B&W 177-FA plants has been incorporated. The purpose of Revision 7 is to:

- Incorporate the appendices from Revisions 1 through 5 into the main body of the report,
- Update the methodology to incorporate the NRC-approved design code COPERNIC (see Sections 4.2 and 9.2.3),
- Provide a summary of the modified zero power physics testing program, and
- Add clarification where needed and remove unnecessary information.
- Provide generic guidelines on the use of limited scope high burnup lead test assemblies (LTAs) and satisfy the requirement to incorporate WCAP-15604-NP, Revision 2-A ("Limited Scope High Burnup Lead Test Assemblies) explicitly into the licensee's Technical Specifications by virtue of it being referenced in BAW-10179P-A.

FANP has developed criteria for determining when a design change must be submitted to the NRC for review and approval. The criteria are:

1. The change meets any of the eight criteria specified in paragraph (c)(2) of 10CFR50.59.
2. A change to the plant technical specifications is required.
3. The applicability of NRC-approved design/analysis evaluation methods is affected.
4. A material not previously qualified for in-reactor operation in a similar application is introduced.
5. Burnup limits are extended beyond those previously approved.

FANP has developed criteria for determining when a change in evaluation methods must be submitted to the NRC for review and approval. The criteria are:

1. The change consists of replacement of an existing approved design code or method.
2. The core power distribution monitoring methodology is changed.
3. An approved method is extended beyond previously acceptable limits.

Any design or evaluation method changes that meet the above criteria must receive NRC review and approval prior to implementation.

Limitations and conditions specified in the safety evaluation reports (SERs) of referenced NRC-approved topical reports are applicable to BAW-10179P-A unless specifically stated otherwise in this report.

All currently operating nuclear power plants store spent fuel assemblies on site. For economic and safety purposes, some of these assemblies are periodically returned to the reactor for additional cycles of operation. These assemblies were analyzed with the methodology that was current at the time the fuel was fabricated. Some of this methodology has changed as nuclear technology evolved over the lifetime of the nuclear power plants. Design parameters (e.g., burnup, pin power, CFM limit, LOCA limit, etc.) for some of the fuel assemblies were determined with codes or methods that are no longer used. However, these methods are NRC-approved and design parameters determined with these methods remain valid and applicable for reinserted fuel assemblies.

This report describes the entire spectrum of methodologies that are applicable to the reload fuel currently supplied by FANP for the B&W 177-FA plants. An overview of the design considerations addressed in these methods is provided in Section 2. A brief description of the Mark-B fuel design is provided in Section 3. The mechanical design methods are described in Section 4. The nuclear design methods are described in Section 5, which also includes the radiological evaluation parameters. The thermal-hydraulics methods are described in Section 6. Section 7 includes the methods for determining reactor protection system (RPS) trip setpoints. Section 8 describes the non-LOCA accident evaluation methods, and Section 9 presents the LOCA accident evaluation methods. Figure 1-1 provides an overview on how the different analytical disciplines interact to complete a reload evaluation. Section 10 provides generic guidelines on the use of limited scope high burnup lead test assemblies.

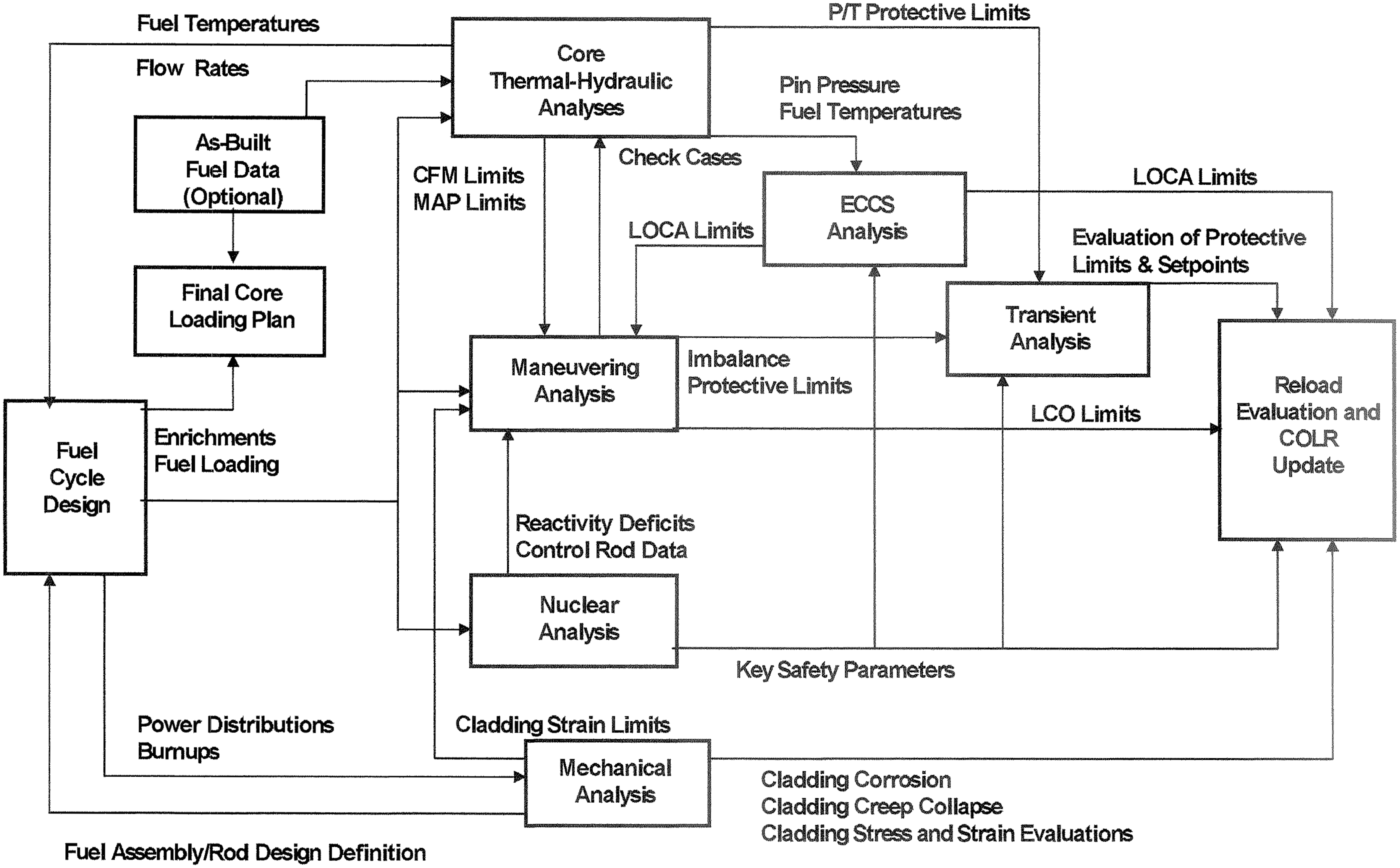
The methodology described in this report is constantly evolving to include improvements and enhancements to analytical techniques. This will result in a succession of updates to BAW-10179P-A. To facilitate these updates FANP will implement the following procedure:

1. For revisions to NRC-approved topical reports already referenced in BAW-10179P-A, such revisions will be incorporated by referencing the latest approved revision in the COLR.

2. For new methodology topical reports, FANP will prepare a corresponding revision to BAW-10179P-A and include it with the submittal. The revision to BAW-10179P-A will include an appendix that provides a brief summary of the methodology topical and its range of applicability. When the NRC completes its review of the methodology topical, a single SER will be issued which approves both the methodology topical and the revision to BAW-10179P-A. Accepted versions of both topical reports will then be prepared and the latest revision of BAW-10179P-A will be available for referencing in the plant COLR. Any NRC conditions or limitations on the methodology will be included in the accepted version of BAW-10179P-A.

Figure 1-1

Reload Evaluation Flowchart



SECTION 2 DESIGN CONSIDERATIONS

2.1 Safety Criteria

The safety criteria for the design of nuclear power plants are provided in various parts of Title 10 of the Code of Federal Regulations (CFR). The radiological dose criteria are located in 10CFR100 (reference 2). The acceptance criteria for emergency core cooling systems are given in 10CFR50.46 (reference 3). The general design criteria (GDC) are found in 10CFR50 Appendix A (reference 3).

The criteria given in 10CFR50.46 and 10CFR100 are quantitative in nature and are well defined. Compliance with these criteria can be demonstrated by analysis. The general design criteria found in Appendix A, as the name implies, are not so specific. Many of these criteria use the term "specified acceptable fuel design limits" (SAFDLs). These SAFDLs take many forms in the mechanical, nuclear, thermal-hydraulic, and safety analyses of light-water reactor fuel. Each design discipline has a set of parameters that is determined to show compliance with the GDC in Appendix A. The specific criteria for each type of analysis are given in the individual sections of this report.

2.2 Plant Conditions

The normal operation and possible transient modes of nuclear plants are categorized into four conditions commonly referred to as normal, moderate frequency, infrequent incidents, and limiting faults. The specific definitions for these conditions are taken from ANSI/ANS-57.5-1981.

2.2.1 Condition I -- Normal Operation and Operational Transients

Condition I events are those that are expected frequently or regularly in the normal course of power operation. The design requirement for these events is that they shall be accommodated with margin between any plant parameter and the value of that parameter which would require either automatic or manual protective action.

2.2.2 Condition II -- Events of Moderate Frequency

Condition II events are those that are expected to occur during the life of a plant that may result in reactor shutdown. The design requirement for these events is that they shall be accommodated with, at most, a shutdown of the reactor with the plant capable of returning to power operation after corrective action.

2.2.3 Condition III -- Infrequent Events

Condition III events are incidents that may occur infrequently, if at all, during the life of the plant. The design requirement for these events is that they shall not cause more than a small fraction of the fuel elements in the reactor to be damaged, although sufficient fuel element damage might occur to preclude resumption of operation for a considerable outage time.

2.2.4 Condition IV -- Limiting Faults

These events are not expected to occur, but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. Condition IV events represent the limiting design case. The design requirement for these events is that they shall not cause a release of radioactive material that results in an undue risk to public health and safety exceeding the guidelines of 10CFR100. A single Condition IV event shall not cause a consequential loss of system functions needed to cope with the event.

The NRC categorizes plant operation into three conditions. They are normal operation, anticipated operational occurrences, and accidents. Figure 2-1 shows the relationship between the NRC scheme and the ANS-57.5 scheme. FANP assures compliance with the NRC regulations by requiring the limiting Condition III transient to meet the acceptance criteria for Condition II events.

Figure 2-1

Categories of Events

EVENT FREQUENCY RANGE (per reactor-year)	NRC 10CFR50	FANP ANSI/ANS-57.5
Planned Operations	Normal	Condition I
	Anticipated Operational Occurrences	Condition II
10^{-1}		
		Condition III
10^{-2}		
10^{-3}		
10^{-4}	Accidents	Condition IV
10^{-5}		
10^{-6}		

SECTION 3 REFERENCE FUEL DESCRIPTION FOR MARK-B DESIGN

3.1 Fuel Pellet

The fuel consists of cylindrical pellets. The pellets are sintered and ground, and contain low enriched uranium dioxide. The pellet ends are dished to minimize differential thermal expansion between the fuel and cladding.

3.2 Fuel Rod

The fuel rod consists of fuel pellets, cladding, a spring system, and end caps. The spring system consists of springs located below and/or above the pellet stack in the fuel rod. The spring system is designed to accommodate maximum thermal expansion of the fuel column without being deflected beyond solid height and to minimize gaps forming within the fuel rod internals during shipping and handling. All fuel rods are internally pressurized with helium.

3.3 Fuel Assembly

The standard Mark-B fuel assembly, as fabricated, consists of 208 fuel rods, 16 control rod guide tubes, 1 instrumentation tube assembly, 8 spacer grids, and 2 end fittings. The guide tubes, spacer grids, and end fittings form a structural cage to arrange the rods and tubes in a 15 x 15 array. The fuel rod outside diameter (OD) is [ ] inches. The typical grid configuration is non-mixing. As an option to enhance fretting resistance, the standard Mark-B product incorporates the HTP grid (reference 4). The center position in the fuel assembly is reserved for instrumentation. Product enhancements, which are described in applicable reload safety evaluation documents, have been made to the Mark-B design. Significant changes include modifications to the end fittings to improve debris resistance and facilitate reconstitution. A [ ] to optimize the hold down system and an enhanced grid restraint system have also been incorporated. The NRC staff has found the standard Mark-B fuel to be acceptable to a rod average burnup of [ ] GWd/mtU.

Table 3-1 lists various parameters and values for the Mark-B standard fuel design. The information provided in this table is typical for Mark-B fuel designs incorporating Zircaloy-4 and M5 material. A typical Mark-B fuel assembly sketch is shown in Figure 3-1.

3.4 Alternate Mark-B Fuel Designs

3.4.1 Small Pin Fuel Assembly Design

BAW-10229P-A (reference 5) provides the licensing bases for the small pin fuel assembly design. The small pin design features a [ ] inch OD fuel rod to reduce uranium requirements and mixing vane grids to provide superior thermal performance.

The NRC staff has found the small pin design to be acceptable to a rod average burnup of [ ] GWd/mtU. Cycle-specific reload safety evaluations performed by licensees incorporating the small pin design address the same criteria as those considered for the standard Mark-B fuel assembly.

Table 3-1

Typical Mark-B Fuel Assembly Parameters †

Assembly Designation	Mark-B
Fuel Rod Array	15x15
Hold Down Spring	Helical Coil Spring or Multiple Leaf Spring
Cladding Material	Zr-4, M5
Guide Tube Material	Zr-4, M5
Assemblies per Core	177
Fuel Rods per Assembly	208
Control Rod/Guide Tube/Instrument Tube Locations Per Assembly	17
Debris Protection Feature	Solid Lower End Plug*
Rod Pitch, mm (inch)	14.4 (0.568)
Fuel Rod Length, cm (inch)	[ ]
Active Fuel Height, cm (inch)	[ ]
Plenum Length, cm (inch)	[ ]
Fuel Rod O.D., mm (inch)	[ ]
Cladding I.D., mm (inch)	[ ]
Cladding Thickness, mm (inch)	[ ]
Diametrical Gap, microns (mils)	[ ]
Fuel Pellet O.D., mm (inch)	[ ]
Fuel Pellet Density, %TD	[ ]
Average LHGR, W/cm (kW/ft)	203 (6.20)
System Pressure, MPa (psia)	15.2 (2200)
Core Inlet Temperature, °C (°F)	292.07 (557.7)
Core Outlet Temperature, °C (°F)	315.7 (600.3)

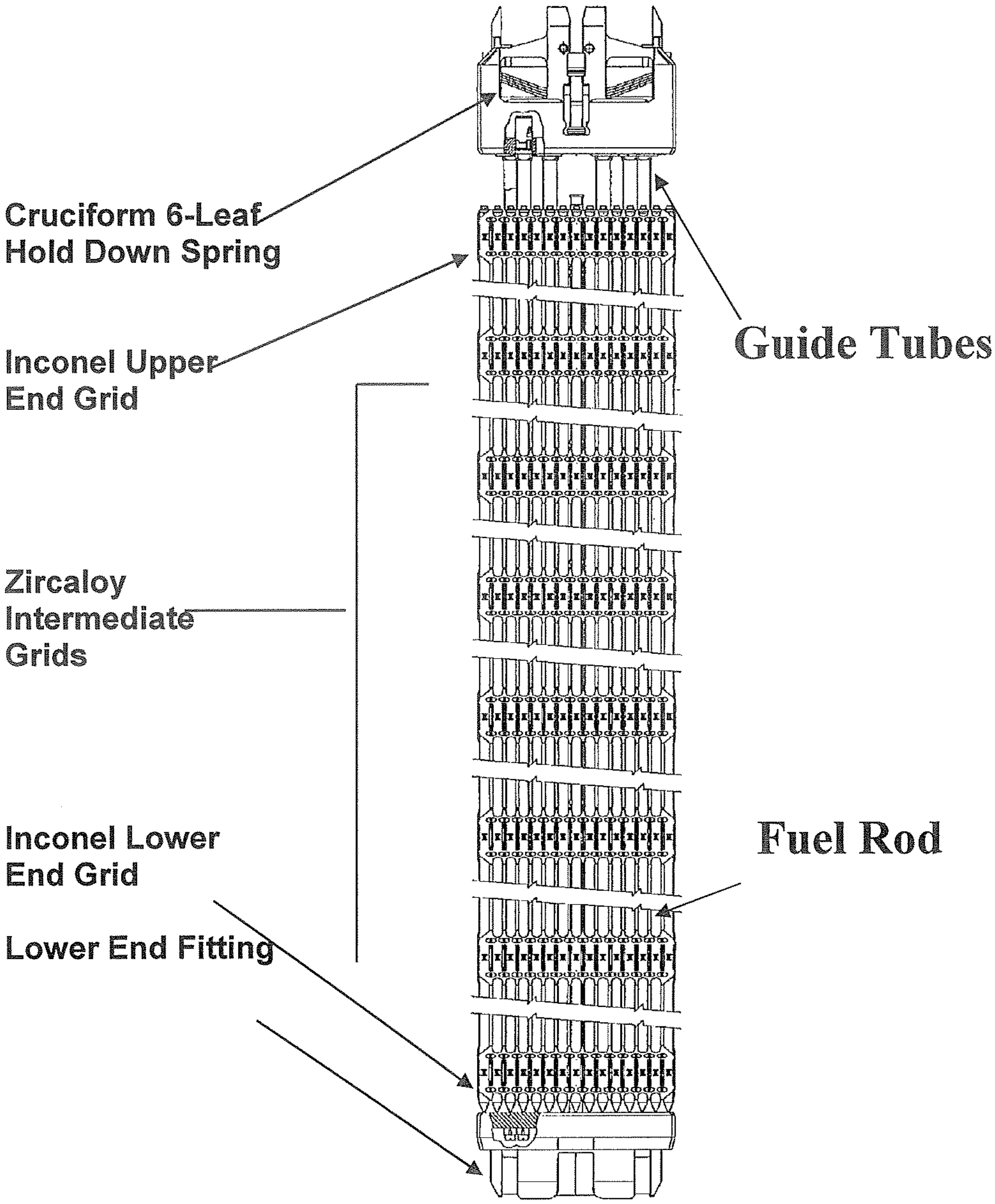
† Designs, materials and dimensions are representative of those used to date. Alternates may be used if they are demonstrated to meet the burnup requirements.

* Other options available.

** Design has used both densities.

Figure 3-1

Typical Mark-B Fuel Assembly Sketch



SECTION 4 MECHANICAL SAFETY AND DESIGN CRITERIA

The mechanical design and operation of the fuel assembly will ensure that under all operating conditions the maximum credible damage will not degrade the design below those capabilities assumed in the safety analysis. The mechanical safety and design criteria compose two areas discussed below.

1. Fuel and control system capabilities are greater than or equal to those assumed in the safety analyses. This criterion is assured when the following three conditions are met:
   a) Fuel rod cladding integrity is maintained.
   b) The control rod insertion path remains open.
   c) A coolable rod geometry is maintained.
2. Fuel and control system dimensions remain within operational tolerances.

Systems that are covered by this section include the fuel assembly and the fuel rod.

4.1 Fuel Assembly Design

The fuel assembly design criteria make certain that the fuel assembly with the maximum credible damage will be able to ensure that a path for control rod insertion remains, that a coolable fuel rod geometry remains and that the fuel assembly dimensions remain within operational limits. Compliance with the criteria in the following sections will ensure that the fuel assembly can meet those requirements.

4.1.1 Growth

4.1.1.1 Analysis Criteria

The gap allowance between the fuel assembly and the reactor internals and the growth allowance gap between the upper end fitting and the fuel rod shall be designed to provide a positive clearance during the assembly lifetime.

4.1.1.2 Analysis Method

4.1.2 Stress

4.1.2.1 Analysis Criteria

The stress intensities in the upper and lower end fittings (excluding the hold down spring), the guide tubes, and the fuel rods (Condition IV only) shall be less than the limits shown below. The fuel rod design criteria for Conditions I and II are covered separately in Section 4.2. All stress nomenclature is per the ASME Boiler and Pressure Vessel Code, Section III (reference 6).

Condition	Stress Intensity Limit
Conditions I & II	$Pm \leq 1.0\ Sm$
	$Pl \leq 1.5\ Sm$
	$Pm + Pb \leq 1.5\ Sm$
	$Pm + Pb + Q \leq 3.0\ Sm$
Conditions III & IV	$Pm \leq 2.4\ Sm$ or $0.7\ Su$
	$Pm + Pb \leq 3.6\ Sm$ or $1.05\ Su$
	whichever is less

where

Pm = general primary membrane stress intensity,

Pl = local primary membrane stress intensity,

Pb = primary bending stress intensity,

Q = secondary stress,

Sm = allowable membrane stress intensity:

= 2/3 Sy or 1/3 Su, whichever is less at room temperature, or 1/3 Su or 0.9 Sy at operating temperature, but not to exceed 2/3 of the minimum specified yield strength at room temperature. These are unirradiated material properties.

Sy = minimum yield stress,

Su = ultimate stress.

4.1.2.2 Analysis Method

Normal operational loads for the guide tubes are discussed in Section 4.1.4.

4.1.3 Hold Down

4.1.3.1 Analysis Criteria

The hold down spring system shall be capable of maintaining fuel assembly contact with the lower support plate during Condition I events. The fuel assembly upper and lower end fittings shall maintain engagement with reactor internals for all Condition I through IV events. The fuel assembly shall not compress the hold down spring to solid height for any Condition I or II event.

4.1.3.2 Analysis Method

4.1.4 Buckling

4.1.4.1 Analysis Criteria

Guide tube buckling shall not occur during normal operation (Condition I) or any transient condition where control rod insertion is required by the safety analysis.

4.1.4.2 Analysis Method

4.1.5 Grids

4.1.5.1 Analysis Criteria

No crushing deformation of the spacer grids shall occur due to normal operation (Condition I) and Condition II event loadings, and the spacer grids shall provide adequate support to maintain the fuel rods in a coolable configuration for all conditions as described in reference 7.

4.1.5.2 Analysis Method

4.1.6 Fretting

4.1.6.1 Analysis Criteria

The fuel assembly design shall be shown to provide sufficient support to limit fuel rod vibration and cladding fretting wear.

4.1.6.2 Analysis Method

4.1.7 Rod Bow

4.1.7.1 Analysis Criteria

Fuel rod bowing shall be evaluated with respect to the mechanical and thermal-hydraulic performance of the fuel assembly. Fuel rod bow shall be shown to be less than the limiting bow developed in reference 9 for the Mark-B design.

4.1.7.2 Analysis Method

[ ]

4.1.8 Seismic

4.1.8.1 Analysis Criteria

The following criteria have been established for the fuel assembly seismic analysis:

1. Operational basis earthquake (OBE) - The fuel assembly is designed to ensure safe operation following an OBE.

2. Safe shutdown earthquake (SSE) - The fuel assembly is designed to allow control rod insertion and to maintain a coolable geometry.

4.1.8.2 Analysis Method

[ ]

4.1.9 LOCA

4.1.9.1 Analysis Criteria

For LOCA or a combined LOCA and SSE, the fuel assembly is designed to allow for the safe shutdown of the reactor by maintaining the overall structural integrity and a coolable geometry within deformation limits consistent with the ECCS and safety analysis as defined in reference 11.

4.1.9.2 Analysis Method

[

4.1.10 Shipping

4.1.10.1 Analysis Criteria

The design condition for shipping is [ ] axial and [ ] lateral load on the fuel assembly. The fuel assembly will be analyzed for these load limits.

The spacer grids will maintain sufficient grip on the fuel rods to prevent axial movement during shipping and handling at axial loads of up to [ ]. Lateral loads of up to [ ] will not cause setting of spacer grid spring stops.

4.1.10.2 Analysis Method

4.1.11 Material

4.1.11.1 Analysis Criteria

The materials used in the manufacture of the fuel assembly and the fuel rod must be compatible with all other materials in the primary system. That is, all core components must continue to meet their required function with the introduction of a new material.

4.1.11.2 Analysis Method

4.1.11.3 Evaluation of Advanced Cladding and Structural Material (M5)

BAW-10227P-A (reference 18) provides the justification for use of the alloy M5 to replace Zircaloy-4 in the construction of fuel assembly components such as fuel rod cladding, guide tubes, and spacer grids. Analysis criteria and methods are similar for Zircaloy-4 and M5. M5 was developed by FANP and is being implemented on a wide scale domestically and internationally. M5 provides improvements that include reduced corrosion, lower hydrogen pickup, decreased axial growth, and lower diametral creep. These improvements provide increased operating margin to the approved fuel rod average burnup limit of [ ] GWd/mtU for the Mark-B fuel designs.

4.1.12 Extended Burnup

Extended burnup operation of Mark-B fuel designs is supported by an extensive series of PIEs carried out on lead test assemblies (LTAs), demonstration assemblies, and production fuel assemblies. Prior to the submittal and approval of BAW-10186P-A (reference 9), the approved document containing the burnup limit for FANP fuel was BAW-10153P-A (reference 19). That limit was [ ] GWd/mtU batch average. Fuel rod burnup is a much better indicator of the phenomena associated with higher burnups.

During the review of BAW-10186P-A, FANP replaced the OXIDEPC corrosion model used for extended burnup applications with the COROS02 corrosion model. The NRC reviewed BAW-10186P-A for compliance with the criteria of Section 4.2 of the Standard Review Plan (NUREG-0800). In all cases, the FANP methodology was found acceptable.

On April 29, 1997, the NRC issued the SER for BAW-10186P-A. The conclusions stated in the SER that are applicable to BAW-10179P-A are as follows:

- BAW-10186P-A is acceptable for licensing applications for Mark-B fuel up to burnup levels of [ ] GWd/mtU rod average.
- The maximum corrosion limit is acceptable up to the value specified in reference 20 and supplemented with the interpretation of the application of that limit in reference 21.
- The COROS02 model is acceptable for use in predicting maximum corrosion levels.

The only limitation specified in the SER is that a penalty factor for thermal conductivity must be applied for burnups greater that [ ] GWd/mtU. This factor is defined in reference 22.

4.1.12.1 Analysis Criteria

The ability of the fuel assembly and fuel rod to maintain mechanical integrity at high burnups must be demonstrated. All design and operational criteria are the same for extended burnup fuel assemblies as for the original Mark-B fuel design.

4.1.12.2 Analysis Method

4.1.13 Stainless Steel Replacement Rod Methodology

The in-field repair of irradiated fuel assemblies with leaking rods involves the replacement of defective fuel rods with heat producing and/or non-heat producing rods. BAW-2149-A (reference 23) provides justification for the use of replacement rods without imposing unnecessary power peaking restrictions on the repaired fuel assemblies. This report addresses the nuclear, thermal-hydraulic, and mechanical aspects of the design that are affected by repair operations. The use of replacement rods for FANP supplied fuel assemblies is determined to be acceptable by the NRC per the SER included in BAW-2149-A.

The stainless steel replacement rods weigh slightly less than Zircaloy-clad fuel rods, but the effect on fuel assembly weight of up to 10 replacement rods is negligible. Therefore, the use of stainless steel replacement rods has an insignificant effect on fuel assembly hydraulic lift.

Stainless steel replacement rods are designed and analyzed to ensure that there is no adverse impact on fuel assembly performance. The rods are designed to ensure that adequate performance with respect to differential thermal expansion, irradiation growth, seismic-LOCA response, grid relaxation, and fretting due to vibration will be maintained. The replacement rods can be installed in any fuel rod location in the fuel assembly.

4.2 Fuel Rod Design

The design of the fuel rod must ensure that the integrity of the cladding is maintained under all Condition I and II events. The integrity of the cladding is maintained by requiring that the fuel rod design meet the constraints discussed below.

Analysis criteria and methods are applicable to both Zircaloy-4 and M5 cladding types as noted. The M5 cladding is approved for use in FANP fuel in reference 18.

4.2.1 Shipping

4.2.1.1 Analysis Criteria

The spring system must limit gap formation in the fuel stack during transport up to an axial loading of [ ]

4.2.1.2 Analysis Method

[ ]

4.2.2 Plenum Space

4.2.2.1 Analysis Criteria

The plenum space must be sufficient so that the spring-spacer system does not go solid with fuel stack swelling.

4.2.2.2 Analysis Method

[ ]

4.2.3 Corrosion

4.2.3.1 Analysis Criteria

The fuel rod maximum acceptable predicted oxide thickness limit is [100 microns] (reference 9) for the maximum burnup fuel rod within each core sub-batch. A sub-batch is defined as fuel that is inserted and discharged from the core at the same time so the fuel assembly residence times are identical.

4.2.3.2 Analysis Method

[

4.2.3.3 Lead Corrosion Assemblies

A lead test assembly program to continue collecting corrosion data at high burnups was approved in reference 9. The LTA program allows up to eight fuel assemblies in each fuel cycle to operate to corrosion levels in excess of the limit specified in Section 4.2.3.1. Such assemblies are designated as lead corrosion assemblies. In a given fuel core, the total number of LTAs (lead corrosion assemblies plus other LTAs) will not exceed twelve.

The eight lead corrosion assemblies may come from different sub-batches, and these assemblies will typically reside in non-limiting core locations with respect to the relative power distribution during the cycle. Corrosion measurements will be taken on the lead corrosion assemblies after they are discharged from the core to verify the cladding corrosion model predictions.

4.2.4 Creep Ovality

4.2.4.1 Analysis Criteria

Creep collapse of the cladding due to creep ovalization shall not occur during the incore life of the fuel rod.

4.2.4.2 Analysis Method

4.2.5 Stress

4.2.5.1 Analysis Criteria

4.2.5.2 Analysis Method

4.2.6 Strain

4.2.6.1 Analysis Criteria

The uniform transient strain (elastic and plastic) should not exceed [ ] This strain is defined as the transient-induced deformation with gage lengths corresponding to the cladding dimensions.

4.2.6.2 Analysis Method

4.2.7 Fatigue

4.2.7.1 Analysis Criteria

The total fatigue usage factor for all Condition I and II events shall not exceed [ ]

4.2.7.2 Analysis Method

4.2.8 Fuel Rod Pressure

4.2.8.1 Analysis Criteria

The criterion used for internal pressure evaluations is that during normal operation the maximum internal fuel rod pressure shall not exceed the pressure that would cause (1) the fuel-clad gap to increase due to outward cladding creep during steady-state operation and (2) extensive DNB propagation to occur.

4.2.8.2 Analysis Method

4.2.9 Fuel Temperature (Centerline Fuel Melt) Limit

4.2.9.1 Analysis Criteria

The predicted maximum fuel temperature at a given burnup value must be less than the melting temperature of UO_2. When the best-estimate TACO3 and GDTACO codes are used, the criterion is that the maximum predicted fuel temperature shall be less than or equal to T_L, a limit value chosen such that there will be a 95% probability at the 95% confidence level that centerline melting will not occur. When the best estimate COPERNIC code is used, the criterion is that the maximum predicted fuel temperature shall be less than or equal to the best estimate fuel melt temperature, T_m, reduced by the uncertainty in the COPERNIC centerline fuel temperature prediction.

4.2.9.2 Analysis Method

SECTION 5 NUCLEAR DESIGN

5.1 Nuclear Design Codes

All neutronic calculations described in this chapter are performed with NRC-approved design codes.

BAW-10180-A (reference 32) presents the NEMO methodology. The NEMO computer code calculates the three-dimensional core power distribution for each pin in a manner that accounts for individual pin burnup and spectral effects. NEMO also calculates control rod worth, and the reactivity effects of moderator density, fuel temperature (Doppler), and xenon. Cross section data supplied to NEMO is generated by CASMO-3 (reference 33).

BAW-10221P-A (reference 34) presents the NEMO-K methodology. NEMO-K is used where three-dimensional time dependent solutions are important. This methodology includes time-dependent solutions for neutronic, fuel temperature, and coolant properties. With the addition of the kinetics equations to the NEMO code, it can be used for static and kinetic solutions of steady state and transient problems, respectively.

5.2 Fuel Cycle Design

5.2.1 Final Fuel Cycle Design

Establishing the final fuel cycle design (FFCD) is the initial portion of the reload safety evaluation for any cycle. The objective of the FFCD is to develop the core loading plan for the fuel, and, if applicable, burnable absorbers. The control rod safety and regulating group locations are also specified, although these do not usually change from cycle to cycle. The radial power distribution is the principal focus in the development of the core loading pattern. Other considerations are discharge burnup, MTCs, ejected rod worth, and shutdown margin (SDM). Although the general pattern of fuel loading is usually predetermined (e.g., in-in-out), the specific placement of the fresh and burned assemblies, and the location and amount of burnable absorbers provide the designer with some degree of control over these parameters. Other variables include the desired cycle length, number of fresh assemblies, and options for control rod group (CRG) 7 and APSR withdrawal, average moderator temperature reduction, and power coastdown.

5.2.1.1 Acceptance Criteria

1. The maximum calculated steady-state fuel rod relative power density (RPD) for the cycle shall be less than that required to meet the safety criteria and sufficiently low to indicate that acceptable operating limits can be achieved in the power distribution analysis described in Section 5.3. The radial power distribution is controlled by the fuel shuffle and placement of burnable absorbers.

2. The discharge rod burnup shall be less than the applicable limit for that fuel design. The limit is established by the fuel mechanical and thermal design considerations (see Section 4) and the

most recent NRC-approved burnup limits (reference 9). Discharge burnup is controlled by the batch size selection for a given cycle length requirement, and the fuel shuffle pattern.

3. The design shall be capable of passing subsequent safety analysis checks described in Sections 5.4 and 8. These include moderator coefficients, ejected rod worth, and SDM among others. While licensing checks are typically not performed at the FFCD stage, preliminary calculations are made if it is judged that a particular parameter may be close to limiting based upon design changes or previous experience.

5.2.1.2 Analysis Methods

5.3 Power Distribution Analysis

5.3.1 Axial Power Imbalance Protective Limits

During power operation of the reactor core, General Design Criterion (GDC) 10 (reference 3a) requires that the fuel not sustain damage as a result of normal operation or anticipated operational occurrences. Therefore, the reactor fuel and cladding must be designed and operated with appropriate thermal margin to assure that specified acceptable fuel design limits are not exceeded. In the design of reload cores it is assumed that immediate fuel damage will result if

1. Centerline Fuel Melting (CFM) occurs,
2. A Transient Cladding Strain (TCS) in excess of 1% occurs, or
3. Steady-state Departure from Nucleate Boiling Ratio (DNBR) limits are violated.

For normal operation and anticipated operational occurrences, fuel and cladding protection are provided by an automatic RPS trip function when the ratio of thermal power to reactor coolant flow reaches specified limits, or when axial power imbalance reaches limits specified during the reload safety evaluation. The RPS relies on the global quantities of thermal power level and axial power imbalance to provide the required trip function.

The trip on thermal power to reactor coolant flow (power-to-flow trip) is required to ensure that the DNB limiting criterion is not violated in steady-state with four reactor coolant pump or partial pump operation. The maximum power levels allowed by the trip setpoints are based on a power-to-flow ratio that has been established to accommodate flow-decreasing transients from high power levels. The power level trip setpoint produced by the power-to-flow ratio provides DNB protection from both high power level and low reactor coolant flow for all modes of reactor coolant pump operation. The flow-dependent portion of this trip function is established for design peaking conditions with zero imbalance.

The trip on design overpower (high flux trip) is required to ensure that the local LHR will not exceed either the CFM or TCS limiting criteria (this trip function does not vary the trip setpoint with changes in the core power distribution). The trip on axial power imbalance ensures that the CFM, TCS or DNB limiting criteria are not violated, taking into account variations in the axial power distribution that may occur as a result of core design features, fuel depletion, xenon distribution, and control component positioning.

5.3.1.1 Acceptance Criteria

To prevent cladding failure, the following safety criteria have been established:

1. The maximum local LHR anywhere in the core must be limited so that CFM does not occur.
2. The maximum local LHR anywhere in the core must be limited so that TCS is maintained below 1%.
3. There must be at least a 95% probability at a 95% confidence level that the hot fuel rod in the core does not experience a DNB condition.

The third criterion is referred to hereafter as the 95/95 DNB criterion. The minimum DNBR value during both normal operation and anticipated transients is limited to the DNBR analysis limit for the particular fuel design in use. The DNBR analysis limit meets the 95/95 DNB criterion and provides an appropriate margin to DNB for all operating conditions. Sections 6.2.8 and 6.2.9 contain a further discussion of DNBR analysis limits.

These acceptance criteria are met by constraining power operation within the RPS axial power imbalance protective limits during normal operation and anticipated operational occurrences. Operation within the RPS axial power imbalance protective limits ensures the maximum allowable LHR based on the CFM, TCS, and DNB peaking criteria will not be exceeded. The following analysis criteria must be preserved by the power distribution analysis:

1. The combination of thermal power and axial power imbalance may not produce a local LHR in excess of the LHR to cause CFM. Section 4.2.9 provides a further discussion of the CFM limit.

2. The combination of thermal power and axial power imbalance may not produce a local LHR in excess of the LHR to cause TCS to exceed 1%. Section 4.2.6 provides a further discussion of the TCS limits.

3. The combination of thermal power and axial power imbalance shall not produce local peaking in excess of the maximum allowable peaking (MAP) limits based on the DNB analysis. Section 6.6 provides a discussion of MAP limits.

5.3.1.2 Technical Specification Limits

The reactor trip based on the axial power imbalance protective limits prevents the maximum LHR, or the maximum local peaking factor, from causing a violation of the thermal design bases during normal operation and anticipated occurrences. Example axial power imbalance protective limits are shown in Figure 5-1. These protective limits must be error-adjusted to account for measurement system observability and equipment uncertainties. The method used to perform the error adjustment is described in Section 7.3. The axial imbalance protective limits and error-adjusted trip setpoints are specified in the COLR.

5.3.1.3 Analysis Methods

5.3.2 Power Distribution-Related LCO Limits

This section addresses the methodology for determining the LCOs related to core power distribution. Limits on the following parameters and process variables are addressed:

- Regulating rod insertion, group sequence, and group overlap
- APSR insertion
- Axial power imbalance
- Quadrant power tilt

The LCOs imposed on control component operation and on monitored process variables ensure that core peaking is maintained within the nuclear heat flux hot channel factor, F_Q^N (or LOCA LHR limits), and the nuclear enthalpy rise hot channel factor, $F_{\Delta H}^N$ (or initial condition DNB peaking limits). The F_Q^N and $F_{\Delta H}^N$ limits may be expressed either in dimensionless peaking units or in LHR units. Operation within the F_Q^N limits given in the COLR prevents power peaks that would exceed the LOCA LHR limits derived by the ECCS analysis described in Section 9. Operation within the $F_{\Delta H}^N$ limits given in the COLR prevents DNB during a loss of forced reactor coolant flow event.

In addition to the F_Q^N and $F_{\Delta H}^N$ limits, certain reactivity limits are preserved by regulating RILs. The regulating RILs restrict the ejected control rod worth to the values assumed in the safety analyses and preserve the minimum required SDM. The reactivity related LCOs are described further in Section 5.4.2. Regulating rod position is measured by using the rod index, defined as the sum of the positions of groups 5, 6, and 7 in percentage withdrawn. Figure 5-3 illustrates the relationship of rod index to regulating bank position. The regulating rod groups operate with a predetermined amount of position overlap in order to approximate a linear relation between rod worth and rod position (integral rod worth). The regulating rod groups are withdrawn and operate in a predetermined sequence. The automatic control system limits reactivity by moving the regulating rod groups in their specified sequence within analyzed ranges.

5.3.2.1 Consequences of Exceeding Limit

The simultaneous occurrence of operation in violation of one or more of these LCOs with the postulated accident could result in (1) exceeding the maximum peak cladding temperature (PCT) of 2200 ^{o}F, (2) a DNBR below the analysis limit, (3) an ejected rod worth greater than the values assumed in the safety analyses or (4) less than the minimum SDM specified in the Technical Specifications (typically 1% $\Delta k/k$). The concern in exceeding the PCT limit is the failure of the cladding by the resulting Zircaloy-water reaction, which could become self-sustaining due to the extreme heat. The concern in exceeding the DNBR limitation is cladding failure due to overheating, which could occur if film boiling prevents efficient heat transfer to the reactor coolant. The concern in exceeding the ejected rod worth limit is fuel failure due to either DNB or fragmentation of the fuel and cladding if an ejected rod resulted in a fission energy release of more than 280 cal/g. The concern in exceeding the SDM limit is failure to shut down the reactor upon an RPS trip, or that the consequences of an accident may be greater than that determined by the safety analyses. In addition, exceeding any of these limits could require significant analyses and/or inspection to justify continued operation of the unit.

5.3.2.2 Acceptance Criteria

The reload safety evaluation methodology must demonstrate that a reasonable probability exists that the fuel design limits can be met and will not be exceeded for normal operation and anticipated operational occurrences over the fuel cycle length. The regulating rod insertion, APSR insertion, axial power imbalance, and quadrant power tilt LCOs preclude core power distributions from occurring that would violate the following fuel design criteria:

1. During a LOCA, the PCT must not exceed a limit of 2200 °F (10CFR50.46, reference 3b).
2. During a loss of forced reactor coolant flow accident, there must be at least a 95% probability at a 95% confidence level that the hot fuel rod in the core does not experience a departure from nucleate boiling (DNB) condition (GDC 10).
3. During an ejected rod accident, the fission energy input to the fuel must not exceed 280 cal/g (GDC 28, reference 3c).
4. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (GDC 26, reference 3d).

These acceptance criteria preserve the accident initial condition assumptions in the safety analyses related to the core power distribution and reactivity. The regulating RILs, APSR insertion limits, axial power imbalance limits, and quadrant power tilt limits specified in the COLR are determined from power peaking and reactivity limits based on these criteria. These LCOs limit the amount of fuel cladding damage during a postulated accident by preserving these initial condition acceptance criteria. These LCO limits must meet the following analysis criteria:

1. The combination of thermal power, axial power imbalance, and regulating rod insertion, including the steady-state quadrant power tilt allowance, must not cause power peaking that would exceed the allowable nuclear heat flux hot channel factor (F_Q^N) or the nuclear enthalpy rise hot channel factor ($F_{\Delta H}^N$) limits.

2. The combination of thermal power and regulating rod insertion must not allow the worth of an ejected rod to exceed the values demonstrated to be acceptable in the safety analyses.

3. The combination of thermal power and regulating rod insertion must not allow the minimum SDM to be less than that specified in the Technical Specifications (e.g., 1% $\Delta k/k$) at hot zero power (HZP), equilibrium xenon with the maximum worth control rod stuck fully withdrawn.

5.3.2.3 Technical Specification Limits

A requirement of 10CFR50.36 (reference 3e) is that LCOs be placed on process variables required for safe operation of the plant. Regulating rod position (rod index), APSR position, axial power imbalance, and quadrant power tilt are process variables that together characterize and control the three-dimensional power distribution of the reactor core. Therefore, administrative limits are established for these variables during the reload safety evaluation, and they are monitored and controlled during power operation to ensure that the power distribution remains within the bounds set by the licensing analysis. If the LCO limits based on power peaking are violated, a short time is allowed for corrective action. However, if the LCO limits are not restored within a reasonably short time, a significant power reduction is required. Operation beyond the SDM RIL is treated differently because it is of common importance to the entire safety analysis. Therefore, immediate action to restore the SDM is required should the SDM RIL be violated.

5.3.2.4 Analysis Methods

5.3.3 EOC Full Power Extension Maneuvers

Fuel cycle designs may include provisions for end-of-cycle (EOC) extension maneuvers to allow for continuing operation at rated thermal power at the end of a fuel cycle. EOC extension maneuvers typically include an EOC T_{avg} reduction and may be coupled with withdrawal of the APSRs. The effects of EOC T_{avg} reductions on the core power distribution have been evaluated to determine their impact on power peaking factors, and allowable values for the T_{avg} reduction have been defined. If the capability for an EOC T_{avg} reduction is designed into the reload core fuel cycle, the core protective and operating limits will be analyzed and set to accommodate its impact on margins to the core peaking limits, i.e., the protective and operating limits specified for the COLR will accommodate the EOC extension maneuver.

5.3.4 Xenon Stability Index

The APSRs are positioned to provide both positive and negative imbalance control to compensate for shifts in the axial power distribution during transient conditions. Utilization of the APSRs by the operator prevents or damps axial xenon oscillations, and controls transient imbalance during power level maneuvers. The APSRs are controlled manually by the operator and do not trip. When the APSRs are withdrawn, power oscillations can be damped by the regulating rods. The reactivity worth of the APSR group is lower than that of the regulating rod group (typically by a factor of approximately five). Therefore, axial power oscillations may be damped easily by manual positioning of the APSRs with little change in core reactivity.

5.3.4.1 Acceptance Criteria

APSRs were incorporated in the design of the nuclear steam supply system (NSSS) to prevent or damp axial power oscillations caused by xenon oscillations. The safety criterion addressed by APSRs is from GDC 12 (Suppression of Reactor Power Oscillations, reference 3f):

> The reactor core and associated coolant systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

In reload core designs, the APSRs are inserted in the core for use in controlling the axial power distribution for up to 90% (typically) of the fuel cycle length. Then the APSRs may be withdrawn fully to remove their negative reactivity, so that the cycle may operate to a longer design length for a given fuel loading. For operation after APSR withdrawal, the following analysis criterion is applied in the power distribution analysis:

> Axial power oscillations induced by an axial xenon oscillation shall be naturally damped.

If this criterion is not met, however, power restrictions are not applied. Instead, this result would be noted in the safety evaluation, and the regulating rods would be used to damp any induced oscillations, according to station procedures.

5.3.4.2 Analysis Methods

5.3.5 Power Level Cutoff Hold Removal

This section is only applicable to those units with a technical specification power level cutoff hold requirement.

The effects of peaking due to transient xenon are explicitly included in the power distribution analysis by direct simulation of limiting power distributions from cycle-specific xenon transients. The simulation of xenon transients as described in reference 36 is further augmented by APSR motion in a manner to generate conservative (higher) peaking factors.

Since all LCO limits are explicitly based on transient xenon data, a power level cutoff hold is not required and the power level cutoff hold value may be set to 100% of RTP.

5.3.6 Overcooling Transient

Overcooling events cause a reduction in reactor coolant inlet temperature. The potential temperature-induced measurement error in the indicated excore neutron power (utilized by the RPS) for some overcooling transients may exceed the value assumed in the safety analyses. Such a condition could cause the actual thermal power level to exceed design overpower without causing a reactor trip, resulting in the potential for CFM, TCS or DNB safety limits to be violated.

5.3.6.1 Acceptance Criteria

The applicable safety criteria are prevention of violating CFM, TCS, and steady-state DNB safety limits, as specified in Section 5.3.1.1. To ensure that the power distribution limits will preserve the safety criteria, the following analysis criterion is applied in the power distribution analysis:

> CFM, TCS, and steady-state DNB peaking margins from power distributions that simulate the overcooling transient shall not violate the CFM, TCS, and steady-state DNB safety criteria.

5.3.6.2 Analysis Methods

5.3.7 Inoperable Control Rods

Inoperable control rod assemblies (CRAs) comprise dropped or misaligned CRAs. These events potentially result in increased power peaking factors. Since inoperable control rods are moderate frequency events, the RPS must prevent the core power distribution from exceeding the core safety limits during those events.

5.3.7.1 Acceptance Criteria

The applicable safety criteria are prevention of violating CFM, TCS, and steady-state DNB safety limits, as specified in Section 5.3.1.1. To ensure that the power distribution limits will preserve the safety criteria, the following analysis criterion is applied in the power distribution analysis:

> CFM, TCS, and steady-state DNB peaking margins from power distributions that simulate single inoperable CRA events shall not violate the CFM, TCS, and steady-state DNB safety criteria.

5.3.7.2 Analysis Methods

5.4 Nuclear Parameters For Safety Analysis

This section describes the methodology used to determine the nuclear parameters that ensure that the safety analyses for the reactor remain valid for the reload cycle. The safety analyses refer to calculations and evaluations performed for the Accident Analysis chapter in the Final Safety Analysis Report (FSAR) and any updates or revisions, hereafter referred to as the Safety Analysis Report (SAR). The methodology supports reload safety evaluations for all Mark-B fuel.

The description of the methodology is divided into four parts, Sections 5.4.1 through 5.4.4. This division is related to the organization of the reload report and the safety evaluations performed for the reload. Section 5.4.1 reviews the procedures for determining the key transient neutronic parameters impacting the safety analyses. This leads into Section 5.4.2, which presents the reactivity related LCOs. Section 5.4.3 describes how radiation parameters, which are important in the source term evaluation, are modeled. Section 5.4.4 addresses the analytical methods employed in developing the actual fuel loading pattern.

Reference 23 addresses the nuclear aspects that are affected by the repair of fuel assemblies with stainless steel replacement rods.

5.4.1 Transient Neutronic Parameters

The safety analysis evaluations performed for the accidents in the SAR assumed values for the neutronic parameters related to the reactivity coefficients and worths, control system reactivity, kinetics, and transient power peaking. When the accidents were analyzed, parametric cases were evaluated to determine the combination of conditions and the key neutronic parameters that would produce the most severe transients. The resulting limiting conditions, and bounding values of the key neutronic parameters, are the ones that must be analyzed for each reload cycle. The methodology used for the reload analysis must consequently ensure that the calculations of the key neutronic parameters are performed with the appropriate limiting conditions. The results are then reviewed using the methods described in Section 8 to ensure that the values are bounded by those used in the safety analyses. If the reload calculations contain the proper conditions, and the results are within the bounds of the reference safety analysis, then the SAR results will continue to be applicable to each respective reload cycle.

5.4.1.1 Acceptance Criteria

1. The key neutronic parameters shall be determined using the appropriate limiting conditions. The key neutronic parameters are then reviewed using the methods described in Section 8 to ensure that the values are bounded by those used in the safety analysis.

2. Design changes shall be reviewed with respect to the accident scenarios in the safety analyses to ensure that the change does not affect the scenario for defining the appropriate limiting conditions for calculating the neutronic parameters.

5.4.1.2 Analysis Methods

5.4.2 Reactivity-Related LCOs

The reactivity-related LCOs are those associated with the control systems. The LCOs are divided into two categories. The first category, and the one that is more complex, is that dealing with the SDM. The second category deals with the ejected rod. The methodologies used to analyze the SDM and the ejected rod are very similar. However, to clarify the discussion, the methodology for analyzing each will be described separately. The SDM discussion is further divided into two areas. The first is the control rod system and the second is the boron injection system.

The methodology for determining the reactivity-related LCOs is based on the control system requirements for specific reactivity margins. These requirements are part of the bounding conditions assumed in the safety analyses for the accidents in the SAR. In order to ensure that the required safety margins are always applicable for the various operating modes of the reactor, many control system requirements have been incorporated into the technical specifications.

5.4.2.1 Shutdown Margin

The accident analyses chapter in the SAR assumes that following a reactor trip the reactor will have a minimum amount of SDM. In fact, whenever the reactor is shutdown, the accident analyses contain the assumption that prior to criticality in the startup operational mode, the reactor will have a minimum SDM. The control rod system and the boron injection system are the two independent systems that ensure that the SDM can always be met. RILs and boron concentration requirements are determined each cycle to preserve the minimum SDM assumption in the safety analyses.

5.4.2.1.1 Acceptance Criteria

1. Limits on the allowed rod index versus power level will be determined each reload to ensure that the SDM obtained with the worth of the scrammable control rods is equal to or greater than that assumed in the safety analyses (e.g., 1% $\Delta k/k$ with the most reactive rod stuck out of the core), including uncertainties.

2. Boron concentration requirements as a function of temperature will be determined for each reload to ensure that the SDM obtained using the boron injection system is equal to or greater than that assumed in the safety analyses (e.g., 1% $\Delta k/k$ with the most reactive rod stuck out of the core), including uncertainties.

3. Design changes that affect the control rods, boron worth, or boron injection system shall be reviewed or analyzed to ensure that they are within the bounds of the safety analyses.

5.4.2.1.2 Analysis Methods

5.4.2.2 Ejected Rod Worth

The methodology for evaluating the ejected rod worth is based on the rod ejection accident presented in the accident analyses chapter in the SAR. The LCOs are based on the safety requirements that establish bounding parameters that limit the consequences of the accident.

5.4.2.2.1 Acceptance Criteria

Rod insertion limits versus power level will be determined each reload to ensure that the ejected rod worths are equal to or less than the values demonstrated to be acceptable in the safety analyses, including uncertainties.

5.4.2.2.2 Analysis Methods

5.4.3 Radiation Parameters

Of the results determined for each of the accidents analyzed in the SAR, the dose consequences are the most important. If there is no increase in the environmental doses from radiation sources following an accident, then the consequences of the accident are of minor importance to the health and safety of the public. The methodology for evaluating the nuclear parameters relating to development of radiation sources is discussed in this section.

5.4.3.1 Acceptance Criteria

The radiation sources are primarily fission products produced in the fuel during power operation. Other sources include materials other than the fuel that are irradiated during operation and become radioactive. Fission products and other radiation sources are not specifically a part of the nuclear parameters evaluated in the nuclear design analysis (Section 5). However, the sources are analyzed and evaluated for each reload as described in Section 8. The accident analysis review assesses the maximum possible doses from the various accidents for each reload by determining the sources from the nuclear parameters. These parameters are those such as burnup, fluences, uranium, and plutonium fissions, etc. Therefore, the methodology relates to calculation of these types of parameters. The acceptance criteria for the 50.59 licensing procedure integrate the nuclear design methodology with the accident analysis review.

1. The methodology and applicable uncertainties for analyzing the nuclear parameters that lead to the radiation sources shall not be changed without verifying the changes are within acceptance criterion 8 of Section 8.2.

2. Design changes that could affect radioactive sources shall be reviewed and analyzed.

5.4.3.2 Analysis Methods

5.4.4 Fuel Loading

Loading the fuel into the core for the reload cycle is simulated during the fuel cycle design process by assuming all fresh fuel is uniformly loaded. In addition, the burned fuel is assumed to have been uniformly loaded initially and to have burned uniformly in the quarter-core. However, during power operation, the fuel may not accumulate exactly symmetric burnup because (1) the manufacturing process creates non-uniformities in the fuel and burnable poison loadings and (2) non-uniformities in the operational characteristics of the core create tilts that lead to non-uniform burnup.

5.4.4.1 Acceptance Criteria

Fuel, burnable poison or other components that could induce a radial power asymmetry shall either be loaded into the core to ensure that there is no impact on the power distribution or specifically analyzed.

5.4.4.2 Analysis Methods

5.5 Startup Physics Testing

The purpose of the design analyses of the reload cycle is to ensure that the reference safety analyses remain applicable. The nuclear design analyses are based on modeling the core characteristics using the methods, procedures, and computer calculations described in Sections 5.1, 5.2, 5.3, and 5.4 of this report. The results of the design analyses show that bounding peaking distributions and bounding nuclear parameters are within the criteria required by the safety analyses. However, there remains an uncertainty related to the accuracy of the design calculations and modeling of the reload cycle characteristics relative to actual measurements. Reload startup physics testing is performed following refueling outages to verify that the core is operating as designed.

The previous cycle design predictions are benchmarked to startup test measurements, and core-follow calculations of the power distributions are benchmarked to measured data. The previous cycle is the reference cycle for the reload core design. If there are no design changes or changes to the manufacturing specifications, then the conclusion could be reached that the design calculations are completely satisfactory to ensure that the safety parameters have been accurately analyzed. This conclusion is further supported by the topical reports on the computer codes, methods and procedures, and uncertainties, which have shown that the design analyses are sufficiently accurate.

However, prudence suggests that some amount of startup physics testing is important to ensure that the safety evaluations are valid. A small probability exists that the calculations will have larger-than-expected deviations simply because the calculational accuracy was established statistically. A small probability also exists that loading or manufacturing deviations may occur. Thus, a startup testing program is part of the reload evaluation process for the nuclear analysis.

5.5.1 Acceptance Criteria

The previous subsections in this nuclear design section have discussed the methodology for performing design analyses to ensure that the characteristics of a reload cycle are bounded by the reference safety analyses. The methodology referenced the calculational codes, models, and procedures that are used to determine the nuclear parameters. The same calculational codes, models, and procedures must be revalidated during the startup of each reload cycle by performing a minimum amount of startup physics tests which compare the resulting measured values to calculational predictions. Design calculations, using the calculational codes, models, and procedures that were used to verify that the nuclear parameters are bounded by the reference safety analyses, shall model startup conditions to produce predictions that can be compared to measurements.

Startup testing requirements should meet the requirements of ANSI/ANS 19.6.1 (reference 42).

The standard startup physics testing scope for B&W-designed plants complies with ANSI/ANS 19.6.1 (reference 42) with the following exceptions:

1. Reference 42 specifies that if the boron dilution method for determining HZP measured rod worth is employed, then measurement of all control rod groups, or at least 3000 pcm is required. FANP has justified a ZPPT program that includes only measurement of CRG 7 (partial – at least 80% of the worth of CRG 7 is measured) and CRG 6. This is typically at least 1500 pcm.

2. Reference 42 suggests (the appropriate specification is contained in the Appendix, which is technically not part of the Standard) that the endpoint worth for CRG 7 is measured for the boron equivalent correction to the measured all rods out critical boron concentration (AROCBC). FANP has justified that up to 200 pcm predicted worth can be used for this correction.

3. Reference 42 requires a measured differential boron worth and application of a test criterion to a comparison of measured to predicted values. FANP has developed a modified differential boron worth measurement technique not included in the Appendix of reference 42. Rather than eliminate the measurement of differential boron worth entirely, this new technique is employed with the results as information only (no test criterion is applied).

These exceptions are discussed in the NRC-approved topical report on ZPPT modifications for B&W-designed reactors (reference 81). The current minimum scope of reload startup physics testing for B&W-designed plants is contained in Table 5-2.

5.5.2 Analysis Methods

Table 5-1

Example Calculation of the Shutdown Margin

[ ]

Table 5-2

Reload Startup Physics Testing for B&W-Designed Plants

Test	Acceptance Criterion	Notes
HZP AROCBC	$\pm$ 50 ppm -- Acceptance (Predicted – Measured) $\pm$ 45 ppm -- Review*	Up to 200 pcm predicted worth of CRG 7 allowed for endpoint correction. * Only applied if predicted worth used.
Isothermal Temperature Coefficient (ITC)	$\pm$ 2 pcm/^{o}F (Predicted – Measured)	
Moderator Temperature Coefficient	< Tech Spec Limit	Measured MTC inferred from measured ITC by application of predicted Doppler coefficient.
Individual CRG Worths	$\pm$ 15% % dev = {(P–M) / P} * 100%	At least 80% of CRG 7 and all of CRG 6. (Consistent with reference 81)
Total CRG Worth	$\pm$ X % % dev = {(P–M) / P} * 100%	X = SDM related uncertainty on rod worth - always between 5-10%, depending on fuel cycle.
Differential Boron Worth	No criterion applied	Ratio of measured rod worth to measured boron differences during CRG worth measurements.
Flux Symmetry Test	| Tilt | < full power limit Symmetric incore detector readings within $\pm$ 10 %	Both of these criteria are considered "review criteria". Evaluation should be accomplished before physics testing is performed at a higher power level.
Intermediate Power Level Core Power Distribution	Several specific acceptance criteria apply, including the criteria in reference 42	Between 40-80 %FP.
HFP AROCBC	$\pm$ 50 ppm	Difference between the HZP AROCBC (P – M) Δ and the HFP AROCBC (P – M) Δ.
HFP Core Power Distribution	Several specific acceptance criteria apply, including the criteria in reference 42	Between 90-100 %FP.

Figure 5-1

Example Axial Power Imbalance Protective Limits

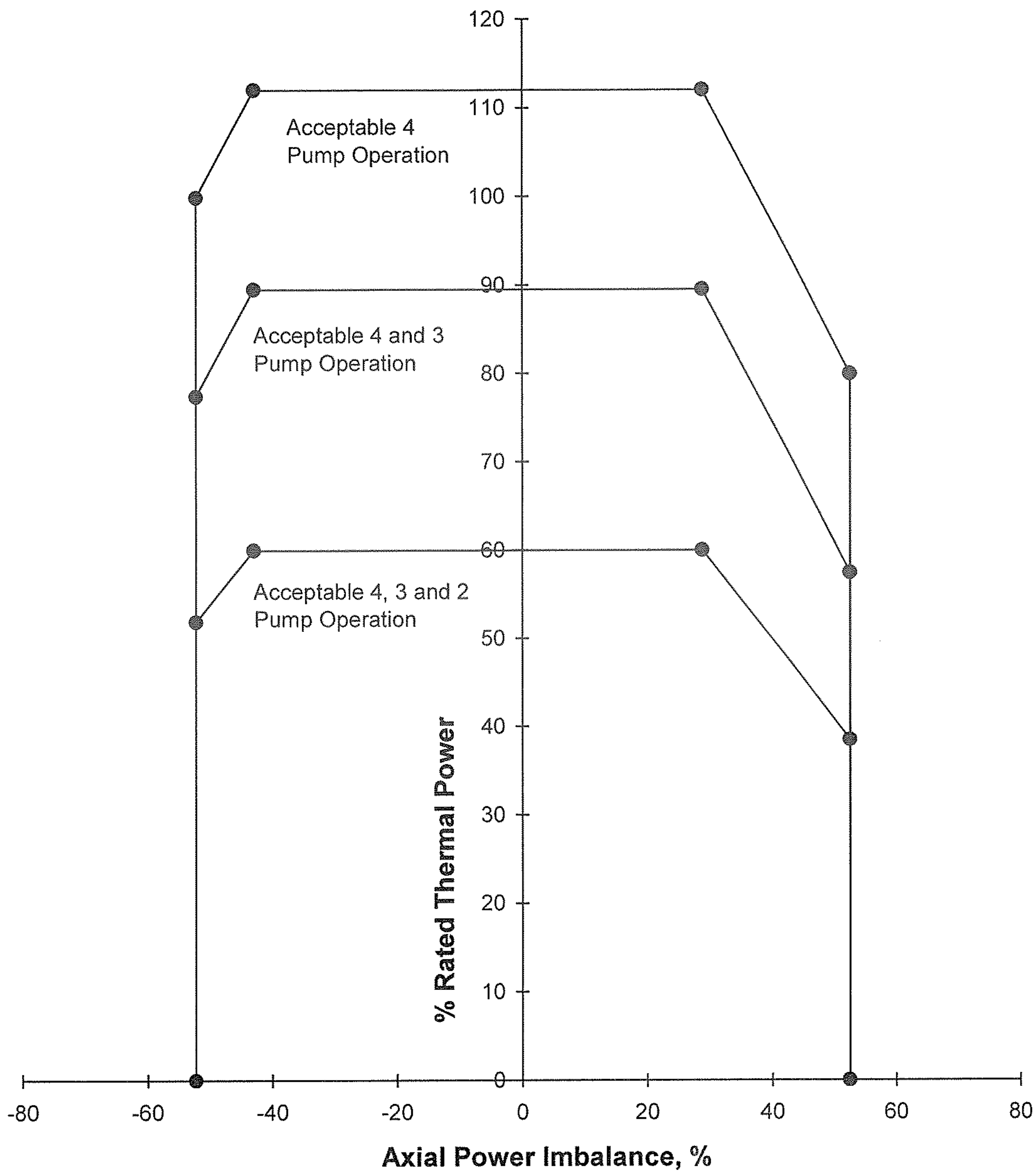


Figure 5-2

Example Centerline Fuel Melt vs. Offset Correlation

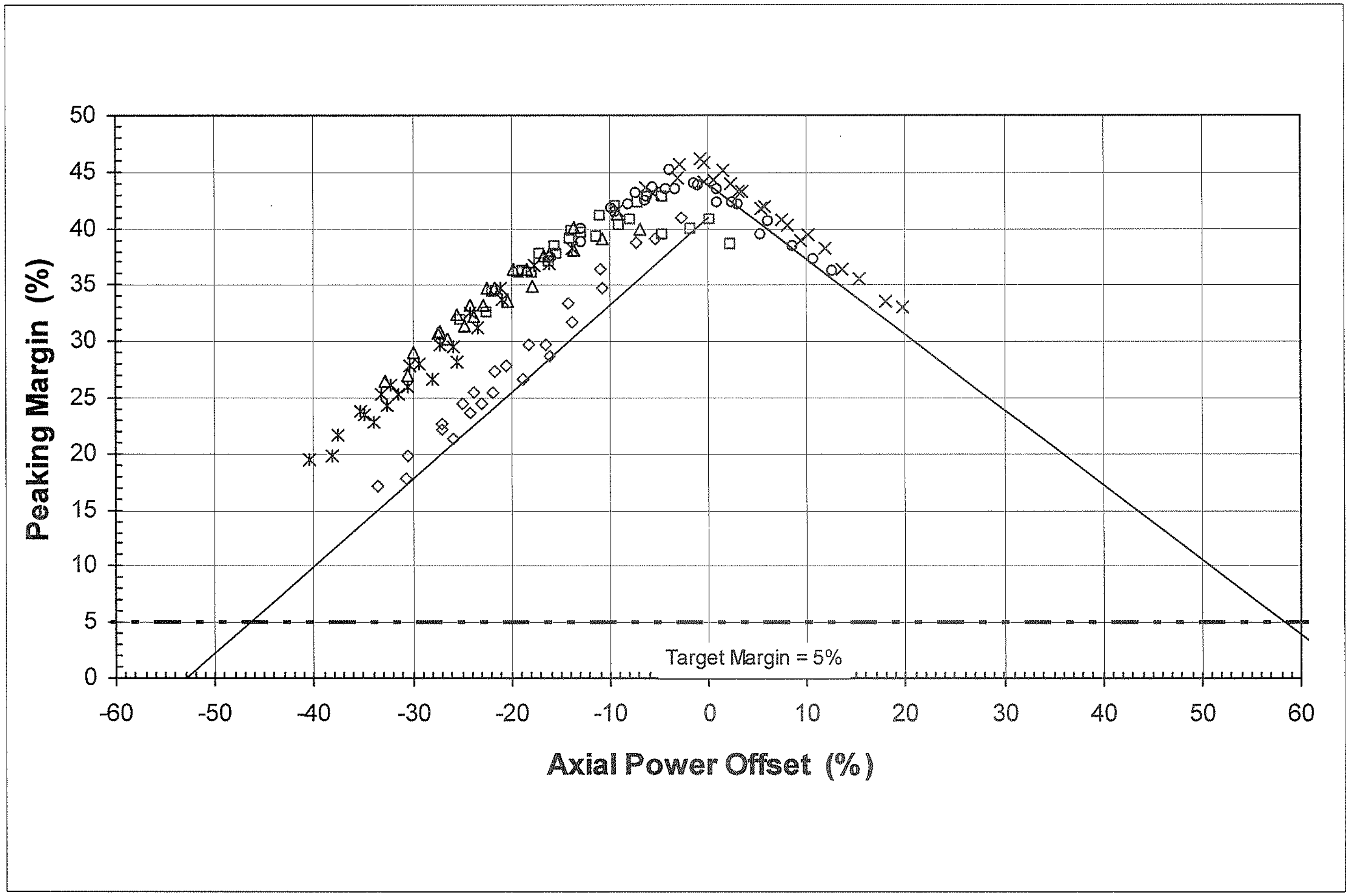
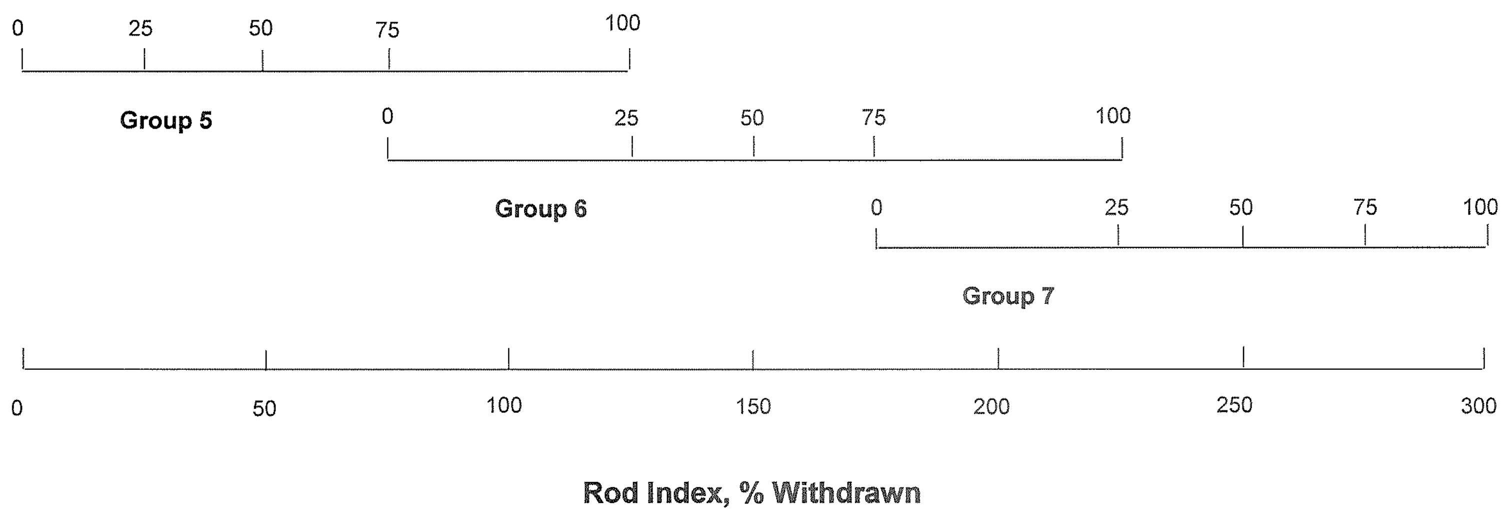


Figure 5-3

Rod Index and Regulating Bank Position Relationship

(Typical 25% Overlap Shown)



SECTION 6 CORE THERMAL HYDRAULICS

6.1 Design Criteria

The DNBR safety criterion is that there shall be at least a 95% probability at a 95% confidence level that the hot fuel rod in the core does not experience a departure from nucleate boiling during normal operation or events of moderate frequency (Condition I or II events). The approach used to ensure that this criterion is met is to apply a corresponding design criterion, which is that the minimum predicted DNBR, using an appropriate critical heat flux (CHF) correlation and computer code, that must be greater than or equal to the DNB design limit value for that correlation. The DNB design limit for each correlation is the DNBR value for which there is a 95% probability at the 95% confidence level that DNB will not occur. Therefore, an alternate protection statement is that 95% of the rods at the DNBR design limit will not experience DNB, with 95% confidence. An appropriate CHF correlation is defined as one that has been approved by the NRC for application to the fuel design type being analyzed. Appropriate computer codes include those approved for use with the CHF correlation being applied. For the B&W 177-FA plants, the approved CHF correlations include B&W-2, BWC, BHTP, and the BWU series (references 44, 45, 4, and 46). The B&W-2 and BWC correlations have been approved for use in the TEMP, LYNX1/LYNX2, and LYNXT (references 47, 48, 49 and 50) codes. Typical DNB design limits are 1.30 for B&W-2 and 1.18 for BWC. The BHTP and BWU series CHF correlations have been approved for use in the LYNXT code. The BWU-N CHF correlation, in the BWU series, is limited to fuel assemblies that have no mixing vane spacer grids (reference 46). The BWU-Z correlation, in the BWU series, is limited to the Mark-B fuel design utilizing mixing vane spacer grids, currently known as the Mark-B11 (references 51 and 52). The BHTP correlation is limited to fuel assemblies utilizing HTP spacer grids (reference 4).

The second criterion addressed in the thermal-hydraulic analyses is that the worst case hydraulic loads should not exceed the hold down capability of the fuel assembly (gravity plus hold down spring capability) during normal operation. This criterion is also discussed in Section 4.1.3.

6.2 Analysis Methods

This section describes the codes and general methods used for core hydraulics and thermal-hydraulics analyses. Analysis techniques associated with specific calculations are discussed in the following sections.

6.2.1 LYNX Codes

The LYNX code series is composed of three codes, i.e., LYNX1, LYNX2, and LYNXT, all of which have received NRC approval (topical reports BAW-10129-A, BAW-10130-A, and BAW-10156P-A, respectively). These codes calculate fluid conditions and CHF for normal and abnormal operating conditions. Mass and turbulent energy interchange between channels are calculated for both steady-state and transient analysis. Steady-state analyses are performed either with LYNX1 and LYNX2, used in tandem, for a two-pass simulation of the reactor core, or with a single-pass LYNXT model. For transient analysis, the LYNXT code calculates fluid conditions, CHF, cladding temperature, fuel pin pressure, and fuel temperature as a function of time.

Each of the LYNX codes has been verified by benchmarking to appropriate test data and to other thermal-hydraulic codes. The three codes have a common set of fluid property relations and use the same void and heat transfer correlations, thus promoting consistency among themselves. LYNX1 has been benchmarked to experimental crossflow data, to incore temperature distribution measurements from Oconee 1 and Zion 1, and to other codes, with excellent results. LYNX2 has been compared with experimental diversion crossflow tests, with experimental CHF data, and with other codes, again with excellent results. LYNXT has been compared to the experimental data used in the LYNX1 and LYNX2 benchmarking as well as to other CHF data. Since the prediction of CHF is dependent on both the thermal-hydraulic code and the CHF correlation performance, LYNX2 and LYNXT have been qualified independently for each CHF correlation by analysis of its database; in each case where this evaluation has been performed, both codes support the licensed DNBR limit. In addition, since LYNXT has transient analysis capability, this code has been benchmarked to transient CHF data. The transient analysis demonstrated that the code, with the B&W correlation form, provides a conservative predictor of transient CHF data.

Typical models used with the LYNX codes for core thermal-hydraulic analysis of the B&W 177-FA plants are described in reference 53.

6.2.1.1 LYNX1

The LYNX1 code performs the steady-state thermal and hydraulic analysis of the reactor core coolant on an assembly basis by solving a set of one-dimensional conservation equations for mass, momentum, and energy. Each individual fuel assembly is taken as a unit in the reactor cross-sectional plane, with the core being divided into a finite number of axial increments. Using an iterative solution method, the program can determine the inlet pressure profile of the reactor core when given an inlet velocity profile. The required boundary condition for the solution is a known exit pressure profile. The program is also capable of determining the inlet velocity profile of the reactor core when given an inlet pressure profile with the same boundary condition. The calculation can be made at full power or at any fraction of the RTP. Inter-assembly diversion crossflow is accounted for in the analysis. LYNX1 is closely related to the subchannel thermal-hydraulic analysis code LYNX2, which takes the calculated inter-assembly crossflow obtained from the LYNX1 solution and performs the thermal-hydraulic analysis on the subchannel basis within an assigned fuel assembly.

LYNX1 is used to predict pressure drop and flow distributions, including diversion crossflow, on a core-wide basis. The core is modeled as a number of parallel, open channels, with each channel representing a single fuel assembly, or a portion of an assembly. A typical model, shown in reference 53, represents 1/4 of the core and extends from below the lower core support plate up to and including the upper grid assembly. Both recoverable and unrecoverable (friction, form-loss) pressure drops are calculated at each axial node. Form-loss coefficients are derived from fuel assembly flow tests in which pressure drop for each component and for the fuel assembly as a whole are measured as a function of flow rate. For DNBR calculations, inter-assembly diversion crossflow at the boundaries of the limiting, or "hot" assembly, are output to the LYNX2 code.

6.2.1.2 LYNX2

The LYNX2 code calculates subchannel conditions by conserving mass, momentum, and energy. Inter-subchannel diversion crossflow is determined from transverse pressure differences. Inter-subchannel turbulent interchange and inter-bundle diversion crossflow are also incorporated in the solution. LYNX2 has evolved from COBRA but has been modified to accept up to 324 sub-channels and 289 fuel rods. For each subchannel, the code allows for the description of geometry, location, area, and gap variations, and hot channel factors. For each rod, the code allows for the description of the geometry, location, radial power factor, and axial flux shape. As many as five engineering hot channel factor sets may be input, and axial heat flux shapes may be defined by using polynomials or section sets, or generated internally. Ten form-loss regions may be modeled using position, length, loss coefficient, and area contraction ratio. Turbulent interchange proportionality factors may be input as a function of quality. CHF ratios may be calculated using any of the current CHF correlations.

LYNX2 is typically used to model an individual fuel assembly, on a subchannel basis, incorporating the pin-by-pin power distribution. LYNX2 accepts the LYNX1-predicted crossflow distribution predicted at the boundaries of the fuel assembly and predicts subchannel mass flow, enthalpy, and DNBR distributions. Figures 2-6 and 2-7 of reference 53 show the subchannel and fuel rod numbering scheme and the pin-by-pin peaking distribution, respectively, for a typical full-bundle LYNX2 model.

6.2.1.3 LYNXT

LYNXT is a versatile thermal-hydraulics crossflow code capable of predicting flow and temperature distributions within the reactor core. The code has the ability to handle a wide range of reactor flow problems, from inter-bundle to inter-subchannel and is ideally suited to one-pass reactor core analyses. In a one-pass analysis, subchannels, groups of subchannels, bundles, and groups of bundles are modeled in one computer simulation. LYNXT is also able to provide an accurate representation of fuel temperature predictions from sophisticated fuel performance codes such as TACO2 (reference 54). The excellent convergence stability and accurate fuel rod modeling of LYNXT permit the analysis of a wide range of reactor transients, from the relatively slow loss-of-coolant-flow (LOCF) accidents to the rapid asymmetric control rod assembly ejection accident. LYNXT, originally based on the COBRAIV-1 code, has the following features not found in COBRAIV-1: a direct solution of the crossflow equation, convergence enhancements to the crossflow equation, a more extensive CHF library, an option for code-generated water thermal properties, and a dynamic gap conductance fuel model.

LYNXT provides the capability for single-pass core thermal-hydraulic analysis for both steady-state and transient conditions. For steady-state performance analyses, LYNXT is typically used in relatively small model configurations (see Figure 2-9 of reference 53, for example) to predict DNBR response to variations in external parameters such as RCS flow, pressure, temperature, and reactor power. The methodology used to develop single-pass LYNXT models has been qualified by the benchmarking of a number of different models to two-pass LYNX1/LYNX2 analyses and is described in BAW-10156P-A. Each new model that is developed is tested by benchmarking either to a more detailed single-pass model that has been compared to LYNX1/LYNX2 or by direct comparison to the LYNX1/LYNX2 calculations. For transients, LYNXT is also used to predict DNBR response, either from state-point analyses, or by simulating the actual core transient. LYNXT includes three different fuel rod models, which provide different levels of sophistication in modeling to accommodate the need for the analysis of different types of transients. Transient inputs can include core inlet flow velocity and temperature (or enthalpy), core exit pressure, and either neutron or thermal power. Both radial and axial power distributions can also be varied during a transient.

The LYNXT topical report BAW-10156P-A (reference 50) incorporates the Pressure-Velocity Implicit Numerical Solution (PV) algorithm. The PV algorithm provides a supplemental solution technique that can be used as an alternative to the original (COBRAIV-1) implicit algorithm. This solution technique is very useful for the analysis of transients that are characterized by low coolant flow rates.

6.2.2 Power Distribution

Reference core DNBR analyses are performed with a design radial power distribution, for which the maximum radial-local peaking factor ($F_{\Delta H}{}^{N}$) increases with decreasing power level. A typical value of $F_{\Delta H}{}^{N}$ versus power is given by

$$F_{\Delta H}{}^{N} = 1.80 * (1 + 0.3 * (1\text{-}P))$$

where P is the *fraction of RTP.*

The design axial power distribution is typically a 1.65 symmetric chopped cosine shape.

6.2.3 Engineering Hot Channel Factors

Engineering hot channel factors are used to account for the effects of manufacturing variations on DNBR *predictions. In addition*, the local heat flux hot channel factor is used to account for variations in local LHR. The individual factors are discussed below.

6.2.3.1 Local Heat Flux Factor

The local heat flux factor, F_q", accounts for the effects of variations in fuel pellet U^{235} content and is used to evaluate the maximum LHR. This factor is determined by statistically combining manufacturing tolerances and/or as-built data for the fuel pellet enrichment and weight. [

] As discussed in references 55 and 56,

no DNB penalty need be taken for relatively small local heat flux spikes, therefore this factor is not required for DNBR calculations.

6.2.3.2 Statistical Hot Channel Factor on Average Pin Power

The statistical hot channel factor on average pin power, F_q, accounts for the effects of manufacturing variations on the total U^{235} per fuel rod. This factor is determined by statistically combining manufacturing tolerances and/or as-built variations in average enrichment and fuel stack weight. [

] F_q is incorporated directly in DNBR analyses as a multiplier on hot pin heat generation rate.

6.2.3.3 Flow Area Reduction Factor

The flow area reduction factor, F_A, accounts for the effects of variations in fuel rod pitch and diameter. The factor is developed by statistically combining the effects of variations in pitch and diameter on subchannel flow area. [

] Although this factor includes axial variation effects, F_A is conservatively applied in DNBR calculations as a flow area reduction over the entire length of the affected subchannel.

6.2.4 Reactor Core Coolant Flow

The reactor core coolant flow, or the flow available for heat transfer, is equal to the RCS flow minus the core bypass flow. The core bypass flow fraction has several components that are fixed by the NSSS design (direct inlet to outlet nozzle leakage, vent valve seepage and core barrel leakage) and has two components that are dependent on the fuel design (the shroud gap flow, which is the flow between the outer edges of the peripheral assemblies and the core baffle wall, and the flow through the control rod guide tubes and incore instrument tubes).

6.2.4.1 Reactor Coolant System Flow

For core thermal-hydraulic (DNBR) analyses the minimum RCS flow from technical specifications is used with the maximum bypass flow fraction to produce the minimum core flow used for design purposes.

For core pressure drop calculations, RCS flow and bypass flow are parameterized to span the range from minimum to maximum core flow. The pressure drop is typically reported for the flow conditions used in the core DNBR calculations (minimum flow).

For fuel assembly lift calculations, the maximum RCS flow is used with a minimum bypass flow fraction to produce the maximum core flow.

6.2.4.2 Core Bypass Flow Fraction

The fuel-dependent components (shroud gap and guide tube bypass flow) of the core bypass flow fraction are determined for each fuel design. The guide tube bypass flow fraction typically varies with reload core designs. The core designs for the B&W 177-FA plants do not use plugging devices in the control rod guide tubes of unrodded assemblies. Burnable poison rod assemblies (BPRAs) are used in some assemblies to provide reactivity hold down, and the BPRA rodlets, which are inserted in the control rod guide tubes, function effectively as plugging devices in those assemblies. In assemblies with neither control rods nor BPRAs, the guide tubes are unplugged. The number of BPRAs, and consequently the number of unplugged guide tubes, varies from one cycle to another, depending on nuclear design requirements. The guide tube bypass flow fraction is determined as a function of the number of unplugged guide tubes and defines the total core bypass flow fraction for a given core configuration.

[

] The bypass flow fraction (or the corresponding number of unplugged guide tubes) is one of the significant parameters that must be checked for each reload cycle to determine applicability of the reference analysis.

6.2.5 Core Inlet Flow Distribution

The reactor vessel internals are designed to provide a relatively uniform inlet flow distribution to the core. Two flow regions are defined: the interior and peripheral regions. The interior region assemblies, which are those locations not on the core periphery, have a slightly greater-than-average inlet flow.

Thermal-hydraulic analyses using either the LYNX1/LYNX2 or LYNXT codes assume a uniform inlet flow distribution core-wide [

] The magnitude of the core inlet flow factor is based on a vessel model flow test (VMFT) that was performed on a 1/6 scale model of a reactor vessel and core (reference 57).

Fuel assembly hydraulic lift analyses use the flat inlet flow velocity assumption [

]

6.2.6 Fuel Rod Bowing

The bowing of fuel rods during reactor operation has the potential to affect both local power peaking and the margin to DNB. Prediction of bow magnitudes and effects is performed with the methods of reference 10. Using these methods, no DNBR penalty due to rod bowing is applied to FANP fuel, [

]

6.2.7 Fuel Densification Effects

The phenomenon of in-reactor fuel densification causes an initial shrinkage of UO_2 fuel pellets, which is counter-acted by thermal expansion of the pellets as they are heated up and later overcome by the effects of irradiation-induced swelling. The concerns associated with densification are stack height shrinkage and the effects of inter-pellet gaps caused by the shrinkage.

[ ]

The formation of inter-pellet gaps due to densification and pellet hang-up can lead to the occurrence of power spikes in adjacent fuel rods. This was a concern in the early 70's, based on evidence of gap formation that occurred in unpressurized, low density fuel at R. E. Ginna. Since that time, industry-wide changes in fuel designs have been made to increase the fuel density, reduce the propensity for densification, and prepressurize the rods. These changes have reduced the probability of inter-pellet gap formation significantly. [

] Thus, the occurrence of power spikes resulting from densification-induced gaps need not be considered.

6.2.8 DNB Design and Analysis Limits

In order to demonstrate that the DNBR criterion is met, DNBR calculational results are compared to a DNB design limit value. A DNB design limit is established for each CHF correlation.

DNBR values greater than the DNB design limit ensure operation within the nucleate boiling regime, where the heat transfer coefficients are very large and, consequently, the cladding surface temperature remains close to the coolant saturation temperature. In order to define the limits of the nucleate boiling regime, CHF correlations are developed to predict the upper boundary, known as the critical heat flux, or the departure from nucleate boiling heat flux. For each CHF correlation a DNB design limit is established, based on its applicable data base, such that 95% of the rods at the DNBR design limit will not experience DNB, with 95% confidence.

In the establishment of core safety limits, the criterion used is that the minimum DNBR shall be greater than or equal to the DNB design limit. In order to reserve margin in the safety limits and safety analyses, a DNB analysis limit may be used in place of the design limit. This reserved margin may then be used to offset small penalties, such as transition core effects or differences in core bypass flow (due to changes in the configuration of plugged versus unplugged control rod guide tubes), or to provide core design flexibility. The reserved margin, expressed as a percentage relative to the design limit, is calculated as follows:

$$\text{Reserved Margin (\%)} = \frac{DNB_{\text{analysis limit}} - DNB_{\text{design limit}}}{DNB_{\text{design limit}}} * 100\%$$

6.2.9 Statistical Core Design

The design philosophy for core departure from nucleate boiling protection described in other parts of Section 6 follows a deterministic approach where uncertainties that affect the minimum DNBR are simultaneously assumed to be at their worst-case values. The minimum core DNBR is calculated using compounding of the uncertainties, for comparison with the DNBR design limit associated with the applicable CHF correlation.

A more realistic assessment of core DNB protection, called Statistical Core Design (SCD), has been developed by application of statistical techniques to treat the core state and bundle uncertainties. SCD is a widely accepted method that is utilized to reduce some of the undue conservatism of traditional methods, while still allowing for the traditional compounding of variables not amenable to statistical treatment.

BAW-10187P-A (reference 62) describes the application of SCD methodology to the analysis of B&W 177-FA plants operating with Mark-B fuel. A response surface model was used to obtain an overall uncertainty on the calculated DNBR. The response surface model was based on a full central composite design method in order to reduce uncertainty in the response surface model fit. The uncertainty distribution for each of the applicable variables was subjected to a Monte Carlo propagation analysis to determine an overall statistical DNBR uncertainty, which was used to establish a Statistical Design Limit (SDL). The SDL is higher than the CHF limit upon which it is based, because it contains allowances for all of the propagated uncertainties as well as the uncertainty on the original CHF correlation. When the minimum DNBR is calculated, the variables treated statistically are entered into the LYNXT thermal-hydraulic calculations at their nominal levels. Variables not treated in deriving the SDL continue to be entered at their most adverse allowable levels.

Generic uncertainty allowances included in the SCD methodology for the B&W 177-FA plants are described in reference 62. Plant-specific verification of these allowances or determination of new allowances to be used with this method is performed for each application.

The SDL, defined in reference 62, provides 95% protection at a 95% confidence level against hot pin DNB. The corresponding core-wide protection on a pin-by-pin basis using real peaking distributions is greater than 99.9%. Thus, adequate core DNB protection is assured and quantified.

The SDL approved for the conditions described in reference 62 is [ ] This value is based on Appendix F of that topical report. Appendix F of reference 62 describes determination of an SDL that is conservative for all axial power shapes, including axial power distributions that cause the minimum DNBR to be located at or near the core exit (core exit-limited cases). FANP determined that a hot pin SDL of [ ] bounds all cases, including core exit-limited cases, and provides a limiting hot pin 95% protection at a 95% confidence level against DNB for the uncertainties in Tables 3-5 and 3-6 of reference 62. Plant-specific SDL values are determined using the approved methodology of reference 62 to address changes in uncertainties and fuel designs.

The thermal design Limit (TDL) defined in Appendix F of reference 62 represents a retained DNB margin. The retained margin is available to offset penalties, such as transition core effects, or deviations in uncertainty values from those incorporated in the SDL, or to provide flexibility in the fuel cycle design.

Section 6.6 describes the development and generation of MAP limits. MAP limits provide linkage between the DNBR analyses, which use design peaking distributions, and the core power distribution analysis described in Section 5.3. The MAP limits are used in DNB peaking margin calculations that determine the core protective and operating limits. SCD-based MAP limits are calculated with the methodology described, however, their generation is based on equivalence to the TDL, i.e., the SDL plus retained margin, instead of equivalence to the base CHF correlation limit. The calculation and application of SCD-based MAP limits in licensing evaluations will remain as described in this report. When DNB peaking margins are calculated, specific allowances will continue to be made for those factors not included in the SDL/TDL limit.

6.3 Fuel Assembly Hydraulics

6.3.1 Fuel Assembly Lift

In order to preclude fuel assembly liftoff during normal operation, the maximum predicted uplift force on the most limiting assembly must be demonstrated to be less than the minimum fuel assembly hold down capability during limiting operating conditions.

Fuel assembly lift calculations are typically performed in a reference analysis since the hydraulic lift forces are functions of the fuel assembly design and of reactor characteristics that are not dependent on the design of reload fuel cycles. A reference analysis is performed for each fuel assembly design and incorporates the anticipated mixed core combinations for the introduction of new or modified fuel assembly designs. Deterministic and statistical fuel assembly hold down methodologies are discussed below.

6.3.1.1 Deterministic Fuel Assembly Hold Down Methodology

Significant parameters that are evaluated for each reload core to ensure the applicability of the reference fuel assembly hydraulic analyses include the RCS flow rate, the core bypass flow fraction, and the fuel assembly pressure drop, as characterized by hydraulic form-loss coefficients.

6.3.1.2 Statistical Fuel Assembly Hold Down Methodology

BAW-10243P-A (reference 82) describes a statistical methodology to calculate net fuel assembly hold down force. The statistical methodology employs a probabilistic (Monte-Carlo) propagation of uncertainties to demonstrate that the fuel assembly design provides sufficient net downward force to counteract the vertical hydraulic lift force created by the core flow rate so that the fuel assembly remains in a seated position during normal operation and anticipated transients. The application of the methodology ensures that the worst-case hydraulic loads for normal operation will not exceed the hold down capability of the fuel assembly (either gravity or hold down springs) with a 95/95 percent level of protection and confidence.

The fundamental governing equations for calculating net hold down force using the statistical method are identical to those employed for deterministic methodology; the difference in application is the propagation of uncertainties through the equations. The fundamental equation contains all the axial forces acting of the fuel assembly.

During application of the statistical methodology, statepoints for evaluation are defined that cover a wide range of plant operating conditions at different burnup steps in order to identify the limiting statepoint. When the core is composed of different fuel designs, the limiting fuel assembly for each fuel design is determined for each of the statepoints. Then, the nominal value and uncertainty distribution for each of the variables are quantified and the uncertainties are propagated. A hydraulic evaluation of the core using an NRC-approved thermal-hydraulic code is then prepared using the plant-specific fuel cycle core configuration, including the inlet flow distribution applicable to the plant design of interest to obtain the pressure drop across the various fuel assemblies in the core. Finally, the propagation model is used to determine the net hold down force for each fuel assembly design at each of the statepoints. From these calculated values, the minimum net hold down value with the statistical protection at the 95% level with 95% protection is selected.

6.3.2 Core Pressure Drop

Core pressure drop is not evaluated for reload applications unless significant fuel design changes are introduced. In evaluating potential design changes, the criterion used is that changes in pressure drop shall not adversely affect the ability of the RCS to remove heat from the core.

Average core pressure drop is calculated with CHATA (reference 63), LYNX1, or LYNXT in a reference analysis. A comparison of core pressure drop calculations performed with the CHATA and LYNX1 codes (presented in reference 48) shows that the results are identical when crossflow is minimal. In reference 50 the core average pressure drop predictions for LYNX1 and LYNXT are essentially identical for the typical crossflow situation in core analyses. For the reference analysis a best-estimate calculation is performed for a nominal, 100% RTP, operating condition. Fuel assembly pressure loss characteristics are represented by the use of analytical factors (form-loss coefficients) derived from tests in which pressure drop is measured as a function of flow rate.

6.3.3 Hydraulic Compatibility

When fuel design changes are introduced, hydraulic compatibility with resident fuel must be established. The analysis criteria require that flow redistribution effects be determined such that each of the co-

resident fuel types can be shown to meet the DNBR criterion (Section 6.1) and the requirements of Sections 6.3

6.4 Reactor Coolant System DNB Safety Limits

The RCS DNB safety limits are established to prevent overheating of the fuel and consequent cladding failure that would lead to the release of fission products to the coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is maintained at a level slightly above the coolant saturation temperature. Restriction of operation to within the nucleate boiling regime is accomplished by placing limits on observable (and controllable) parameters, including reactor coolant temperature and pressure, core power, and power peaking, such that the DNBR criterion is met.

6.5 Flux/Flow Protective Limit

In B&W 177-FA plants, protection against LOCF transients is provided by the power/pump status and power/imbalance/flow trip functions. The power/pump status function trips the reactor if power is lost to two or more pumps. This trip function therefore provides protection against a complete loss of coolant flow and for some partial loss of coolant flow events. For those partial LOCF events where the power/pump status trip is not activated, and for steady-state operation with 3 or 2 pumps, protection is provided by the power/imbalance/flow trip function. The flow-dependent portion of this protection limit is derived for zero imbalance, design peaking conditions, and is commonly known as the power/flow or flux/flow limit. The limit value is developed from transient analysis of a pump coastdown event, with the criterion being that the minimum DNBR must be greater than the DNB analysis limit. The flow transient analyzed is either a 1-pump or a 2-pump coastdown, depending on the degree of redundancy of the pump power monitors for a specific reactor. The pump power monitors provide pump status information.

6.6 Maximum Allowable Peaking Limits

6.6.1 Calculation of Safety Limit MAPs (RPS MAPS)

6.6.2 Calculation of Operating Limit MAPs

6.6.3 Verification of Calculated Margins (Physics Check Cases)

6.7 Transient Core Thermal-Hydraulic Analysis

For Condition II events, the minimum DNBR must be greater than the DNB design limit. The reload safety evaluation is performed to determine whether any changes introduced by the reload core affect the validity of the existing (reference) transient analyses. The key parameters reviewed to assess the need for recalculation of the DNB-limited transients are those shown in Table 6-3. Limiting conditions for operation (LCOs) are established for pressure, temperature, and flow to ensure the continuing validity of these analyses during plant operation. Criteria for determining the LCOs on DNB parameters are as follows:

1. The LCO on RCS pressure is established such that during normal plant operation the steady-state core exit pressure will be maintained at a level greater than or equal to the value assumed for DNBR analysis of Condition II transients from full power.

2. The LCO on RCS hot leg temperature is established such that during normal plant operation the steady-state RCS hot leg temperature will not be greater than that corresponding to the initial conditions assumed for DNBR analysis of Condition II transients from full power.

3. The LCO on RCS flow rate is established such that during normal plant operation the steady-state RCS flow rate will not be less than the initial condition value assumed for DNBR analysis of Condition II transients initiated from full power.

Transient core thermal-hydraulic analyses are performed with either a single-pass LYNXT model or the closed-channel RADAR code initialized to a LYNXT initial condition DNBR prediction. For full-power transients, the initial conditions modeled are as described in Table 6-2. Transient inputs to the core thermal-hydraulic calculation typically include RCS flow and pressure, core inlet temperature, and neutron power.

6.8 Stainless Steel Replacement Rod Methodology

BAW-2149-A (reference 23) defines a methodology for the use of stainless steel replacement rods in Framatome ANP fuel designs. Reference 23 addresses the nuclear, thermal-hydraulic, and mechanical aspects of the fuel design that are affected by the use of stainless steel replacement rods.

NRC approval has been obtained for the methodology for the use of as many as ten stainless steel replacement rods within a single fuel assembly. From a DNB perspective, the impact of the stainless steel rods on the peaking of adjacent heat-producing fuel rods is to be explicitly examined on a cycle-by-cycle basis to ensure continued compliance to the $F_{\Delta H}^{N}$ design limit.

The stainless steel replacement rods weigh slightly less than Zircaloy-clad fuel rods, but the effect on fuel assembly weight of up to 10 replacement rods is negligible. Therefore, the use of stainless steel replacement rods has an insignificant effect on fuel assembly hydraulic lift.

Stainless steel replacement rods are designed and analyzed to ensure that there is no adverse impact on fuel assembly performance. The rods are designed to ensure that adequate performance with respect to differential thermal expansion, irradiation growth, seismic-LOCA response, grid relaxation, and fretting due to vibration will be maintained.

The impact of the stainless steel replacement rods on the LOCA evaluation will be considered on a cycle-specific basis as noted in Section 9.4.

6.9 Reload Safety Evaluation

The reload safety evaluation is performed to determine whether any changes introduced by the reload core affect the validity of the existing safety limits or reference analyses. The key parameters reviewed to assess the need for recalculation of the DNBR safety limits are those shown in Table 6-3. The same parameters plus the key transient input values are reviewed to determine the need for reanalysis of DNB-limited transients.

Table 6-1

Core Inlet Flow Factors

	Interior Region		Peripheral Region	
Operating Condition	Min	Max	Min	Max
Four Pumps	[			]
Three pumps	[			]

Table 6-2

Initial Conditions for Transient DNBR Calculations

Parameter	Value
Reactor Power	nominal + power uncertainty
RCS Flow Rate	minimum
Core Bypass Flow Fraction	maximum
Reactor Inlet Temperature	nominal + temperature uncertainty
Reactor Exit Pressure	nominal – pressure uncertainty
Radial * local peaking ($F_{\Delta H}^{N}$)	1.800
Axial power distribution	1.65 chopped cosine

Table 6-3

Significant Parameters - Core Thermal-Hydraulics

Parameter	Limiting Direction
Core power	Maximum
Reactor Coolant Inlet Temperature	Maximum
Reactor Coolant Pressure	Minimum
Reactor Coolant Flow	Minimum
Core Bypass Flow	Maximum
Radial Peaking, $F_{\Delta H}^{N}$	Maximum
Axial Peaking, F_Z	Maximum
Engineering Hot Channel Factors	Maximum

Figure 6-1

Example of Reactor Core Safety Limits

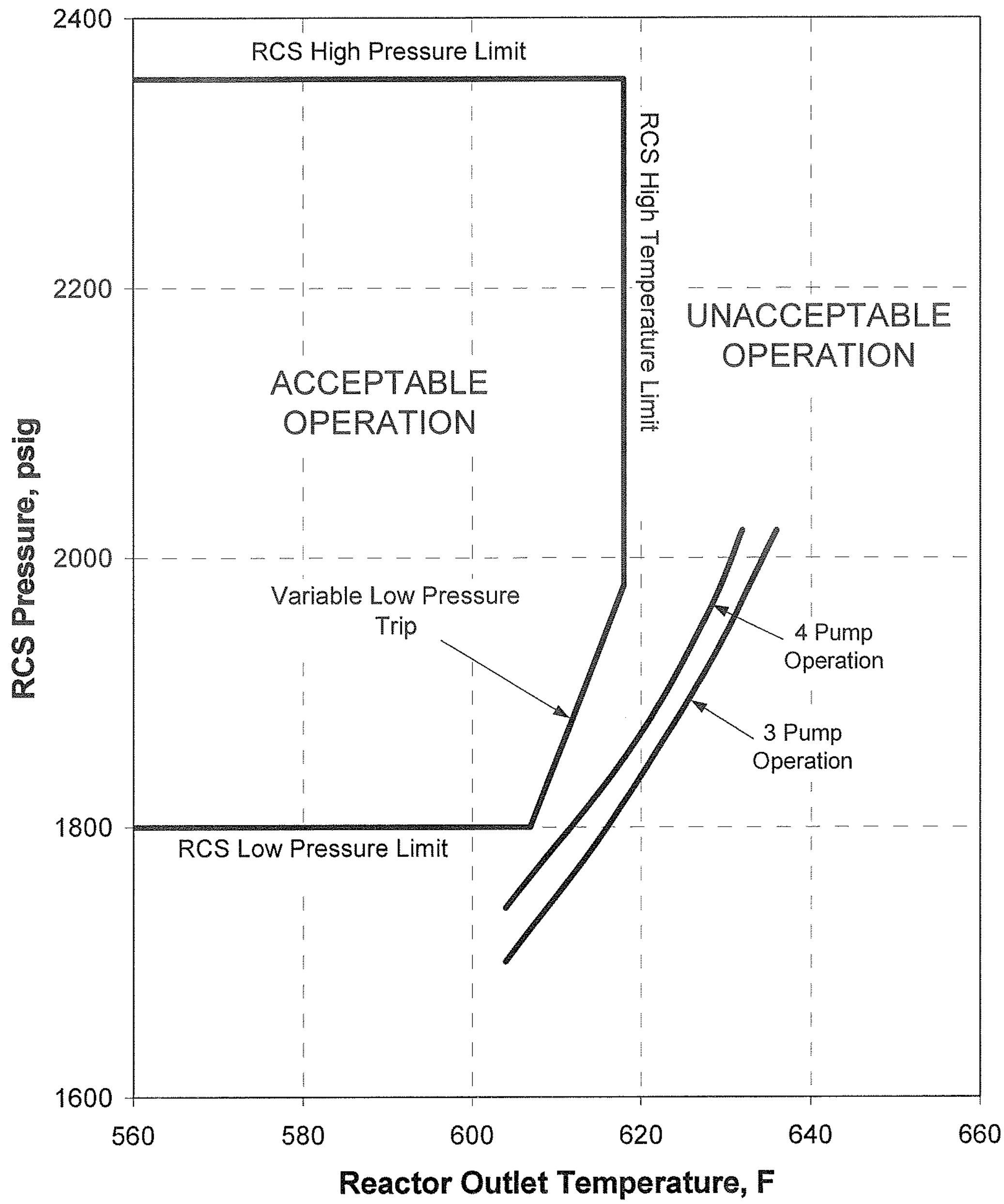


Figure 6-2

Example of Maximum Allowable Radial Peaks

Figure 6-3

Example of Maximum Allowable Total Peaks

SECTION 7 REACTOR PROTECTION SYSTEM SETPOINTS

7.1 High Reactor Coolant System Pressure Trip

7.1.1 Functional Description

The pressure input for the high RCS pressure trip function of the RPS is obtained from the hot leg pressure tap. In general terms, the high RCS pressure trip provides protection for the RCS high pressure safety limit defined by the ASME code. The high RCS pressure safety limit is defined as 110% of the RCS design pressure. For the RCS design pressure of 2500 psig, the safety limit is 2750 psig. The reactor trips on high RCS pressure during transients that result in a net increase in heat production versus heat removal, which is accompanied by an increase in the RCS pressure. The reactor trip limits the severity of the heat removal mismatch and hence the stored energy in the primary system.

7.1.2 Analysis Criteria

The plant systems response analysis (safety analysis) criteria for the high RCS pressure trip are given below. Some of the criteria provide constraints on the assumptions used to determine the setpoint rather than on the setpoint itself.

1. The setpoint shall lie within the detection window of the instrumentation. The RPS pressure instrumentation has a range of 1700 to 2500 psig.

2. The setpoint shall assure a reactor trip 95% of the time at a 95% confidence level, including instrumentation uncertainties. This criterion provides a constraint on the instrumentation error uncertainties from the qualification program and on the combination of the module uncertainties used to determine the total string uncertainty. The instrumentation errors are calculated by a methodology that considers a square root of the sum of the squares of the random terms, to which the bias or correlated terms are then added.

3. The setpoint shall preclude steady-state operation at or above the pressurizer code safety valve setpoints. This criterion is intended to avoid a situation in which a reactor trip on high RCS pressure could be prevented by the lifting of the pressurizer safety valves.

4. The setpoint shall ensure that peak RCS pressure during Condition I and Condition II events remains below the high RCS pressure safety limit. This is an acceptance criterion that applies to the safety analyses that use this trip for reactor protection, rather than being applicable to the trip setpoint itself.

7.1.3 Analysis Trip Setpoint Calculations

The high RCS pressure trip is used for steady-state and transient protection of the reactor. The RPS trip setpoint is based on the pressure at the hot leg pressure tap. The SAR transient analyses for which the high RCS pressure trip is used are: control rod withdrawal at startup, rod withdrawal accident at rated power, moderator dilution event, feedwater line break, control rod ejection, and loss of main feedwater. The setpoints used in the analysis are provided in each plant's SAR.

7.1.4 Delay Times

In addition to the errors discussed above, the accident analyses also assume a delay time that corresponds to the time response characteristics of the equipment associated with the trip. This equipment includes the modules in the instrumentation string as well as the trip breakers and the control rod drive mechanism (CRDM). The actual delay time is the sum of the time responses of each component, from the time the trip condition occurs until the time the control rods are released and begin to insert into the core. The actual delay time is based on equipment testing and bounds the range of parameter changes for which the equipment is assumed to respond. The actual delay time is represented in the analyses as a single value.

7.2 Low Reactor Coolant System Pressure Trip

7.2.1 Functional Description

The pressure input for the low RCS pressure trip function of the RPS is obtained from the hot leg pressure tap. In general terms, the low RCS pressure trip provides protection against DNB during steady-state and transient operation. By initiating a reactor trip during decreasing pressure events, the low RCS pressure trip provides protection for the fuel cladding, fuel, reactor building, and, ultimately, the environment. The DNBR safety limit describes a locus of points for core conditions that mark the transition in heat transfer regimes from nucleate boiling to film boiling along the fuel pins, as determined analytically. The point at which the heat transfer changes from nucleate boiling to film boiling is termed DNB. At this point, the heat transfer from the fuel pin declines due to the insulation of the fuel cladding surface by the steam film. The heat flux necessary to cause the transition from nucleate boiling to film boiling is termed the CHF. The decrease in heat transfer associated with entering film boiling leads to fuel cladding temperature increases. The DNBR is the ratio of the critical heat flux to the actual calculated heat flux.

7.2.2 Analysis Criteria

The analysis criteria for the low RCS pressure trip are given below. Some of the criteria provide constraints on the assumptions used in determining the setpoint rather than on the setpoint itself.

1. The setpoint shall lie within the detection window of the instrumentation. The RPS pressure instrumentation has a range of 1700 to 2500 psig.

2. The setpoint shall assure a reactor trip 95% of the time at a 95% confidence level, including instrumentation uncertainties. This criterion provides a constraint on the instrumentation error uncertainties from the qualification program and the combination of the module uncertainties into the total string uncertainty. The instrumentation errors are calculated by a methodology that considers a square root of the sum of the squares of the random terms, to which the bias or correlated terms are then added.

3. The setpoint shall be approximately 250 psi greater than the emergency core cooling system bypass setpoint. This setup provides sufficient margin following a reactor trip for the operators to bypass the safety injection systems, if desired.

4. The setpoint in conjunction with the high-temperature and variable-low- pressure trip shall prevent the core from exceeding the steady-state DNBR safety limit.

7.2.3 Analysis Trip Setpoint Calculations

The low RCS pressure trip is used for steady-state and transient protection of the reactor. The RPS trip setpoint is based on the pressure at the hot leg pressure tap.

SAR transients for which the low RCS pressure trip is credited result in an RCS pressure decrease and include the SBLOCA, steam generator tube rupture (SGTR) and steam line break. Although not credited in the SAR analyses for LBLOCA, the low RCS pressure trip also provides reactor protection for LBLOCA. In this case, the setpoint is a low parameter setpoint, and the safety limit is defined in terms of DNBR and not strictly pressure.

7.2.4 Delay Times

The concept and methodology for determining the analysis values of the delay times associated with the low RCS pressure trip are the same as those for the high RCS pressure trip, as is discussed in Section 7.1.4.

7.3 Power/Imbalance/Flow Trip

7.3.1 Functional Description

A complete description of the protection provided by this trip is included in Section 5.3.1. The function of this trip is to develop an allowed maximum power level for the current flow and axial power imbalance conditions in the plant. The process can be explained in the following way: The reactor coolant (RC) flow is measured and multiplied by the flux/flow setpoint to arrive at a maximum allowed power with no imbalance. The function generator uses this power level to assess a power penalty based on the measured imbalance conditions. This final allowed power is then compared to total measured power to determine whether a trip condition exists. Figure 7-2 shows the general shape of the power/imbalance/flow trip envelope. The plant can safely operate within this envelope, but once power and imbalance values exceed those of the trip envelope, the reactor is placed in a trip condition. At lower power levels, imbalance protection is not required. However, due to current RPS hardware limitations, a trip setpoint is still defined for the lower power levels.

7.3.2 Analysis Criteria

The analysis criteria for the power/imbalance/flow trip are listed below.

1. The setpoint shall prevent violation of the power-imbalance protective limit and the power/flow protective limit.

2. The setpoint shall be chosen such that the trip condition can be measured by the out-of-core neutron detectors. This criterion ensures that the core condition can be measured at the point where a trip is desired.

3. The setpoint shall assure a reactor trip 95% of the time at a 95% confidence level, including instrumentation uncertainties. This criterion provides a constraint on the instrumentation error uncertainties from the qualification program and the combination of the module uncertainties into the total string uncertainty. The instrumentation errors are calculated by a methodology that considers a square root of the sum of the squares of the random terms, to which the bias or correlated terms are then added.

4. The RC flow noise at the plant shall be less than that assumed in the setpoint calculations. A typical value of flow noise is between 1.5% and 2.4% of full flow. The flow noise may be minimized through mechanical or electronic filtering.

5. The RC flow input to the RPS shall be calibrated to indicate 100% full flow, or less, when the plant is at full power with four pumps operating. This requirement ensures that the assumptions of the accident analysis are maintained in the plant system calibration.

7.3.3 Analysis Trip Setpoint Calculations

The value of the power/imbalance/flow trip setpoint is a fuel cycle-specific envelope that is dependent upon the cycle core peaking. The flux/flow limit value (S_m) is generated with a reference design peaking distribution (Section 6.2.2) during the analysis of loss of RCS flow events for each cycle. The peaking for each cycle is used to generate the power-offset limits. S_m is used with the power-offset limits to determine the power/imbalance/flow setpoint.

7.3.3.1 Flux/Flow Setpoint Error Adjustment

7.3.3.2 Power-Imbalance Envelope Determination

7.3.4 Delay Times

The only delay time of interest for the power/imbalance/flow trip is associated with the flow measurement. Since the other functions of the trip are not considered in the analyses, their delays do not significantly affect the analyses that rely on the power/imbalance/flow trip. The delay time for the flow string is determined by adding the response characteristics of the different modules to arrive at the total string delay time. The trip breaker and CRDM release times are included in the total delay.

7.4 High Flux Trip

7.4.1 Functional Description

The neutron flux input for the RPS high flux (nuclear overpower) trip function of the RPS is obtained from the out-of-core power range neutron detectors. In general terms, the high flux trip provides protection for the DNBR, TCS and CFM limits. By initiating a reactor trip during increasing reactivity events, the nuclear overpower trip also provides some protection for the pressure boundary and the reactor building.

7.4.2 Analysis Criteria

The analysis criteria for the high flux trip are listed below.

1. The setpoint shall assure a reactor trip 95% of the time at a 95% confidence level, including instrumentation uncertainties. This criterion provides a constraint on the instrumentation error uncertainties from the qualification program and the combination of the module uncertainties into the total string uncertainty. The instrumentation errors are calculated by a methodology that considers a square root of the sum of the squares of the random terms, to which the bias or correlated terms are then added.

2. The setpoint shall be chosen such that the trip condition can be measured by the out-of-core neutron detectors. This criterion ensures that the core condition can be measured by the trip string at the point chosen to keep thermal power less than design overpower.

7.4.3 Analysis Trip Setpoint Calculations

The nuclear overpower trip provides protection during steady-state and transient conditions. The accidents for which the nuclear overpower trip provides protection are the startup accident, the rod withdrawal accident, the rod ejection accident, and the steam line break accident. A reactor trip on high flux provides direct protection of the DNBR, TCS and CFM safety limits.

The nuclear overpower trip value used in the original licensing systems response analyses was 114 %FP. Additional studies performed in the early 1970's utilized better predictions of the performance of fuel in reactors than were available for the original analyses. The results of the later analyses led to changes in the assumptions of the physical properties of the fuel following irradiation. The effect noted at that time, and evaluated for each plant, is fuel densification. This effect causes the fuel pin heat transfer to change, thus affecting various DNB and CFM calculations. The fuel densification reports for each plant describe the evaluation performed for the safety analysis and fuel cycle design parameters. Application of a densification power spike penalty is no longer required for thermal-hydraulic analyses (see Section 6.2.7). As a result of the later studies, the peak power considered in subsequent analyses was reduced to a power level which corresponds to the design power limit for the B&W 177-FA plants (typically 112 %FP).

Additional analyses have been performed to verify the acceptable core response to overcooling conditions when the nuclear overpower trip string errors are effectively increased due to shielding of the out-of-core neutron detectors by the cooler reactor vessel downcomer fluid. The evaluations provided assurance that sufficient margin exists for both DNBR and kW/ft conditions even when the actual core power exceeds the design power limit.

7.4.4 Delay Times

The delay time is based on equipment testing and bounds the range of rates of parameter changes to which the nuclear overpower string is assumed to respond. The delay time is composed of the time constants of the various modules contained in the trip string. The total trip delay time used in systems response analyses also includes contributions from the trip breakers and the CRDM. The individual component delay times are summed to arrive at the total trip delay time accounted for in the analyses.

7.5 High Reactor Coolant Outlet Temperature Trip

7.5.1 Functional Description

The temperature for the RPS high RCS Outlet Temperature trip function is obtained from the temperature sensors in the hot legs. The high RCS outlet temperature trip establishes an absolute upper limit on RCS outlet temperature. In doing so, the limit restricts the range over which the RPS variable low reactor coolant pressure trip (VLPT) function must provide protection. This trip provides backup protection for RCS overheating events. The trip is not used as the primary trip function for any SAR accident systems response analyses.

7.5.2 Analysis Criteria

The analysis criteria for the high reactor coolant outlet temperature trip are given below.

1. The setpoint shall ensure a reactor trip 95% of the time at a 95% confidence level, including instrumentation uncertainties. This criterion provides a constraint on the instrumentation error uncertainties from the qualification program and the combination of the module uncertainties into the total string uncertainty. The instrumentation errors are calculated by a methodology that considers a square root of the sum of the squares of the random terms, to which the bias or correlated terms are then added.

2. The setpoint in conjunction with the low-pressure and variable-low- pressure trip shall prevent the core from exceeding the steady-state DNB safety limit.

3. The setpoint shall lie within the detection window of the instrumentation. The RPS temperature instrumentation has a range of 520 to 620 $^{\circ}$F.

7.5.3 Analysis Trip Setpoint Calculations

Since the high RCS outlet temperature trip is not required to meet any acceptance criteria of any transient considered in accident analysis, the accident analysis setpoint is assumed to be at the upper limit of the temperature measurement instrumentation. This value (620 $^{\circ}$F) is used in steady-state calculations in which the high-temperature trip is used for limiting the temperature range of the variable-low-pressure trip.

7.5.4 Delay Times

The delay time for this trip is determined from the summation of the time constants of each module. The delay time itself, however, is not limited by the accident analysis since the high RCS outlet temperature trip is not assumed to terminate any events. The limitation on the total allowed string delay time is based on engineering judgment and guidance provided by the NRC in various regulatory documents including Regulatory Guides and Generic Letters. Currently, a delay time of 6 seconds on the temperature string is considered conservative and appropriate for the safety functions that this string performs.

7.6 Variable Low RC Pressure Trip

7.6.1 Functional Description

The VLPT provides primary steady-state DNBR protection. This protection is accomplished by tripping the reactor before system parameters reach a P-T combination that could lead to DNB in the core, assuming conservative core power and peaking conditions. The protection provided by this trip amounts to a "floating" low pressure trip that is dependent upon the measured core outlet temperature.

7.6.2 Analysis Criteria

The analysis criteria for the VLPT are given below.

1. The setpoint shall ensure a reactor trip 95% of the time at a 95% confidence level, including instrumentation uncertainties. This criterion provides a constraint on the instrumentation error uncertainties from the qualification program and the combination of the module uncertainties into the total string uncertainty. The instrumentation errors are calculated by a methodology that considers a square root of the sum of the squares of the random terms, to which the bias or correlated terms are then added.

2. The setpoint, in conjunction with the low-pressure and high-temperature trips, shall prevent the core from exceeding the steady-state DNBR safety limit.

3. The setpoint shall lie within the capabilities of the instrumentation. This criterion places a limitation on the slope of the VLPT setpoint.

7.6.3 Analysis Trip Setpoint Calculations

The accident analysis value for this trip is actually an adjusted P-T limit that is specified by DNB analysis. The DNB limits are evaluated for each allowed reactor coolant pump status at the maximum allowed steady-state power level for a particular pump combination. The four-pump limit is the design overpower value, typically 112 %FP. The three- and two-pump limits are evaluated at the maximum power level, as determined using the flux/flow setpoint and assuming the worst case instrument errors. Additional assumptions on design power peaking in the core are used in determining these power limits.

7.6.4 Delay Times

The delay time can be calculated for both pressure and temperature changes by summing the time constants of each module. The signal converter and the temperature sensor are not included as part of the pressure delay, since the pressure is assumed to move toward a setpoint that is remaining constant. The pressure transmitter is excluded from the temperature delay since the temperature portion of the setpoint moves toward the fixed pressure value. As with the other instrumentation strings, the contribution of the trip breakers and the CRDM release are included in the total delay time.

7.7 High Reactor Building Pressure Trip

7.7.1 Functional Description

The high reactor building pressure trip initiates a reactor trip whenever the building pressure increases beyond the pressure setpoint. These conditions indicate that a high energy line break is present. This trip is expected to be the primary trip for small breaks in high-energy lines inside containment, such as LOCAs, steam line breaks, and feedwater line breaks. Although this would be the primary trip that provides protection for the plant, this trip is not directly credited in the accident analyses.

7.7.2 Analysis Criteria

The analysis criteria for the high reactor building pressure trip are given below.

1. The setpoint shall ensure a reactor trip 95% of the time at a 95% confidence level, including instrumentation uncertainties. This criterion provides a constraint on the instrumentation error uncertainties from the qualification program and the combination of the module uncertainties into the total string uncertainty. The instrumentation errors are calculated by a methodology that considers a square root of the sum of the squares of the random terms, to which the bias or correlated terms are then added.

2. The setpoint shall lie within the measurement range of the instrumentation.

7.7.3 Analysis Trip Setpoint Calculations

This trip is not used in the NSSS systems response analyses. Both the LOCA and non-LOCA analyses use either the low RCS pressure trip or the high flux trip for transients where the high reactor building pressure trip would be expected to provide primary core protection. Therefore, this trip remains as a backup to the other RPS trip functions for the systems response analyses. The high reactor building pressure trip is used for containment pressure and temperature analyses. The nominal trip setpoint for the containment analyses is taken as 4 psig, but varies from plant to plant.

7.7.4 Delay Times

The delay time for this trip is determined from the various time delays of the different components. As with the other instrumentation strings, the contribution of the trip breakers and the CRDM release are included in the total delay time.

7.8 Power/Pump Monitors Trip

7.8.1 Functional Description

The power-to-pump monitors trip ensures a reactor trip when no reactor coolant pumps are operating in one steam generator loop (0/0, 1/0, 0/1, 2/0 or 0/2 operation). This trip also provides the primary protection for the following events:

1. Multiple reactor coolant pump coastdowns.
2. Single reactor coolant pump coastdown from partial pump operation.
3. Reactor coolant pump coastdowns resulting in the loss of both pumps in either loop.

The trip is designed to operate in a nearly binary manner: power operation allowed or power operation not allowed. The determination is based on the comparison of the measured neutron power to the allowed power for the pump combination operating. The pump monitor in the trip string rapidly determines allowed power level for the pump combination and the bistable then determines the need for trip.

7.8.2 Analysis Criteria

The analysis criteria for the power-to-pump monitors trip are given below.

1. The setpoint shall ensure a reactor trip 95% of the time at a 95% confidence level, including instrumentation uncertainties. This criterion provides a constraint on the instrumentation error uncertainties from the qualification program and the combination of the module uncertainties into the total string uncertainty. The instrumentation errors are calculated by a methodology that considers a square root of the sum of the squares of the random terms, to which the bias or correlated terms are then added.

2. The setpoint shall lie within the measurement range of the instrumentation.

3. The trip shall prevent operation with an idle steam generator loop. Initial calculations and testing indicated the need to preclude idle loop power operation, and this trip function has been used to ensure that the condition is met.

7.8.3 Accident Analysis Trip Setpoint Calculations

The reactor coolant pump power monitor trip is used for steady-state and transient protection of the reactor. The SAR transient applicable for the trip is the LOCF event, which is initiated by a reduction in or loss of forced flow through the RCS. The reduction in flow may be due to a mechanical failure in the reactor coolant pumps or a loss of electrical power to the reactor coolant pump motors. With the reactor at power, the result is an increase in the RCS temperature and a reduction in the heat removal capability of the reactor coolant. These two conditions could result in DNB in the core.

As modeled in the accident analyses, the power/pump monitor trip is a digital type trip. If the accident analyses assume a loss of primary flow from a reactor coolant pump trip, the trip condition is based on the initial core power. If the core power is found to be greater than the allowed final pump status power level, a trip condition exists and a reactor trip is initiated. Since the consequences of this transient are more severe at high power levels, a specific analysis setpoint has not been defined by the analyses. For all idle loop conditions, the allowed power level is 0 %FP. For the condition of only one pump operating in each loop, the setpoint is either 55 %FP or 0 %FP, depending on whether or not the plant is licensed to operate in this configuration.

7.8.4 Delay Times

The accident analyses assume a response time characteristic for this instrumentation, i.e., the delay time. The delay time is a single number that accounts for the response of the modules in the string as well as the delay time contributions from the breakers and the CRDM. The delay time is based on equipment testing and bounds the range of parameter changes to which the reactor coolant pump power monitor trip string is assumed to respond. The delay time is composed of the time constants of the various modules. A simple sum is taken to arrive at the total trip delay time. A typical delay time assumed for this trip is 0.620 second, although plant-specific evaluations have been performed to allow a longer delay time.

Figure 7-1

General Setpoint Philosophy

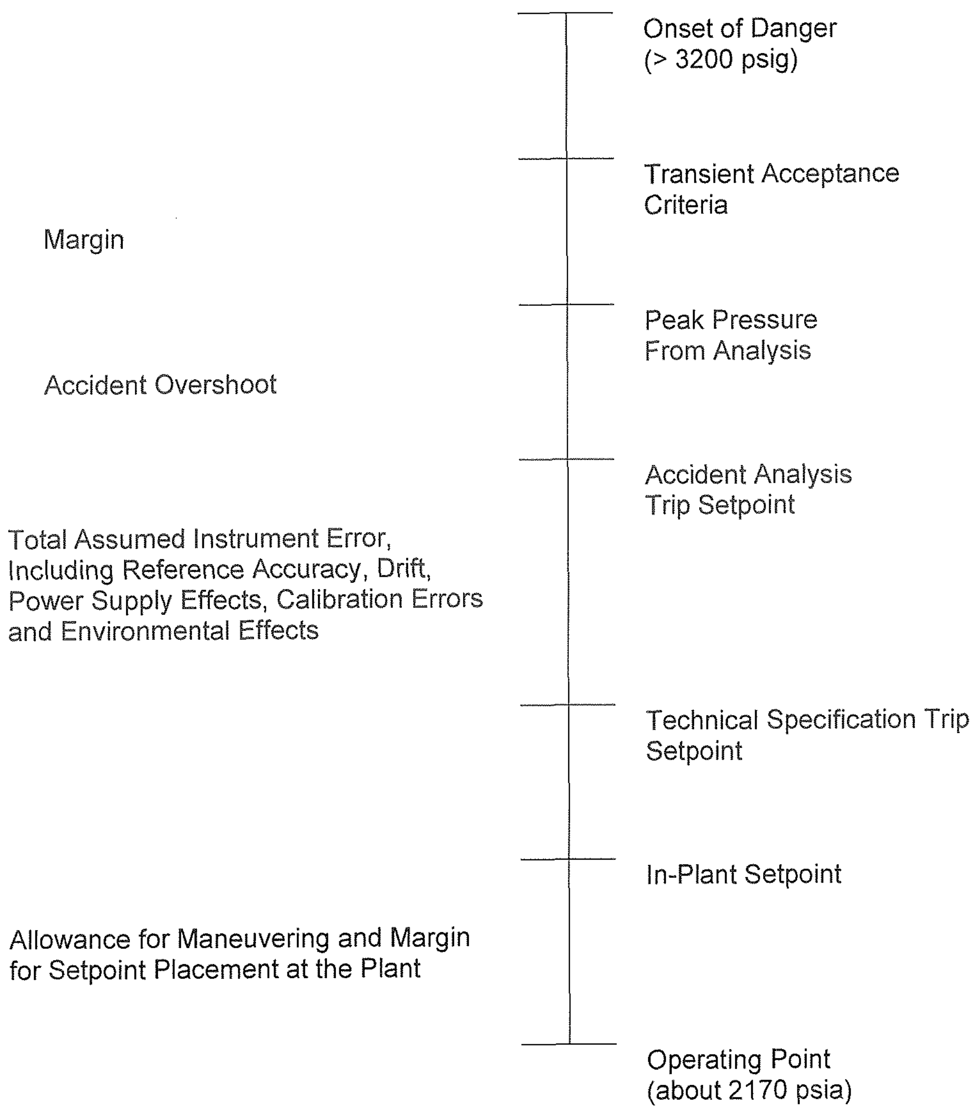


Figure 7-2

Power/Imbalance/Flow Trip Setpoint

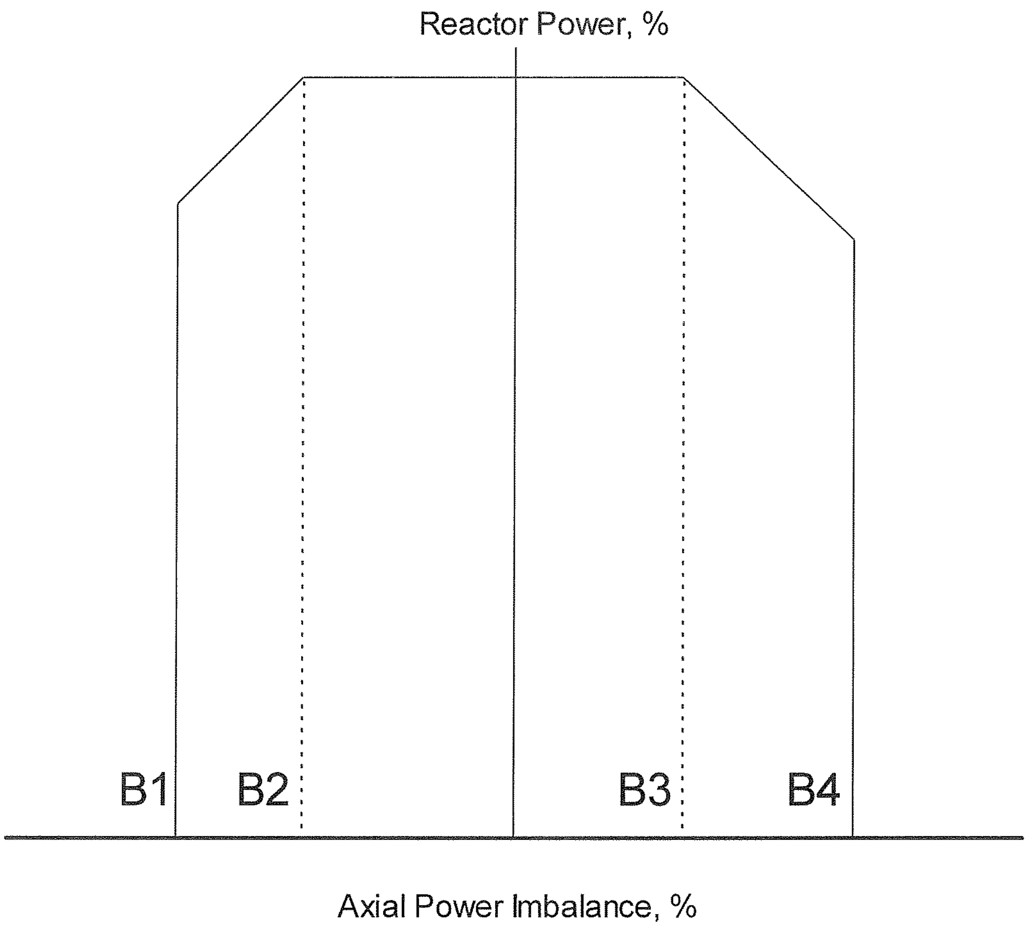


Figure 7-3

Power/Imbalance/Flow Detectability Envelope

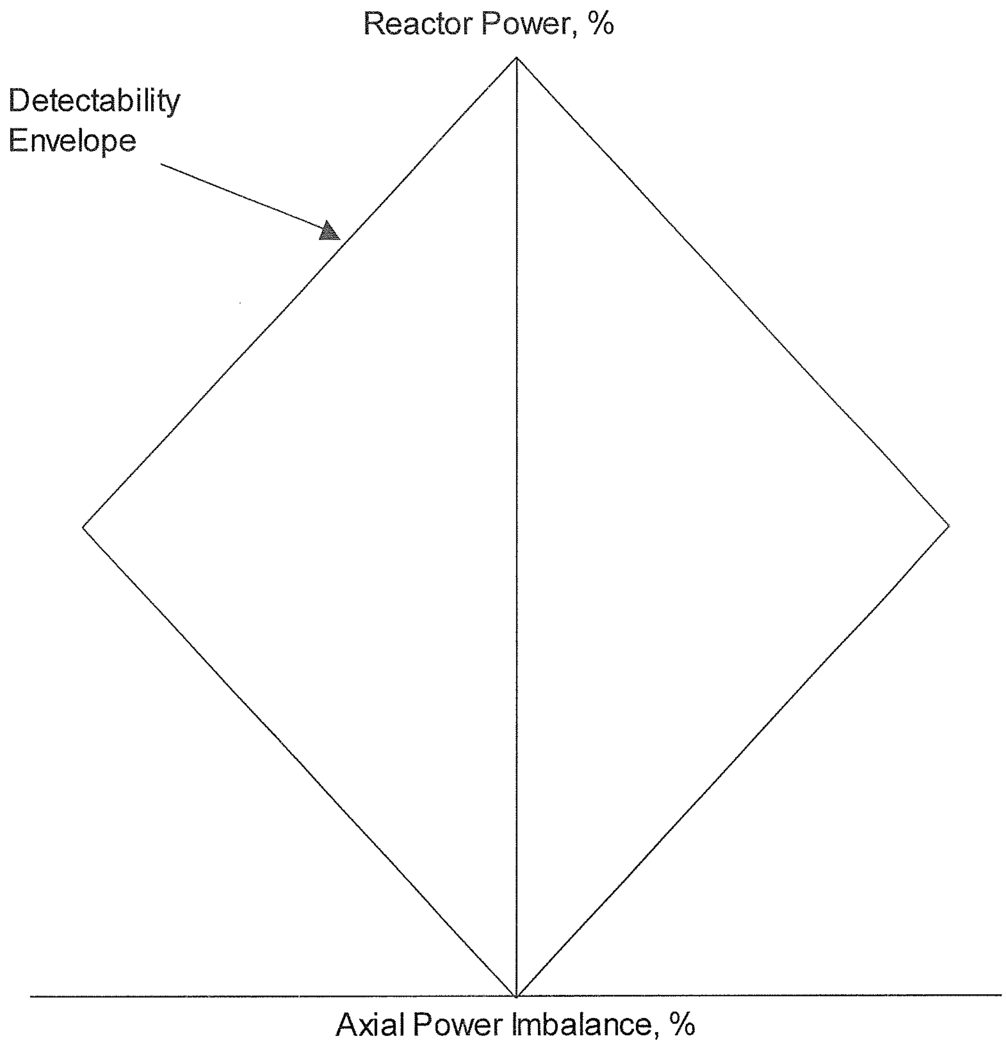


Figure 7-4

Power/Imbalance/Flow Trip Setpoints for Four, Three and Two Pump Operation

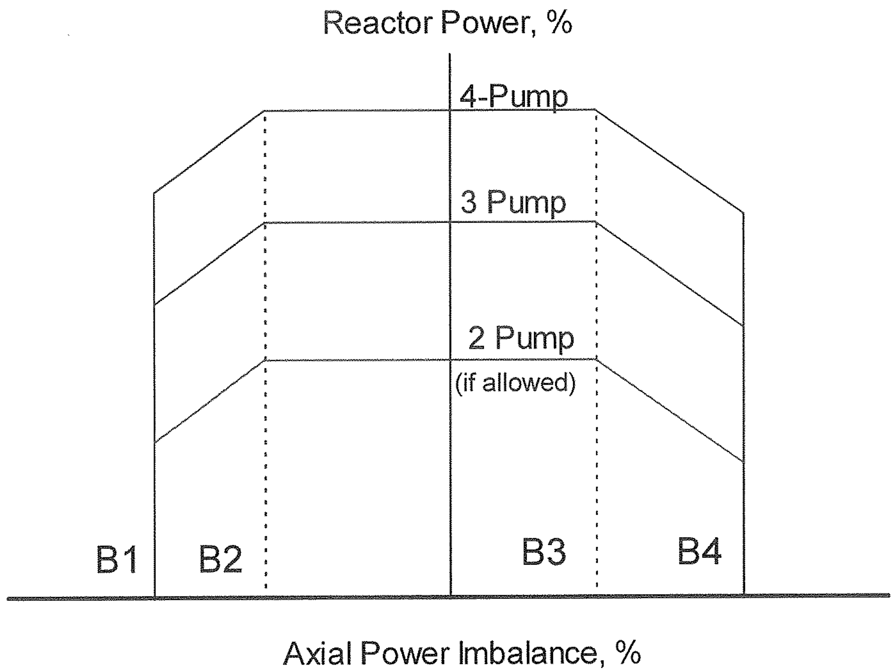


Figure 7-5

Variable Low Pressure Trip Setpoint

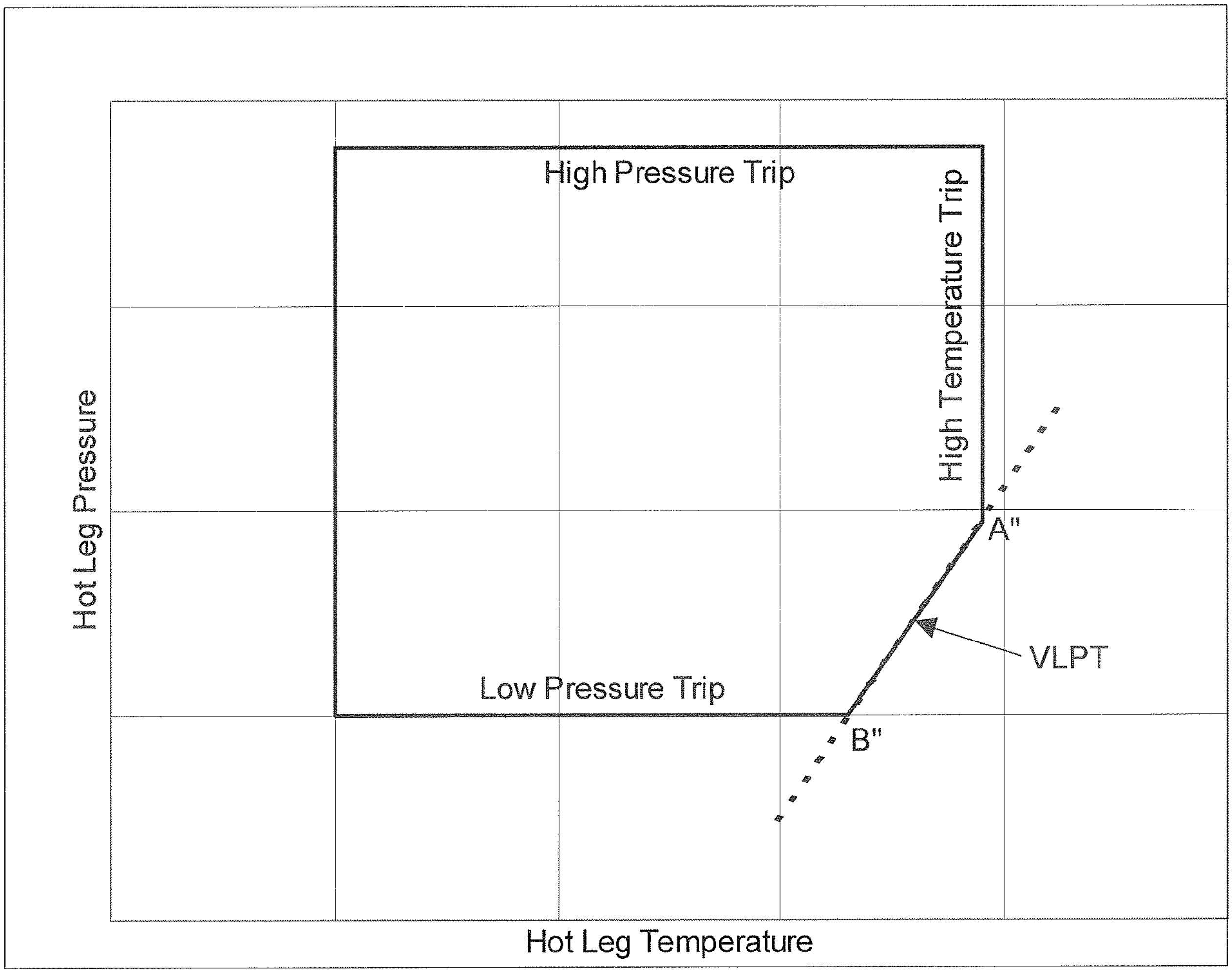


Figure 7-6

Variable Low Pressure Setpoint Slope Rotation

SECTION 8 NON-LOCA ACCIDENT EVALUATION

8.1 Systems Response Analyses

Systems response analyses have been performed for the events discussed in the plant SARs. The assumptions used in the SAR analyses must be shown to bound the corresponding cycle-specific parameters in order to avoid a re-evaluation of the accidents for each fuel cycle. Therefore, the general requirement is that the predicted parameter values of a reload fuel cycle lie within the current licensing base of the plant. A discussion follows on how each of the key parameters is evaluated for each fuel cycle against the licensing analysis for each plant.

8.1.1 Moderator and Doppler Coefficients

The Doppler and moderator coefficients are a measure of the reactivity change resulting from changes to the fuel and moderator temperatures, respectively. The cycle-specific moderator and Doppler reactivity coefficients shall lie within the analyzed range considered in the SAR accident analyses. This requirement ensures that the core power response, were an accident to occur during the fuel cycle, would be bounded by a response that has been shown to meet the applicable acceptance criteria in the accident analyses. Depending on the event being considered, the moderator and Doppler coefficients are evaluated either individually or on a combined-effects basis.

8.1.1.1 BOC Conditions

The beginning-of-cycle (BOC) Doppler and moderator coefficients are used in the systems response analysis of BOC-limited events, i.e., events that result in an increase to the fuel and RCS fluid temperatures. Examples of heatup events include:

1. Moderator Dilution Event
2. Rod Withdrawal from Rated Power
3. Startup Event
4. Loss of Main Feedwater
5. Feedwater Line Break
6. Rod Ejection
7. Loss-of-Coolant Flow
8. Loss of Electric Power

For heatup events, a less-negative reactivity coefficient results in less negative reactivity addition to the reactor core. Consequently, more power is generated in the core and more heat is added to the RCS, thereby producing more severe results. The combined effects of the cycle-specific BOC Doppler and moderator coefficients must provide a smaller positive reactivity addition during heatup events than was calculated in the accident analyses. If this condition is met, the cycle-specific BOC reactivity coefficients are deemed acceptable.

8.1.1.2 EOC Conditions

The EOC Doppler and moderator coefficients are used in the systems response analyses of EOC-limited events, i.e., events that result in a decrease to the fuel and RCS fluid temperatures. Examples of cooldown events include:

1. Steam Line Breaks
2. Cold Water Event (Pump Restart)

The EOC Doppler and moderator coefficients are more negative than their BOC counterparts. The negative Doppler and moderator coefficients, combined with a fuel and moderator temperature decrease, result in positive reactivity insertion into the core. The more negative the coefficients, the greater the positive reactivity insertion. Maximizing the post-trip positive reactivity insertion is conservative for the overcooling events. The combined effects of the cycle-specific EOC Doppler and moderator coefficients must provide less positive reactivity during cooldown events than was calculated in the accident analyses. If this condition is met, the cycle-specific EOC reactivity coefficients are deemed acceptable.

8.1.2 BOC Soluble Boron Concentration

The initial (BOC) boron concentration is of importance to the moderator dilution event analysis. The reactivity insertion rate for this event depends on the total reactivity worth of the boron in the system. The total boron worth is determined by dividing the initial boron concentration, in units of ppm, by the inverse boron worth, in units of ppm/% reactivity. The result is the total reactivity held by the boron that is dissolved in the reactor coolant, i.e., the boron worth. During a boron dilution event, this ratio provides a means of determining the rate of reactivity addition as the boron concentration is diluted. The greater the cycle-specific boron worth, the more rapid the reactivity insertion as the dilution accident proceeds. Thus, the cycle-specific values must yield a lower boron worth than that assumed in the accident analysis. If this criterion is met, the boron worth is deemed acceptable.

8.1.3 Dropped Control Rod Worth

The dropped rod worth is used in the systems response analyses of a dropped or stuck rod (CRA). The consequences of the dropped rod accident bound those of the stuck-in or stuck-out rod events. The higher the dropped rod worth, the greater the resultant power suppression will be. The power suppression results in a coolant temperature decrease, which coupled with negative reactivity coefficients causes a positive reactivity insertion. The analyses have shown acceptable results for a range of dropped rod worths and reactivity coefficients that represent fuel burnup conditions from BOC through

EOC. Therefore, in order to be acceptable, the maximum cycle-specific dropped rod worth must lie within the range of the rod worth values considered in the analyses.

8.1.4 Ejected Rod Worth

The ejected rod worth is used in the systems response analysis of the rod ejection accident. For this event, a larger ejected rod worth results in more severe reactivity insertion and subsequent power excursion. The analyses have shown acceptable results for a range of ejected rod worths. Therefore, in order to be acceptable, the maximum cycle-specific ejected rod worth must be less than the maximum rod worth considered in the analyses.

8.1.5 Single and All-Control-Rod-Group Worth

The all-rod-group worth at HZP is used in the systems response analysis of the startup event. The higher the rod worth, the higher the rate of positive reactivity insertion into the core during the event. The analyses have shown acceptable results for a range of rod worths and corresponding reactivity insertion rates. Therefore, in order to be acceptable, the maximum cycle-specific all-rod-group worth must lie within the range of the reactivity insertion rates considered in the analyses of the startup event.

The analyses of the rod withdrawal accident at rated power were based on reactivity insertion rates independent of actual group worths. Spectrum studies of rod worth included reactivity insertion rates ranging from that corresponding to less than a single rod to that equivalent to withdrawing all the rods at once. The spectrum study demonstrated the acceptability of all reactivity insertion rates considered. Within the SARs, reference points were provided to give the reader a gauge as to what reactivity insertion rates approximately corresponded to a single control rod, a single group, and an all-group condition. These worths were intended to serve as typical values based on the original core designs. As long as the cycle-specific all-rod-group worth falls within the range that corresponds to the reactivity insertion rates considered in the analyses, the cycle-specific all-rod-group worth is deemed acceptable.

8.1.6 Delayed Neutron Fraction (β_{eff})

The effective delayed neutron fraction is another kinetics parameter that is used in the systems response analyses. However, β_{eff} is not a significant factor in the systems response to a postulated transient. Accordingly, a nominal or representative value, corresponding to the time-in-life that is being analyzed, is used as analysis input. The accident analysis philosophy is based on performing overall-conservative analyses, which incorporate bounding values for the significant parameters, rather than bounding every parameter. To that end, the conservatism that is built into the analyses is a result of the use of bounding values for moderator and Doppler coefficients, and includes conservative boundary conditions that are imposed on the RCS and secondary plant systems. Therefore, a rigorous check of the cycle-specific β_{eff} values is not required.

8.1.7 Additional Cycle-Specific Evaluations

In conjunction with the fuel reloads, changes to the fuel design may be made that are not typically evaluated in the systems response analyses. Examples include:

1. Modifications to the fuel assembly hold-down springs,
2. Addition of debris filters,
3. Improvements to the fuel assembly that facilitate recaging,
4. Plugging or sleeving of additional steam generator tubes,
5. Replacement of damaged fuel pins in reinserted fuel assemblies with stainless steel replacement rods, or
6. Modification of the fuel spacer grids or fuel cladding to improve thermal and wear performance.

Typically, these changes do not have a significant effect on the overall system response. Nevertheless, limits have been established, such as the allowable number of plugged or sleeve SG tubes or the number and location of inserted stainless steel rods, which govern the extent and consequences of these changes. Evaluations are performed each cycle to ensure that the appropriate limits are not violated and the overall conservatism of the boundary conditions and key input parameters used in the analyses is maintained. If these evaluations determine that systems response analyses of one or more of the SAR accidents is warranted because of a design change, then the appropriate analyses are performed.

8.1.8 EOC Average Reactor Coolant Temperature Reduction

Fuel cycle designs frequently include provisions for EOC maneuvers to ensure that cycle operation is maintained for the entire designed length. These EOC maneuvers typically include an EOC T_{avg} reduction. The effects of such EOC T_{avg} decreases on the systems response analyses have been evaluated, and allowable values for the T_{avg} reduction have been defined. As long as the specified cycle-specific maximum EOC T_{avg} reduction is within the defined limit, with measurement uncertainty accounted for, the analyses of record will remain applicable for the cycle.

8.2 Accident Evaluation

The process of accident evaluations for a given plant is recurrent and on-going during its years of operation, as plant modifications and additional regulatory issues have arisen. To ensure a consistent set of analyses, common acceptance criteria have been used for determining the acceptability of the consequences of a given accident. As long as it can be shown that the acceptance criteria are met for each reload fuel cycle, the cycle design is considered acceptable. Many of the criteria are related to the parameters discussed in sections 5, 6 and 8.1, although some are dependent on other non-fuel-cycle-dependent parameters.

The acceptance criteria are aimed at measuring the performance of the radiation barriers both during and after an accident. The barriers between the fuel radioactivity and the environment are described in the Code of Federal Regulations as the fuel cladding, the RCS pressure boundary, and the reactor building.

Some of the acceptance criteria are event-specific and reflect the approach to risk as a combination of the event consequences and the probability that the event can occur. In general, the risk to the public can be described by a multiplication of these two factors, with units chosen as desired to create common units across all events. Thus, a given level of risk can be achieved by either a high probability event with low consequences, or a low probability event with high consequences. The criteria are set to provide this type of risk management for the various events. For each acceptance criterion, the basis for its use and how it is applied to a fuel reload evaluation will be discussed.

1. Condition I and Condition II events shall not lead to fuel failure, thus maintaining offsite doses at the level consistent with normal steady-state operation.

The Condition I and II events include normal operation and events with an expected frequency of 10^{-1} events per year. Since these conditions could reasonably be expected to occur during the fuel cycle, the consequences must be small to assure minimum risk to the public. The choice of no fuel cladding failure, as a consequence of normal operation or moderate frequency events, addresses the performance of the first barrier between the radioactivity contained in the fuel and the environment. By precluding fuel cladding failures during these conditions, no additional radioactivity beyond that found under normal conditions will be released to the RCS. Consequently, the resulting radiation releases to the environment are no higher than normal conditions allow.

An accident is considered to meet this acceptance criterion if the DNBR and TCS remain within limits, and the LHR for CFM is not exceeded. The accident analyses demonstrate the ability of the plant configuration and fuel cycle design to meet these requirements.

2. The ejection of the maximum worth rod shall result in a peak fuel enthalpy less than or equal to 280 cal/gm, the threshold for gross fuel failure.

The rod ejection accident results from a postulated rapid ejection of a single CRA from the core region during operation. The CRA ejection is driven by the pressure differential between the RCS and the containment. This pressure differential acts on the control rod following a postulated breach of the RCS pressure boundary in the control rod drive housing. The ejection is assumed to occur in less than 0.20 second. As the rod is ejected, positive reactivity is added to the core. The amount of positive reactivity addition is based on the worth of the ejected rod. The neutron power rise is extremely rapid, resulting in a near-adiabatic fuel temperature increase. The Doppler feedback limits the power increase prior to a reactor trip. The rapid increase in core power and fuel temperature could lead to fuel failures.

There are three modes by which the fuel can fail due to a rod ejection. The first is associated with low worth rod ejections and leads to very little fuel fragmentation internal to the cladding. Localized cladding degradation due to fuel pin pressure increases is insufficient to rupture the cladding outright, but could weaken it so that DNB could lead to failure. The second fuel failure mode is linked to higher reactivity insertion rates, which lead to significant fuel melting in the ejected rod region. This failure mode includes rupture of the cladding and dispersion of the fuel and cladding into the coolant. In the third and most serious failure mode, the fuel quickly vaporizes. The result is a rapid pressurization in the fuel pin, which can lead to cladding failures without elevated cladding temperatures.

Investigations into the fuel failure modes have been made with the transient test reactor TREAT to determine the conditions for fuel failures (reference 66). The onset of fragmentation of the fuel is marked by a fuel enthalpy of 280 cal/gm. Between 400 and 500 cal/gm, the transition from the second failure mode to the third mode occurs. The transition is a result of pressure pulses that are not evident below ~400 cal/gm.

The acceptance criterion of 280 cal/gm is established at the fuel enthalpy associated with the beginning of fuel fragmentation, but much lower than the +400 cal/gm associated with the rapid change in phase from solid to vapor. Core designs are restricted to those that do not lead to excessive fuel enthalpies, in order to avoid gross fuel failure.

This acceptance criterion is met each fuel cycle by limiting the ejected rod worth to less than that shown to be acceptable in the analysis. This topic was also discussed in section 8.1.4.

3. The peak fuel cladding temperature during a LOCA shall not exceed 2200 ^{o}F based on an approved evaluation model (EM) analysis that incorporates 10CFR50 Appendix K models.

Discussions of LOCA acceptance criteria and evaluations are provided in Section 9.0.

4. The peak reactor building pressure during high energy line breaks shall not exceed the design pressure.

Since the reactor building provides the third and ultimate barrier between the fuel radioactivity and the environment, its continued performance during all accidents must be demonstrated. Of utmost importance is the structural integrity of the containment under those accident conditions that could result in fuel failures. The accidents that provide the largest challenge to the reactor building structural integrity are the high energy line breaks, which release a significant quantity of mass and energy to the reactor building. Examples of this type of accident are the LOCA, steam line break, and feedwater line break accidents.

Analyses of these events demonstrate that the LOCA provides the largest mass and energy release to the reactor building, and as such provides the greatest challenge to its structural integrity. Mass and energy releases from conservative LOCA analyses have been evaluated for each plant and shown to provide acceptable reactor building pressures. Each cycle, the key parameters related to the reactor building pressurization analyses are reviewed to ensure that the licensing analyses are still bounding. This review may occur at either the utility or FANP during the normal cycle review process.

Sensitivity studies have shown that the largest factor in the reactor building pressure analysis is the set of initial RCS conditions, which defines the average enthalpy released over the first 20 to 30 seconds of a LBLOCA. Since this is the case, any control changes affecting the RCS initial conditions are reviewed for impact on the mass and energy releases to the reactor building.

5. The reactor shall not return to criticality following a reactor trip during a steam line break event.

During a steam line break, the RCS is cooled as the secondary system is depressurized and cooled by the fluid discharge through the break. This event is EOC-limited because the more-negative moderator reactivity coefficient values at EOC, combined with the decrease in the moderator temperature, cause a positive reactivity addition. For an extended RCS cooldown resulting from a large steam line break, the negative moderator reactivity coefficient could more than offset the negative reactivity inserted by the control rods after reactor trip. As required by the GDC contained in Appendix A of 10CFR50, a redundant means of reactivity control must be provided. The B&W-designed NSSS uses soluble boron in the fluid injected by the high pressure injection (HPI) system. The injected boron will ensure long-term reactivity control. In the short-term, however, the reactor could return to criticality if other constraints were not placed on the core and plant design.

The potential return to criticality does not necessarily lead to fuel failure. However, the confirmatory analytical effort involves using three-dimensional models and is extensive and complex. Therefore, a design constraint has been placed on the steam line break analyses that the core must remain subcritical throughout the event.

Immediately following the reactor trip, the delayed neutron precursors continue to provide an additional source of neutrons that can cause fission. This source of neutrons decreases based on the half-life of each species. However, while significant numbers of delayed neutrons are still being emitted, the core will experience subcritical multiplication. During subcritical multiplication, power production is occurring but the reaction is not self-sustaining. Analyses have shown that this condition is acceptable and will not lead to fuel failures. Consequently, post-trip power production due to subcritical multiplication is allowed. However, continued power production in a self-sustaining reaction, i.e., a critical condition, is not allowed.

Each cycle, the acceptability of the core design with respect to steam line breaks is determined by a review of the moderator and Doppler reactivity coefficients, as described in section 8.1.1. Key plant parameters that can significantly affect the outcome of the steam line break analysis are also reviewed to ensure that the values incorporated into the analysis remain bounding. These parameters are the initial secondary steam generator inventories and the main and emergency feedwater system performance.

6. The peak primary system pressure shall not exceed the 2750 psig safety limit, except for ATWS events.

The RCS pressure boundary is the second barrier between the fuel radioactivity and the environment. As such, a breach of this boundary can lead to higher radioactivity releases to the environment. The plant is designed to the applicable requirements of the ASME Code and ANSI standards, which define a safety limit of 110% of the design pressure. For an RCS design pressure of 2500 psig, the safety limit is defined as 2750 psig. By maintaining pressures below the safety limit, the design margins in the stress calculations ensure that the pressure boundary will remain intact and the radioactivity releases to the environment will be limited. The plant setpoints and system performance characteristics are reviewed for each cycle to ensure that the licensing analysis assumptions remain bounding.

The ATWS events are exempt from the 2750 psig safety limit because they define a set of accidents that assume a non-mechanistic failure of the RPS, a safety-grade system, to trip the reactor. The nature of the failure assumptions associated with an ATWS event allow the maximum RCS pressure acceptance criterion to be increased. With the addition of DSS and AMSAC functions at each plant, the failure of the RPS should not result in peak pressures greater than 3250 psig. This limit is considered to be conservative for the expected frequency of an ATWS event.

7. The peak secondary system pressure shall not exceed 110% of the design pressure.

The secondary system has also been designed in accordance with the applicable sections of the ASME Code and ANSI standards. For a secondary system design pressure of 1050 psig, the safety limit is 110% of this pressure, or 1155 psig. The main steam safety valves, in conjunction with the RPS, provide the over-pressure protection for the secondary system pressure boundary. By maintaining the secondary pressure less than the safety limit, the failure of the secondary system can be prevented and the heat removal capability through the steam generators can be maintained.

The limiting event with respect to secondary system over-pressurization is a turbine trip event from full power. The major parameters that affect the peak secondary system pressure for this and other events are reviewed each cycle to ensure that the assumptions used in the licensing analysis remain bounding. The major parameters that can affect the secondary system pressure response are the nominal main steam safety valve lift setpoints, allowed lift setpoint tolerances and flow capacities, the RPS high pressure trip setpoint and the core power level.

8. Doses for Condition III and Condition IV events shall be less than the 10CFR100 limits.

The common parameter used to assess the risk to the public is the calculated offsite dose. The cycle-specific dose calculations account for the performance of each of the barriers between the radioactivity source (the fuel) and the environment. The calculations also consider other factors, such as meteorological conditions, that can affect the dose calculations. As required by 10CFR50, the predicted doses for all Condition III and IV events must be less than the limits provided in 10CFR100. The predicted doses for the events evaluated in the plant SARs are well below the allowed limits, with conservative assumptions applied in the evaluations. The assessment is consistent with NSAC-125.

SECTION 9 LOSS-OF-COOLANT ACCIDENT EVALUATION

LOCAs are hypothetical accidents that result from the loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system. The RCS pipe breaks considered range in size from the smallest breaks that exceed the makeup capacity up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the RCS. Analyses for a given plant consider applicable fuel assembly designs, plant boundary conditions and modifications, and additional regulatory issues that may arise.

In general, the most likely change in a fuel reload that could impact the LOCA analysis is a change in the peaking, which can affect the initial core power distribution. This change is controlled within a set of maximum allowed LOCA kW/ft limits, defined to represent initial condition requirements for the LOCA analysis. (These should not be confused with kW/ft limits based on CFM or transient strain criteria used in the determination of the RPS setpoints.) The LOCA limits thus defined are used as input to the reload safety evaluation, which ultimately determines the allowed rod position and axial power imbalance limits for plant operation. Thus, by definition, the LOCA limits are accounted for in the core design.

LOCA analyses are performed based on an approved EM developed according to Appendix K of 10CFR50. The analyses are performed within all limitations and restrictions imposed via the SERs on the EM and its associated computer codes.

9.1 Analysis Criteria

The results of the LOCA analyses are compared against the criteria listed in 10CFR50.46 to define the maximum allowable LHR limit as a function of axial core elevation. The five criteria of 10CFR50.46 are listed below.

1. The calculated maximum fuel element cladding temperature shall not exceed 2200 °F.
2. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.
3. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.
4. Calculated changes in core geometry shall be such that the core remains amenable to cooling.
5. After any successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

9.2 Analysis Methods

There are currently two approved EMs that may be used for licensing application to B&W-designed plants. The original LBLOCA and SBLOCA licensing analyses for the B&W plants were based on blowdown analyses performed with the CRAFT2 computer code, as described in references 67 and 68. More recent LBLOCA and SBLOCA licensing analyses for the B&W plants have been performed based on the RELAP5/MOD2-B&W (RELAP5) computer code, as described in reference 69. These EMs may be used separately or in combination to cover all fuel types over the entire spectrum of breaks that must be considered in the EM analyses. Each of these EMs is discussed briefly in Sections 9.2.1 and 9.2.2.

All currently operating B&W plants have converted their licensing basis for the LOCA LHR limits used to define the core operating limits to the RELAP5 EM discussed in Section 9.2.2. However, previous sensitivity studies and results of the CRAFT2-based EM licensing cases may be used to define plant containment and equipment qualifications, long-term core cooling, and the bases for emergency operator procedure guidance. Additionally, some historical CRAFT2 sensitivity studies have been applied in the definition of the RELAP5 EM methods used. Therefore, continued reference to this approved EM may be made for justification of certain licensing applications not directly related to defining the LOCA LHR limits.

9.2.1 CRAFT2 EM

The analysis methodology of the CRAFT2-based EM is described in reference 67 for LBLOCA and reference 68 for SBLOCA analyses. The CRAFT2-based EM utilizes a suite of computer codes in order to predict the LOCA consequences. Steady-state fuel data for UO_2 and gadolinia fuel is provided to the CRAFT2-based EM by approved fuel codes, which are discussed in Section 9.2.3.

For LBLOCA analyses, the CRAFT2 code (reference 70) calculates the system hydrodynamics and core power generation during blowdown. The REFLOD3 code (reference 71) is used to determine the length of the RV lower plenum refill period and the flooding rates during reflood. The CONTEMPT code (reference 72) analyzes reactor building pressure. Finally, THETA1-B (reference 73) is used with output from CRAFT2, REFLOD3 and CONTEMPT to determine the PCT response.

For SBLOCA analyses, the CRAFT2 code is used to predict the hydrodynamic behavior of the RCS. If CRAFT2 predicts that the core will be covered with liquid throughout the transient, no core heat-up is predicted, and therefore, no thermal analysis is required and compliance with 10CFR50.46 is ensured. Otherwise the FOAM2 computer program (reference 74) is used to determine the mixture height within the reactor core and the thermal response of the hottest fuel pin is calculated using the THETA1-B computer program.

LOCA licensing calculations for older fuel batches currently located in the spent fuel pool may have been performed with the CRAFT2-based EM. In the event that an assembly in this category is reinserted into a new core design being licensed based on the RELAP5-based EM, additional evaluations and/or analyses are performed if necessary to demonstrate acceptability of the allowed LOCA LHR limits with respect to the new EM and current or existing plant boundary conditions. These evaluations are typically based on sensitivity studies that consider important boundary conditions, such as tube plugging, power level, ECCS capacity, etc, in order to determine an applicable PCT or LHR limit penalty.

9.2.2 RELAP5 EM

The analysis methodology of the RELAP5-based EM (reference 69) centers on the RELAP5/MOD2-B&W (RELAP5) computer code (reference 75). However, several other computer codes may also be used to provide input to RELAP5 in order to obtain the prediction of the PCT. Steady-state fuel data for UO_2 and gadolinia fuel is provided to the RELAP5-based EM by approved fuel codes, which are discussed in Section 9.2.3. The first three 50.46 criteria are evaluated on a fuel-specific basis. The coolable geometry is evaluated with a combination of generic and fuel-specific analyses as described in Section 4. The long-term core cooling is generic and relies on operator actions based on the plant specific EOPs.

The relationships between the computer codes for evaluation of the LBLOCA and SBLOCA transients are discussed in detail in the BWNT LOCA EM (reference 69), and are summarized here to provide reference to the individual codes that make up the EM. The CONTEMPT (reference 72) computer code defines the containment pressure response during the LBLOCA transient based on the limiting ECCS injection conditions. This response, which is determined via iteration with the RELAP5 and REFLOD3B codes, is input to the RELAP5 and REFLOD3B analyses for evaluation of the LBLOCA transient. This code is generally not used for SBLOCA evaluations because the break remains choked and the results are independent of the containment pressure response. In these cases, a maximum containment pressure is used. The RELAP5 computer code (reference 75) calculates the system thermal-hydraulics, core power generation, and cladding temperature response during the blowdown portion of the LBLOCA transient, and for the entire SBLOCA transient. The initial conditions input to the REFLOD3B refill and reflood system thermal-hydraulic computer code (reference 76) represent the end of blowdown conditions from the RELAP5 analysis of the LBLOCA transient. REFLOD3B determines the core inlet fluid temperature and the core reflooding rate. Finally, the BEACH computer code (reference 77), which is equivalent to the RELAP5 code with the fine mesh rezoning option activated, determines the cladding temperature response during the reflood period with input from REFLOD3B. Modifications to the EM and the associated computer codes for application to the M5 cladding material was approved in reference 18. Approval for the SBLOCA reactor coolant pump modeling utilized when simulating an operator action to trip the pump is contained in References 79 and 80.

All LOCA analyses are performed in accordance with the limitations and restrictions placed on the EM and the individual codes, and this is verified during the execution and documentation of the analyses. This includes any approvals and limitations on the LOCA EM that may be included in associated topical reports. Additionally, any EM error corrections or changes made through the reporting requirements of 10CFR50.46 Section (a)(3)(ii) and/or through resolution of Preliminary Safety Concerns (PSCs) are considered or included in all licensing cases used to define the LOCA LHR limits that validate the core operating limits. In general, any EM corrections and changes made through the 10CFR50.46 process add conservatism to the analyses.

9.2.3 Steady-State Fuel Data Input to LOCA EMs

Steady-state fuel rod data, such as local volumetric fuel temperature as a function of LHR, fuel rod internal gas pressure, gap gas composition, and fuel rod dimensions and characteristics, are determined by an NRC-approved steady-state fuel rod computer code. The TACO3 (reference 24) fuel rod design code is one of the codes that may be utilized to provide steady-state fuel rod input data for UO_2 fuel with either Zircaloy-4 or M5 cladding. The TACO3 predicted best-estimate fuel temperatures are adjusted by an uncertainty factor to ensure that a 95%/95% upper bound tolerance on the volume average temperature is used in the LOCA applications. The EM and steady-state fuel code provide information used to define the uncertainty factors that are applied, since the value of the 95%/95% uncertainty factor is dependent on the bundle or pin that is modeled. Reference 9 approves the use of TACO3 for fuel-rod analysis up to a burnup of 62 GWd/mtU, provided that a bias factor is used to account for the reduced fuel thermal conductivity at burnups greater than 40 GWd/mtU. This burnup-dependent fuel thermal-conductivity bias factor increases the 95%/95% uncertainty factor applied to the TACO3 predicted fuel temperatures input in the EM analyses. The GDTACO (reference 28) fuel rod design code also predicts best-estimate fuel temperatures that are augmented by a 95%/95% upper bound tolerance factor for use in LOCA applications. GDTACO may be utilized for analysis of gadolinia fuel with either Zircaloy-4 or M5 cladding. The fuel thermal-conductivity bias applied to the TACO3 volume-averaged fuel temperatures is also applied to the GDTACO results at burnups greater than 40 GWd/mtU. If no impact on operational limits is expected, fuel data for higher concentrations of gadolinia may be optionally selected to conservatively bound those for a lower concentration.

9.3 Generic LOCA Evaluations

LOCA analyses are generally performed based on a full core simulation of a single fuel assembly design with UO_2 fuel and result in the definition of an envelope of allowed LOCA LHR limits as a function of core elevation and time-in-life (TIL). For those plants that utilize different fuel assembly types, each assembly type has its own set of full core LOCA LHR limits. The 10CFR50.46 criteria summarized above have been used for determining the acceptability of the consequences of a given accident for each assembly type. However, possible combinations of fuel assembly types are also evaluated by performing a mixed-core analysis, if necessary due to hydraulic resistance differences.

The mixed-core analysis determines any LHR limit or PCT penalty that must be applied to the results of the full core analysis. A penalty may be necessary due to the flow redistribution during the transient arising from differing fuel assembly hydraulic designs, including but not restricted to, differing grid or end fitting resistances, as well as different fuel rod geometries. Mixed-core analyses specifically for fuel rod material changes (specifically Zircaloy-4 to M5) are not necessary, as described in the approved M5 cladding topical report (reference 18). Mixed-core LOCA analyses are primarily necessary for flow dominated transients such as the LBLOCA. The SBLOCA transient evolves much slower and core flow is relatively stagnant during the core uncovering phase. Therefore, mixed-core SBLOCA analyses are not typically needed because there is no flow diversion potential. Nonetheless, FANP will evaluate mixed-core conditions for SBLOCAs if there are significant future evolutionary fuel assembly design changes that could potentially cause flow diversion during SBLOCA transients.

LOCA analyses for gadolinia pins are also performed to determine the reduction in allowable LHR limit necessary to account for the decrease in the fuel thermal conductivity compared with a UO_2 fuel rod of the same design. These evaluations are typically performed only at those elevations that have the limiting LOCA margin in the core power distribution analyses. Therefore, analyses that model the gadolinia fuel steady-state data are generally performed with axial peaking at the core inlet and sometimes for the core exit elevations. All burnup ranges and corresponding fuel thermal conductivity inputs based on GDTACO are supplied to RELAP5 in order to determine the LHR limit for the gadolinia pins. Analyses may be performed for each gadolinia concentration, or results obtained for a higher concentration may be conservatively applied to a lower concentration of gadolinia. The gadolinia LHR limit reduction is applied to the UO_2 LOCA LHR limits in order to define the envelope of maximum allowed LOCA LHR limit versus axial elevation and TIL for each analyzed gadolinia concentration.

Additional generic analyses and/or evaluations are performed as necessary to consider changes in plant operating conditions, such as degraded plant ECCS injection capacity. The analyses are performed according to the methods described in Section 9.2 and may define new LOCA LHR limits, or may impose penalties (LHR limit or PCT increase) on the results of a previous set of analyses.

Finally, special assembly conditions are evaluated as those conditions are identified. In some cases, generic studies may be performed that may define checks that must be validated on a cycle-specific basis. In other cases, cycle-specific evaluations are performed to justify operation for a single cycle. Examples include, but are not limited to, stainless steel replacement rods (approved in reference 23), slipped grids, loose rods and LTAs.

9.4 Application Of Generic Evaluations To Cycle-Specific Plant Conditions

The plant design and core arrangement are evaluated each fuel cycle to ensure that the results of the licensing LOCA analyses remain bounding for the cycle-specific plant conditions. This includes the consideration of LHR or PCT penalties associated with a mixed-core configuration, or other special considerations or plant changes.

Fuel manufacturing changes in a given design are evaluated for each fresh batch of fuel to assure that the generic LOCA analyses remain applicable. This evaluation includes the examination of initial pin pressures, plenum volumes and fuel densities, and the calculated fuel pin pressures and temperatures over the life of the fuel assembly. In each case, the predicted fuel pin pressure and temperature response to burnup is compared with the boundary conditions used in the generic LOCA analysis to determine the acceptability of the fuel. Since the dynamics of the fuel during LOCAs involve both the internal pin pressure and temperature, these two parameters are evaluated together. Thus, variations in the pin pressure and fuel temperature are used to demonstrate the acceptability of the combination of the two. This approach uses existing sensitivity studies that define the relationships between these parameters. Based on these analyses, the acceptability of a given combination of pressures and temperatures is determined. A fuel design that leads to conditions not bounded by existing analyses requires new analyses or evaluations (either cycle-specific or generic) to be generated in order to define a new maximum LOCA LHR limit or to determine PCT penalties that are applied to previous analyses.

In addition to defining the maximum allowable LOCA LHR limit for each fuel design, cycle-specific checks are performed to assure that the results of the generic analyses remain applicable. The checks are related to resolutions for LOCA-related PSCs, and other cycle-specific conditions, such as any planned EOC RCS average temperature reduction maneuver, LTAs, etc. Specific limitations related to the approval of use of stainless steel replacement rods (reference 23) have been generically dispositioned and checks are made on a cycle-specific basis to ensure the applicability of the generic results.

As long as the acceptance criteria can be met for the combination of fuel assembly types for each reload fuel cycle, the cycle design is considered acceptable. Power distributions for the reload core design are evaluated at the limits of normal operation as described in Section 5.3 to ensure that the maximum LHR does not exceed the limits established by LOCA analyses performed to demonstrate compliance with the 10 CFR 50.46 criteria.

SECTION 10 LIMITED SCOPE HIGH BURNUP LEAD TEST ASSEMBLIES

10.1 Use of Lead Test Assemblies

WCAP-15604-NP, Revision 2-A (reference 78) provides generic guidelines for the irradiation of a limited number of Lead Test Assemblies (LTAs) to rod burnups greater than the current licensed lead rod average burnup limit. The NRC staff has reviewed this report and all conclusions apply to the entire commercial nuclear power industry, i.e., all pressurized water reactors (PWRs) and boiling water reactors (BWRs). The generic LTA guidelines will ensure uniformity in data collection, make the evaluation of new fuel properties or limits more predictable, ensure a structured process for data feedback to the NRC, and provide to fuel vendors and licensees a uniform process for implementing LTA programs. The inclusion of WCAP-15604-NP, Revision 2-A into BAW-10179 satisfies the requirement to incorporate this WCAP report explicitly into the licensee's Technical Specifications by virtue of its reference in BAW-10179.

10.2 Definition of a Lead Test Assembly

A limited scope LTA is a fuel assembly that is based on a currently available design and is capable of reaching higher burnups than currently used. The fuel cladding material is an NRC-approved cladding material. The assembly will receive pre-characterization prior to undergoing exposure in the "test" cycle that would permit the assembly to exceed current burnup limits. The fuel assembly shall be evaluated against and must meet all current design criteria even though the current analytical methodologies may not be approved for use at the higher burnups.

10.3 Conditions for Limited Scope High Burnup LTA Program

The following conditions for a limited scope high burnup LTA program must be met per the NRC Safety Evaluation (SE) referenced in WCAP-15604-NP, Revision 2-A:

1. If the COLR analytical methods listed in the licensee's Technical Specifications were approved up to a specified burnup limit, a license amendment is required to add this topical report to that list in order for the licensees to be able to use WCAP-15604-NP, Revision 2-A.

2. The number of fuel assemblies with fuel rods exceeding the current lead rod average burnup shall be limited to a total of nine in PWRs. No fuel rods shall exceed peak rod burnups greater than 75 GWd/mtU.

3. The fuel shall be typical production fuel and be pre-characterized before operation above the current lead rod average burnup limit. The fuel may also be an LTA that was characterized during fabrication and was designed to test aspects of the fuel assembly but was not initially identified as a high burnup LTA. The latter fuel shall be pre-characterized before operation

above the current lead rod average burnup limit. The fuel clad material is a NRC-approved clad material.

4. The pre-characterization of the fuel shall consist of at least the following examinations: clad oxidation, rod/assembly growth, and visual examinations for PWRs.

5. The post-irradiation examinations of the fuel shall consist of at least the following examinations: clad oxidation, rod/assembly growth, and visual examinations for PWRs. Current or modified fuel performance methods and codes shall be used.

6. The fuel shall be evaluated against and must meet all current design criteria even though the current analytical methodologies may not be approved for use at the higher burnups.

7. For all fuel rods in the LTAs, the predicted oxidation shall be less than 100 microns on a best-estimate basis with prediction of no blistering or spallation based on current data.

8. A licensee using the limited scope high burnup LTA program shall submit two reports to the NRC for information.

   The first report shall be a notification of intent to irradiate LTAs above the current maximum burnup limit. It shall contain at least the following information:

   - Licensee name
   - Plant name
   - Cycle and date when the LTA shall be inserted
   - Number of LTAs
   - Location of the LTAs
   - Anticipated pre- and post-cycle burnups for each LTA
   - Purpose of the LTAs
   - Estimated dates for pre- and post-irradiation characterizations or the results of the pre-characterization and an estimation of the date for the post-irradiation characterization
   - Estimated date of the second report
   - Statement that the LTAs will not be irradiated if Conditions 6 and 7 are not met or if the pre-characterization examinations show anomalous results

The second report shall give the results of the pre- and post-irradiation examinations. It shall consist of at least the following information:

- Licensee name
- Plant name
- Assembly identification number
- Specific measurements – actual data and predictions
- Comment section

10.4 Nuclear Design and Safety Analysis Considerations

10.4.1 Core Loading Pattern Development

As cited in Condition 2 above, the maximum number of LTAs per cycle per core will be limited to nine assemblies. However, the NRC has recognized that to determine if an LTA meets the need for which it was designed, it must experience the same limiting conditions as other fuel in the reactor and should not be restricted in power or core location except as needed to meet design criteria. The unique aspect of the LTAs is that they are normal production fuel assemblies that will fall into two general categories. The LTAs will either be fuel assemblies that are reinserted for additional exposure after achieving a burnup instead of being discharged or fuel assemblies that have normal in-core residence times, but are positioned in-core so that the power level results in the highest current burnup limit being exceeded. The maximum lead rod average burnup that these limited scope LTAs would experience is 75 GWd/mtU.

10.4.2 Safety Analysis

The inclusion of WCAP-15604-NP, Revision 2-A into BAW-10179 satisfies the requirement to incorporate this WCAP report explicitly into the licensee's Technical Specifications by virtue of its being referenced in BAW-10179. BAW-10179 defines the safety criteria and methodologies used for reload safety analyses. These same safety criteria and methodologies will be used to analyze a core containing high burnup LTAs. Hence, the inclusion of WCAP-15604-NP, Revision 2-A in BAW-10179 will preclude the submittal of a License Amendment Request by the licensee to use high burnup LTAs as long as the conditions defined in Section 10 are met.

As part of the safety analysis, an assessment must be made of the models that have been reviewed and approved by the NRC for the purpose of evaluating the performance of the LTAs beyond current burnup limits. The analytical models used to evaluate the performance of the LTAs beyond current burnup limits may need to be modified versions of the models reviewed and approved by the NRC. In some cases, conservatism may be added, as appropriate. If available data indicates that the approved models are appropriate, then no modifications to the approved models will be necessary. The revised models would be used only for the limited scope high burnup LTAs and not for any other assemblies in the core. The justification of the model revisions will be documented and available for NRC review in accordance with the 10 CFR 50.59 criteria.

10.5 Evaluation of Limited Scope High Burnup LTAs

Section 10 summarizes the requirements for a limited scope high burnup LTA program. Sections 1 through 9 of this topical report describe the methods and models used to evaluate core and fuel assembly performance up to the current burnup limits. These same methods and models delineated in Sections 1 through 9 will be used to evaluate core and LTA fuel assembly performance in a limited scope high burnup LTA program. All model revisions made to show satisfactory LTA performance must meet the requirements discussed in Section 10.4.2.

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