ATTACHMENT 3

"TMI-1 Reload Design and Setpoint Methodology Topical Report 092, Revision 0, Dated January 23, 1996"

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TMI-1 RELOAD DESIGN AND SETPOINT METHODOLOGY

TOPICAL REPORT 092

(REV. 0)

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ABSTRACT

This report describes GPU Nuclear Corporation's reload design methodology and corresponding safety criteria to be applied for Three Mile Island Unit-1 (TMI-1). Included in this report are descriptions of fuel cycle design, nuclear design, core thermal-hydraulic design, maneuvering analysis, setpoint methodology, and accident analyses. Also included are comparative analyses for the setpoint methodology with existing B&W Fuel Company (BWFC) generated TMI-1 Cycle 10 reload design setpoints. The overall good agreement in these comparisons demonstrates GPUN's ability to perform reload design and setpoints analyses for TMI-1.

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TABLE OF CONTENTS

1.0 INTRODUCTION 9
1.1 Purpose 10
1.2 Report Structure 10
1.3 Reload Design Process Overview 11
1.4 Safety Criteria Overview 16

2.0 NUCLEAR DESIGN 20
2.1 Analysis Methods 20
2.2 Preliminary Fuel Cycle Design 21
2.3 Final Fuel Cycle Design 21
2.3.1 Acceptance Criteria 22
2.3.1.1 Radial Power Distribution 22
2.3.1.2 Maximum Discharged Fuel Assembly/Rod Burnups 23
2.3.1.3 Total Control Rod Worths 23
2.3.1.4 Dropped Control Rod Worth 23
2.3.1.5 Ejected Control Rod Worth 23
2.3.1.6 Shutdown Margin 24
2.3.1.7 Moderator Temperature Coefficients and Deficits 24
2.3.2 Final Fuel Cycle Design Depletion 25
2.3.2.1 Power Distribution Analysis 25
2.3.2.2 Maximum Discharged Fuel Assembly/Rod Burnups 26
2.3.3 Control Rod Worths 26
2.3.3.1 Control Rod Grouping 27
2.3.3.2 Individual Group and Total Control Rod Worths 28
2.3.3.3 Ejected Control Rod Worth 29
2.3.3.4 Dropped Control Rod Worth 30
2.3.4 Shutdown Margin 31
2.3.5 Reactivity Coefficients and Deficits 32
2.3.5.1 Doppler Coefficient 32
2.3.5.2 Moderator Temperature Coefficient 33
2.3.5.3 Total Temperature Coefficient 34
2.3.5.4 Power Coefficient and Power Deficit 35
2.3.6 Boron Related Parameters 36
2.3.7 Xenon Worth 37
2.3.8 Kinetics Parameters 37
2.4 Startup Tests 38

3.0 CORE THERMAL-HYDRAULIC DESIGN 42
3.1 Design Criterion and Design DNB Limit 42
3.2 Reference Design Thermal-Hydraulic Parameters 43
3.2.1 Core Inlet Conditions (Core Flow and Inlet Temperature) 43
3.2.2 Core Bypass Flow 45
3.2.3 System Pressure 46
3.2.4 Power Distribution Factors 46
3.2.5 Engineering Hot Channel Factors 47
3.2.6 Fuel Rod Bowing and Densification Effects 47

3.3 Reference Design DNBR Analysis 48
3.4 Determination of Pressure-Temperature Core Protection Limits 49
3.5 Transient Core Thermal-Hydraulic Analysis 51
3.5.1 Flux/Flow Protective Limits 52
3.6 Maximum Allowable Peaking (MAP) Limits 55
3.6.1 Determination of RPS MAP Limits 56
3.6.2 Determination of LCO MAP Limits 59
3.6.3 Verification of MAP Margins 60

4.0 MANEUVERING ANALYSIS 73
4.1 Power Distribution Analysis 74
4.1.1 Control Rod Scan at Various Cycle Depletion Points 74
4.1.2 Control Rod Scan in the Design Power Maneuver 75
4.2 Margin Evaluation for the Setpoint Determination 77
4.2.1 Peaking Augmentation Factors 77
4.2.2 Margin Analysis Against CFM, DNBR and LOCA Limits 78

5.0 SETPOINT METHODOLOGY 81
5.1 Introduction 81
5.2 Protection System Maximum Allowable (PSMA) Trip Setpoints (Pressure-Temperature Trip Setpoints) 82
5.2.1 Variable Low Pressure (VLP) Trip 83
5.2.2 High RCS Pressure Trip 85
5.2.3 Low RCS Pressure Trip 87
5.2.4 High RCS Temperature Trip 88
5.3 High Flux Trip 89
5.4 Protection System Maximum Allowable (PSMA) Setpoints for Axial Power Imbalance 90
5.4.1 Flux/Flow Trip 90
5.4.2 Axial Power Imbalance Core Protection Safety Limits 91
5.4.3 PSMA Setpoints for Axial Power Imbalance 93
5.5 Limiting Conditions for Operation (LCO) Alarm Limits 96
5.5.1 Power Imbalance Alarm Limits 98
5.5.2 Control Rod Insertion Limits 99
5.5.3 Core Quadrant Power Tilt Limit 100

6.0 ACCIDENT ANALYSIS REVIEW 119
6.1 Overview 119
6.2 Discussion of Individual Accidents 120
6.2.1 Moderator Dilution Accident 120
6.2.2 Rod Withdrawal Accident 121
6.2.3 Startup Accident 122
6.2.4 Uncompensated Operating Reactivity Change 123
6.2.5 Rod Ejection Accident 123
6.2.6 Loss of Coolant Flow 125
6.2.7 Loss of Electric Power 125
6.2.8 Steam Line Break 127
6.2.9 Cold Water Accident 127

6.2.10 Stuck-Out, Stuck-In, or Dropped-In Control Rod Accident 128
6.3 Radiation Dose Evaluation 130

7.0 CONCLUSION 133

8.0 REFERENCES 134

APPENDIX 136
A.1 Introduction 136
A.2 Discussion of Setpoint Results 136
A.3 References 139
[ ] 140
141
142
143
144
145
146
147
148
149

LIST OF TABLES

Table 1.1 TMI-1 Core Design Data 18
Table 3.1 Design Thermal-Hydraulic Conditions 61
Table 3.2 Core Inlet Flow Factors 62
Table 3.3 Typical Limiting Thermal-Hydraulic Conditions for P-T Analysis 63
Table 3.4 Typical Values of Total Flux Measurement Errors for Flux/Flow Instrument String 64
Table 3.5 Initial Conditions for Transient DNBR Analysis 65
Table 3.6 Limiting Thermal-Hydraulic Conditions for MAP Analysis 66
Table 5.1 Typical RPS Trip Instrument Errors and Response Time 102
Table 5.2 Ejected Control Rod Worth Limits 103
Table 5.3 Calculation of Shutdown Margin and Allowable Inserted Control Rod Worth 104
Table 5.4 Determination of Error-Adjusted Quadrant Tilt Alarm Setpoints 105
Table 6.1 Key Safety Parameters for Accident Analysis Review 131
Table 6.2 Safety Parameter Matrix Against Affected Safety Analyses 132

LIST OF FIGURES

Figure 1.1 Reload Analysis Flow Chart 19
Figure 2.1 Typical Control Rod Group Configuration 40
Figure 2.2 Typical Integral Rod Worth Curve 41
Figure 3.1 Reference Design Axial Power Shape (1.65 Cosine With Tails) 67
Figure 3.2 Typical Core Protection Safety Limit 68
Figure 3.3 Typical Core Protection Safety Bases 69
Figure 3.4 Determination of Flux/Flow Protective Limits 70
Figure 3.5 A Typical Set of MAP Limits 71
Figure 3.6 P-T State Points for the RPS MAP Analysis 72
Figure 5.1 Determination of Protection System Maximum Allowable Setpoints 106
Figure 5.2 Relationship of Trip Setpoint To Accident Analyses and Instrument Errors 107
Figure 5.3 Typical Power-Imbalance-Flow Trip Setpoints 108
Figure 5.4 Typical Power-Imbalance-Flow Safety Limits 109
Figure 5.5 Determination of Axial Offset Limit Based on CFM and DNB Margins 110
Figure 5.6 Power Imbalance Detectability Envelope 111
Figure 5.7 Typical LOCA Limited Linear Heat Rate 112
Figure 5.8 Typical Power Imbalance Alarm Limits 113
Figure 5.9 Typical Control Rod Insertion Limits 114
Figure 5.10 LOCA Margin Contour Curve 115
Figure 5.11 Core Offset Contour Curve 116
Figure 5.12 Determination of Allowable Inserted Control Rod Worth Based On Ejected Control Rod Worth Limit 117
Figure 5.13 Integral Control Rod Worth Curve and Determination of CRI Limits 118

1.0 INTRODUCTION

The TMI-1 reactor core is designed to operate at the reference design core power of 2568 MWth with sufficient design margins to accommodate transient operation and instrument error without damage to the core. The TMI-1 reactor core design characteristics are provided in Table 1.1 and are described in detail in Section 3 of Reference 1. The reactor is operated with full length control rods out, in a feed and bleed mode. Core reactivity control is mainly by soluble boron shim, burnable poison rod assemblies (BPRAs), and gadolinia-bearing fuel pins.

The reactor is refueled at an interval of approximately 24 months. The refueling of a reactor consists of removing a portion of the core and replacing it with fresh fuel assemblies and, if needed, with previously burned fuel assemblies. Each reload design may result in changes to the operating characteristics of the core. The fresh fuel has a noticeable effect on both nuclear and thermal-hydraulic behavior of the core. These effects would consist of changes in power distribution, core reactivity, neutronics parameters, control rod worth, and departure from nucleate boiling (DNB) conditions. The reload fuel may also affect the input parameters to the plant safety analyses of the reference cycle. All of these effects must be considered for the safe operation of the plant. Therefore, it is important to perform the reload safety evaluation to confirm the validity of the existing safety analyses. The existing safety analyses are defined as the reference safety analyses and are intended to be valid for all plant cycles by selecting input parameters which will bound those values of all subsequent cycles.

TMI-1 cycle-dependent variables have been removed from TS (References 2 through 4) and are included in the Core Operating Limits Report (COLR). The majority of these cycle-dependent variables were from Section 3 of the TMI-1 TS, which contains the Limiting Conditions for Operation (LCO), and Section 2, which contains the cycle-dependent protective and maximum allowable setpoints.

Topical reports for the computer codes to be utilized in the GPUN reload analyses have been submitted previously for approval. They are CASMO-3/SIMULATE-3 (Reference 5) for nuclear analysis, RETRAN-02 (Reference 6) for system transient analysis, and VIPRE-01 (Reference 7) for core thermal-hydraulic analysis.

The fuel vendor will perform fuel thermal/mechanical analysis and LOCA analysis, as necessary.

1.1 Purpose

The purpose of this topical report is two-fold:

1. To describe the GPUN reload design methodology applied to nuclear, thermal-hydraulic, and safety analyses, and

2. To present setpoint methodology including safety and acceptance criteria to determine various safety limits, reactor trip setpoints and alarm limits.

The GPUN reload methodology will be applied to the licensing analyses for the future TMI-1 fuel cycles. This topical report will be referenced in Section 6.9.5 of the TMI-1 TS.

1.2 Report Structure

The methodologies provided in this report are applicable to the reload fuel currently supplied by B&W Fuel Company (BWFC) for the 177 fuel assembly class of plants. Future reload core designs that use fuel designs which are compatible with the methods and models described herein, will

reference this report as their approved licensing basis.

This report describes the GPUN methods that are applied for each reload fuel cycle. The remainder of Section 1 provides an overview of the reload design process and safety criteria. Section 2 describes the nuclear design aspects of reload analysis which consists of fuel cycle design and core physics parameter development. Core thermal-hydraulic design methodology is described in Section 3. Section 4 describes the maneuvering analysis methodology, which consists of power distribution analyses considering all the possible modes of power operation including power load swing and rod positions from beginning-of-cycle (BOC) to the end-of-cycle (EOC). Section 5 provides a discussion of setpoint methodology, which include LCO alarm limits and reactor protection system (RPS) trip setpoints. The accident analyses are described in Section 6. In Appendix A of this report, results of a demonstration analysis of the setpoint methodology are provided with comparisons against the vendor's results.

1.3 Reload Design Process Overview

The reload design is essentially a series of analyses with the objective to design the reload core in such a manner that the reactor can be operated at a specified power level for a specified number of days satisfying various safety criteria and other design constraints. Two cycles are of interest in a reload licensing analysis. The "reload cycle" is the upcoming cycle to be designed and evaluated. The "reference cycle" is the cycle to which specific "reload cycle" parameters are to be compared. The appropriate reference cycle parameters could therefore be either from the current operating cycle or a bounding cycle, including parameters in the approved Final Safety Analysis Report (FSAR).

In the following paragraphs, a brief overview of the reload design criteria and major elements of the reload design process are provided. Detailed discussions of design methods including specific safety criteria are given in subsequent sections of this report. The following reload design criteria are the requirements based on safety considerations, operational considerations, and fuel economics:

1. The BOC core excess reactivity of the reload cycle will be sufficient to enable safe, full power operation for the desired length of cycle.

2. Both LCO and RPS trip setpoints in either the TS or COLR should have adequate margin covering all possible operating conditions throughout the reload cycle so that sufficient operating flexibility can be maintained.

3. The fuel assemblies to be discharged at the EOC will attain maximum permissible burnup and minimal fuel content so that maximum fuel utilization can be achieved.

4. The predicted values of important core parameters for the reload cycle are conservatively bounded by the values of the reference cycle (or FSAR). If not bounded, reanalysis of applicable transients/accidents is performed.

5. The predicted power distributions for all possible power operation modes during the entire reload cycle will not exceed the specified thermal design limit and the LOCA-limited linear heat rate (LHR).

6. The integrity of the fuel assembly mechanical design should be maintained throughout the reload cycle and during the life of the fuel assembly.

Key elements of the reload design activities, as shown in Figure 1.1, are: (1) fuel cycle design, (2) nuclear analysis, (3) thermal-hydraulic analysis, (4) fuel thermal/mechanical analysis, (5) maneuvering analysis, (6) determination of setpoints and alarm limits for TS and COLR development, (7) safety analysis review, and (8) preparation of the reload report, COLR, and other relevant documentation.

The fuel cycle design (FCD) establishes the material quantities and core loading pattern for the fuel assemblies and burnable poisons (BPRAs and integral burnable poisons). The core loading pattern of fuel assemblies and core components is determined in such a manner that the specified criteria on power distribution, energy production, fuel burnup, and control rod worth requirements are satisfied.

In nuclear analysis, the important core physics parameters are determined based on the core loading pattern and cycle depletion, and are compared to the corresponding safety criteria and with those parameters of the reference cycle. These physics parameters consist of the moderator and Doppler reactivity coefficients, the critical boron concentration as a function of fuel burnup, ejected and stuck rod worth, total control rod group worth, shutdown margin, maximum LHR of the fuel rod at various elevations in the core, core excess reactivity, boron reactivity worth, xenon and samarium reactivity worth, and effective delayed neutron fractions. Other physics parameters are also calculated to enable a safe startup and operation of the cycle. Comparisons of calculated and measured data are also made during performance of startup testing and throughout core follow.

The thermal-hydraulic (T-H) design criterion is that the hot fuel rod in the core shall not experience DNB during both steady state operation and anticipated transients. Based on this criterion, the T-H analysis establishes the maximum allowable peaking (MAP) limits for various flow conditions, pressure-temperature limits, core pressure drop, the flux/flow protection, and the DNB ratio for the design overpower condition.

The fuel thermal analysis establishes the maximum permissible power density of a fuel rod to preclude centerline fuel melting during both steady-state and transient conditions. The thermal analysis needs to be performed only if there is a change in the fuel design.

The fuel mechanical analysis covers the mechanical performance of the fuel assembly, fuel rod, and control component assemblies: control rod assemblies (CRAs), axial power shaping rod assemblies (APSRAs), and burnable poison rod assemblies (BPRAs). The mechanical safety and design criterion address that the fuel and control system capabilities are no less than those assumed in the safety analyses. Main items considered in this criterion are maintaining a coolable fuel rod geometry and integrity, and keeping the CRA insertion path open. As mentioned earlier in this section, the fuel thermal/mechanical analysis is not included in this report and will be provided by the fuel vendor, as necessary.

The maneuvering analysis predicts three-dimensional fuel rod power distributions throughout the reload cycle considering all possible modes of reactor operation including plant maneuvers. The plant maneuvering includes control rod position changes during either a recovery operation from a scram and dropped rods or a power transient and consequent xenon oscillation due to an

unplanned load swing. These control rod movements will have effects on the radial and axial power distributions in the core. The power distribution data generated in the maneuvering analysis are used to determine the core safety limits, RPS trip setpoints, and LCO alarm limits.

The setpoint analysis consists of determining the core safety limits, RPS trip setpoints (also called limiting safety system settings - LSSS), and LCO alarm limits based on the nuclear and thermal-hydraulic characteristics of the reload core and applicable accident analyses. The associated safety criteria parameters are the centerline fuel melting (CFM) limit, DNBR limit, LOCA LHR limits, shutdown margin limit, and ejected rod worth limits.

The key parameters that have the greatest effect on the outcome of transients and accidents can typically be classified into three major areas: core thermal parameters, thermal-hydraulic parameters, and kinetic parameters including the reactivity feedback coefficients and control rod worths.

The safety impact of the parameter changes due to the reload design are evaluated against those in the reference cycle and the updated safety analysis report (USAR) to ensure that thermal performance during hypothetical transients and accidents is not degraded. A comparison of the radionuclide source inventory of the reload cycle is made against those in the USAR to confirm that there is no significant decrease in the margin to allowable dose limits.

The final phase of the reload design is the integration and documentation of the analyses results into the reload report. The cycle-dependent reactor trip setpoints and LCO alarm limits are incorporated into the COLR. Changes to the Technical Specifications may also be required.

1.4 Safety Criteria Overview

The safety philosophy upon which the design and safety analyses of nuclear plants are based is to prevent or to minimize the radiation release to the public through the fission product barriers. The fission product barriers are typically the fuel cladding, reactor vessel and primary system piping, containment building, and the plant exclusion boundary.

The safety criteria are provided in various parts of Title 10 of the Code of Federal Regulations (10CFR - Reference 8). 10CFR100 (Reference 9) provides the radiological dose criteria. The acceptance criteria for emergency core cooling systems are given in 10CFR50.46 (Reference 10). The general design criteria (GDC) in 10CFR50, Appendix A (Reference 10) provide guidance in assuring the various fission product barriers provide adequate public protection. Many of these criteria in the GDC of 10CFR50, Appendix A are given in terms of "specified acceptable fuel design limits" (SAFDLs) and are not given in numerical terms. These SAFDLs take many forms when quantified by different reactor vendors even though each vendor demonstrates compliance with the GDC in Appendix A.

In order to meet the GDC, design basis events (DBEs) are postulated to evaluate the challenge to the fission product barriers and the potential release of radioactivity from the plant. The event acceptance limits are specified for DBEs to meet the SAFDL guidelines. The event acceptance

limits used in this report are identical to those in Reference 4. The specific safety criteria and acceptance limits used in the nuclear, thermal-hydraulic and safety analyses are discussed in this report.

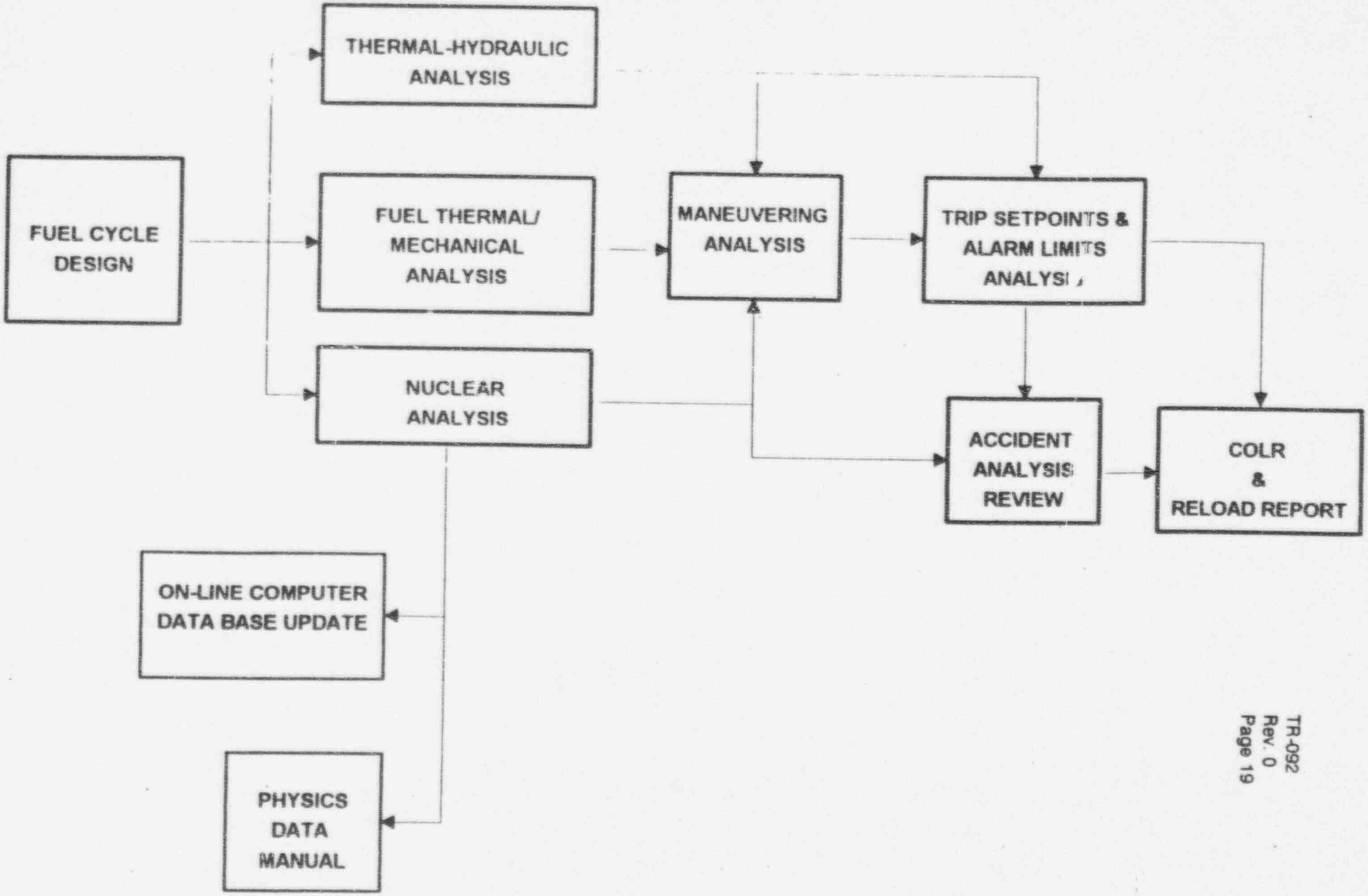
Table 1.1
TMI-1 Core Design Data

REACTOR:	VALUE
Design Heat Output, MWth	2568
Vessel Coolant Inlet Temperature, °F	555.7
Vessel Coolant Outlet Temperature, °F	602.3
Core Operating Pressure, psia	2200
Reactor Coolant Flow, % Design Flow	106.5

CORE and FUEL ASSEMBLIES:	VALUE
Total No. of Fuel Assemblies (FAs)	177
No of Fuel Rods per FA	208
No. of Control Rod Guide Tubes per FA	16
Fuel Rod Outside Diameter, inches	0.43
Cladding Thickness, inches	0.0265
Fuel Rod Pitch, inches	0.568
Fuel Assembly Pitch, inches	8.587

CORE COMPONENTS:	VALUE
No. of Control Rod Assemblies (CRA)	61
CRA Neutron Poison Material	Ag-In-Cd
No. of Axial Power Shaping Rod Assemblies (APSRAs)	8
APSRA Neutron Poison Material	Inconel
No. of Burnable Poison Rod Assemblies (BPRAs)	Varies per Cycle
BPRA Neutron Poison Material	B_4C

Figure 1.1 Reload Analysis Flow Chart



2.0 NUCLEAR DESIGN

The nuclear design process for a reload cycle normally consists of two phases; the preliminary fuel cycle design and the final fuel cycle design. During the initial phase, a preliminary loading pattern that meets the required energy output defined in the design specification is determined. Additional constraints placed on the preliminary cycle design include 2D assembly power peaking restrictions, fuel assembly burnup limits and neutron fluence minimization at critical reactor vessel welds. Once the preliminary loading pattern is established, the final design phase is begun to ensure that all nuclear-related safety parameters meet the design criteria to ensure safe reactor operation. This section describes the methods and criteria used in the two phases of nuclear design.

2.1 Analysis Methods

The methodology benchmarked in Reference 5 is used to simulate the TMI-1 reactor core. The lattice physics code CASMO-3 (Reference 11) is used to generate the cross sections of each fuel lattice in the reload design. The cross sections are tabulated in TABLES-3 (Reference 12) as functions of exposure, moderator temperature, soluble boron concentration, fuel temperature, presence of control rod assemblies, burnable poisons, moderator temperature history, boron concentration history and burnable poison history. SIMULATE-3 (Reference 13), an advanced two-group nodal code, is used to perform reactor physics calculations such as core power distributions, control rod worths and reactivity coefficients as described in Section 2.3. Unless specifically mentioned, a three dimensional quarter core with rotational symmetry is modelled in SIMULATE-3 using a 2x2 radial mesh per assembly and 24 equally spaced axial nodes.

2.2 Preliminary Fuel Cycle Design

The objective of the preliminary fuel cycle design is to determine the number of assemblies and their enrichments for the feed batch as well as an efficient core loading pattern which will meet the energy requirements for the reload cycle. The preliminary design is evaluated for reasonable radial power distributions. Burnable poisons, burnable poison rod assemblies (BPRAs) and/or integral burnable poisons are strategically located with varying poison loadings to keep radial power peaking equal to or below levels which have historically resulted in cycle designs with acceptable core operating limits. Other factors which are considered during the preliminary design phase include: maximum fuel assembly burnups for the different fuel types in the design, reload batch size, fuel enrichments, neutron fluence at the critical reactor vessel weld(s) and beginning-of-cycle boron concentration. Effects of the previous cycle's design burnup window are also considered during the preliminary design.

Typically, several preliminary designs with different design assumptions (e.g. cycle length) or philosophies (e.g., use of integral burnable poisons) are evaluated. The design which satisfies the acceptance criteria for radial power distribution, fuel discharge burnups, etc. and is most economically attractive will be designated for further evaluation in the final fuel cycle design phase.

2.3 Final Fuel Cycle Design

The preliminary cycle design chosen in Section 2.2 is evaluated in more detail during the final fuel cycle design phase. In addition to the acceptance criteria mentioned above, 2D fuel rod power peaking and burnups are evaluated using the SIMULATE-3 fuel rod power reconstruction

methodology. If necessary, small perturbations to the preliminary cycle design such as changing burnable poison concentrations, varying number and/or locations of integral burnable poison rods (if used), use of zone loaded fuel rod enrichments, and even reshuffling of the core loading pattern are made to satisfy fuel rod power peaking limits and maximum burnup limits. Once an acceptable final fuel cycle design is established, a detailed evaluation is performed to ascertain if all safety parameters are within the acceptance criteria of the safety analysis. The acceptance criteria evaluated during this design phase are described below as well as the methods used to determine the related physics parameters.

2.3.1 Acceptance Criteria

The acceptance criteria for the physics-related parameters of the final fuel cycle design are discussed below.

2.3.1.1 Radial Power Distribution

The maximum calculated steady-state fuel rod relative power density for the cycle shall be less than that required to meet the safety criteria and shall be sufficiently low to allow acceptable operating limits generated in the maneuvering analysis as described in Section 4. The radial power distribution is controlled by the fuel shuffle and by the placement of BPRAs, integral burnable poison, and/or zone loaded fuel rod enrichments.

2.3.1.2 Maximum Discharged Fuel Assembly/Rod Burnups

The discharged fuel assembly and fuel-rod burnups shall be less than the applicable limits for all respective fuel designs. Discharge burnups are controlled by cycle length, feed batch size and fuel shuffle pattern.

2.3.1.3 Total Control Rod Worths

The total control rod worth should be less than the value used in the reference safety analysis, currently 12.9% $\Delta k/k$. This is to ensure that the maximum reactivity insertion is not excessive during either the startup accident or control rod withdrawal accident.

2.3.1.4 Dropped Control Rod Worth

The maximum dropped rod worth shall be less than the value used in the reference safety analysis; currently 0.46% $\Delta k/k$. This is to ensure that the change in reactivity and power peaking disturbance are not excessive during a potential dropped control rod accident.

2.3.1.5 Ejected Control Rod Worth

Ejected control rod worths at hot full power (HFP) and hot zero power (HZP) shall be less than or equal to the values demonstrated to be acceptable in the safety

analysis, currently 0.65 $\Delta k/k$ and 1.0% $\Delta k/k$, respectively. Calculations are performed to verify that this condition is maintained over the entire range of allowable control rod insertion positions.

2.3.1.6 Shutdown Margin

The safety analysis assumes that the reactor can be taken subcritical with a shutdown margin of 1% $\Delta k/k$ at any time during the cycle even with the highest worth control rod stuck out of the core. The shutdown margin for the reload cycle is evaluated throughout the cycle depletion to verify this assumption. Limits on the amount of regulating control rods insertion are determined to ensure that the minimum shutdown margin requirement is maintained.

2.3.1.7 Moderator Temperature Coefficients and Deficits

The moderator temperature coefficients (MTC) for the bounding beginning-of-cycle (BOC) and end-of-cycle (EOC) state points shall be within the upper and lower limits, respectively, of the values assumed in the safety analyses. In Chapter 14 of the TMI-1 FSAR, the BOC MTC is used for reactor coolant heatup accidents while the EOC MTC is used for events resulting in reactor coolant temperature decreases. The BOC MTC is primarily related to the initial excess reactivity which is controlled by soluble boron shim; it is affected by the cycle length and burnable poison loading. The EOC MTC is related principally to the core average burnup which is controlled by feed batch size and cycle length.

The temperature deficit is defined as the cumulative reactivity addition over a temperature range. Due to the rapid cooldown of the reactor coolant during a steam line break event, the temperature deficit, instead of the moderator temperature coefficient is used. The SLB safety analysis sets the limit for the EOC temperature deficit as 0.6% $\Delta k/k$ when the moderator temperature changes from 532°F to 524°F.

2.3.2 Final Fuel Cycle Design Depletion

The final fuel cycle design is depleted to the end of the designed cycle length using SIMULATE-3 with control rods set at their nominal operating positions. The depletion is performed with burnup steps no greater than 50 EFPD apart. At selected cycle burnup points, exposure and power distribution information are written to restart files for later use in the reload analysis such as for the control rod scans and design power transients described in Section 4.1.

2.3.2.1 Power Distribution Analysis

The fuel assembly radial powers and fuel rod peak powers are maintained below levels which provide sufficient margin to design limits. This ensures that acceptable core operating limits will be obtained from maneuvering analysis.

2.3.2.2 Maximum Discharged Fuel Assembly/Rod Burnups

One of the design criteria is that the maximum fuel assembly and fuel rod burnups for each fuel type in the reload design do not exceed the respective fuel design burnup limits. This criteria is confirmed based on results obtained from the cycle depletion. Maximum fuel assembly burnups calculated by SIMULATE-3 at the end of the cycle design length are compared to the assembly burnup limits, which are primarily set by fuel assembly growth characteristics such as guide tube and fuel rod growth. The maximum fuel rod burnup for each assembly is obtained using the fuel rod power reconstruction capability of SIMULATE-3. The calculated fuel rod burnups are compared to the fuel rod burnup limits, which are primarily set by thermal-hydraulic constraints (i.e. fuel melt, LOCA initialization and internal rod pressure criteria) as well as mechanical constraints (i.e. clad creep collapse, clad strain and clad stress).

2.3.3 Control Rod Worths

The primary function of control rods is to provide adequate shutdown capability during normal and accident conditions. SIMULATE-3 is used to calculate the reactivity worth of various control rod configurations. Individual control rod group worths are calculated for comparison to measured control rod worths obtained during startup physics testing which is discussed in Section 2.4. Maximum stuck rod, dropped rod, ejected rod and total rod worths are calculated to verify that assumptions used in the safety analysis bound the reload cycle. Total control rod worths and maximum stuck rod worths are also utilized to

verify that adequate shutdown margin is available throughout the reload cycle.

2.3.3.1 Control Rod Grouping

The TMI-1 reactor contains 61 full length control rods which are divided into seven (7) groups. Groups 1 through 4 contain the safety rods which are fully withdrawn during normal operation and are dropped into the core on a scram signal. Groups 5, 6 and 7 contain the regulating rods which are used to control plant power during normal operations. The core locations of the control rod drive mechanisms are fixed but the rods assigned to a particular group may be cycle dependent. The control rod group assignments are determined by nuclear calculations which evaluate the effects that a particular group has on power distributions, control rod group worths and ejected rod worths.

In addition to the full length control rods, there are also 8 partial length axial power shaping rod assemblies (APSRAs) which are designated as Group 8. The APSRAs are not required for reactivity control but are used to control the core axial power shape during plant transients. The poison material used in APSRAs is inconel which has significantly lower reactivity worth than the Ag-In-Cd material used in the full length control rods. APSRAs are typically positioned near the core midplane for the entire cycle to minimize their effect on core offset and to allow operators to correct offset in either direction as quickly as possible.

A typical control rod group map is shown in Figure 2.1.

2.3.3.2 Individual Group and Total Control Rod Worths

Calculation of the individual group worths used for comparison to startup physics testing measurements are performed using SIMULATE-3 in quarter core geometry with HZP, no xenon and peak samarium conditions. Individual group worths are calculated beginning with all control rods withdrawn followed by the sequential insertion of Group 7, Group 6 and Group 5.

Total control rod worths are also calculated in quarter core geometry at both HFP and HZP. The difference in core reactivity from the all rods out (ARO) condition to the all rods in (ARI) condition is the total control rod worth. The total control rod worth must not be greater than the value assumed in the safety analysis.

Integrated control rod worth curves are calculated for the regulating rods (Groups 5, 6 and 7) with Groups 1 through 4 fully withdrawn. These curves are generated using SIMULATE-3 in quarter core geometry at HFP and HZP. The integrated control rod worths range from a control rod index of 0% withdrawn (WD) (i.e. Groups 5 to 7 fully inserted) to a control rod index of 300% WD (i.e. Groups 5 to 7 fully withdrawn). The plant control rod drive hardware is set up to maintain a 25% overlap between regulating control rod groups as one group approaches the full withdrawn position and withdrawal of the succeeding group begins. This overlap results in a fairly linear integrated control rod worth curve and is modelled as such using SIMULATE-3. The integrated control rod worth curves are used in the shutdown margin and ejected control rod worth analyses to determine control

rod index limits. The effect of xenon and samarium on the integrated control rod worth shapes are accounted for in the analysis. A typical integrated control rod worth curve is shown in Figure 2.2.

2.3.3.3 Ejected Control Rod Worth

The maximum allowable ejected control rod worth is limited to prevent the stored energy of the fuel from exceeding the values used in the safety analysis for the control rod ejection accident. The ejected control rod worth limits assumed in the safety analysis are 0.65% $\Delta k/k$ at HFP and 1.0% $\Delta k/k$ at HZP (Reference 1). The ejected control rod worth limit is assumed to be a linear function of power between HZP and HFP. The ejected control rod worth limits are adjusted by 15% which bounds the control rod worth reliability factor determined in Reference 5 which accounts for calculation uncertainty.

Ejected control rod worth calculations are performed using a SIMULATE-3 full core geometry model. Fission products and thermal-hydraulic conditions are not allowed to change during the control rod ejection due to the extremely short time frame the event spans. Because of Technical Specifications and operation limits, the following control rod positions are modeled:

- At HFP, only Group 7 is inserted since, based on ECCS (i.e., LOCA) limits, HFP operation with a control rod index below 200% WD is not permitted.

- Below HFP, only control rod Groups 5, 6 and 7 are inserted since Groups 1 through 4 must be withdrawn prior to approach to criticality as required by Technical Specifications.

- At HZP, control rod Groups 1 through 7 are fully inserted.

APSRAs are positioned at the core midplane for all ejected control rod worth calculations. The ejected control rod worths are also calculated for different initial xenon conditions and the most limiting results are chosen for the ejected control rod worth.

2.3.3.4 Dropped Control Rod Worth

The maximum dropped control rod worth in the core is limited to ensure that the change in reactivity and power peaking during a potential dropped control rod accident do not exceed the values assumed in the safety analysis. Dropped control rod worths are calculated using a SIMULATE-3 full core model with thermal-hydraulic feedback for this relatively slow event. Reactor power is maintained at HFP, even after the control rod is dropped. Cases are run with HFP equilibrium xenon and a HFP boron concentration with all rods out and APSRAs positioned at the core midplane. All control rods in a quarter of the core are evaluated to determine the maximum dropped control rod worth. To account for calculational uncertainty, the calculated results are adjusted by 15% which bounds the control rod worth reliability factor determined in Reference 5 at BOC, MOC and

EOC. The adjusted values are compared with the dropped control rod worth safety analysis limit shown in Table 6.1.

2.3.4 Shutdown Margin

The Technical Specifications require that the reactor must be capable of shutting down with a minimum of 1% $\Delta k/k$ shutdown margin with the highest worth control rod stuck out of the core. This minimum shutdown margin is also an assumption used in the safety analysis. The shutdown margin for the final fuel cycle design must be demonstrated to meet this requirement throughout the cycle length. The shutdown margin for the reload cycle is conservatively calculated at various state points throughout the cycle as the difference between the minimum available control rod worth and the maximum required reactivity worth.

The available control rod worth is the control rod worth of all the control rods with the strongest control rod stuck out. Both the control rod worth and the strongest stuck control rod worth are determined using SIMULATE-3. Since the Group 7 control rods are partially inserted during the cycle, there is a reduction of control rod worth due to depletion of these control rods. After corrected for the depletion effect, the calculated control rod worth is again reduced by the control rod worth reliability factor determined in Reference 5 to account for the calculation uncertainty. This is considered as the minimum available control rod worth.

The required reactivity worth is the sum of: (1) the reactivity inserted when shutting down

from HFP to HZP, (2) the reactivity increase due to xenon redistribution, and (3) the reactivity corresponding to the maximum allowable inserted control rod worth. The first two components can be determined using SIMULATE-3 and the last component is available from the control rod worth curve discussed in Section 2.3.3.2.

2.3.5 Reactivity Coefficients and Deficits

The reactivity coefficient defines the reactivity change for a small perturbation in a core parameter while all other parameters are held constant. The core parameters of interest are the moderator temperature, fuel temperature, power level and soluble boron. These coefficients are generally input to safety analysis and are used to model reactor responses during accidents and transients. On the other hand, the reactivity deficits usually apply to large reactivity changes, such as going from HFP to HZP.

In the safety analysis, bounding values of the reactivity coefficients are used. The reactivity coefficients calculated for each reload must be bounded by these values in order for the safety analysis results to remain valid. This is further discussed in Section 6 of this report.

2.3.5.1 Doppler Coefficient

The Doppler coefficient, η_D, is the change in core reactivity ($\Delta\rho$) per degree change in the average fuel temperature (ΔT_{fuel}). As the fuel temperature increases, the resonance absorption cross sections of Uranium-238 and Plutonium-240

broaden, thus increasing neutron absorption and producing a negative reactivity effect.

The HFP and HZP Doppler coefficients are determined at BOC and EOC using SIMULATE-3. The reactivity of the reference condition is first determined. While all other conditions remain the same, the reference condition is perturbed by a uniform change in the fuel temperature. The reactivity of the perturbed case is then determined and the Doppler coefficient is calculated as

$$\eta_D = \frac{\Delta\rho}{\Delta T_{fuel}} \qquad \text{(Eq. 2.1)}$$

2.3.5.2 Moderator Temperature Coefficient

The moderator temperature coefficient, η_M, is the change in reactivity per degree change in the average moderator temperature (ΔT_{mod}). For a given increase in moderator temperature, the moderator density decreases, thus decreasing the moderation and producing a negative reactivity effect. Concurrently, the soluble boron concentration decreases, and there is a positive reactivity effect from the decrease in neutron absorption. The positive effect from the decreased absorption can become greater than the negative effect from the decreased moderation at high boron concentrations. Therefore, the moderator temperature coefficient can be positive due to a relatively large presence of soluble boron in the moderator. The Technical Specification specifies that the moderator temperature coefficient

has to be negative when the core power is greater than 95% rated power.

The HZP and HFP moderator temperature coefficients are calculated at BOC and EOC using SIMULATE-3. The reactivity of the reference condition is first determined. While all other conditions remain the same, the reference condition is perturbed by a change in the moderator inlet temperature. The reactivity of the perturbed case is then determined and the moderator coefficient is calculated as

$$\eta_M = \frac{\Delta \rho}{\Delta T_{mod}}$$

(Eq. 2.2)

2.3.5.3 Total Temperature Coefficient

The total temperature coefficient, η_T, is the change in reactivity ($\Delta\rho_T$) associated with an equal change in the fuel and moderator temperature divided by the change in the average moderator temperature. This is equal to the sum of the Doppler coefficient and the moderator temperature coefficient.

The HZP and HFP total temperature coefficients are calculated at BOC and EOC using SIMULATE-3. The reactivity of the reference condition is first determined. While all other conditions remain the same, the reference condition is perturbed by a uniform change in both fuel temperature and moderator inlet temperature. The reactivity of the perturbed case is then determined and the total temperature

coefficient is calculated as

$$\eta_T = \frac{\Delta \rho_T}{\Delta T_{mod}} = \eta_D + \eta_M \qquad \text{(Eq. 2.3)}$$

At hot zero power, the total temperature coefficient is also the isothermal temperature coefficient since the fuel temperature and the moderator temperature are the same. During the startup physics test, the HZP isothermal temperature coefficient is measured and compared with the calculated HZP isothermal coefficient at BOC.

2.3.5.4 Power Coefficient and Power Deficit

The power coefficient, η_P, is the change of reactivity per a percent change in reactor power level. For power greater than 15%, TMI is operated with a constant average moderator temperature and the reactivity change is due to the change in fuel temperature as the power level changes.

The power coefficient is determined using SIMULATE-3 by first calculating the reactivity of the reference condition. The power level is then disturbed by a small change in core power. The reactivity of the perturbed state is determined using the reference case power distribution. The power coefficient is calculated as

$$\eta_p = \frac{\Delta \rho}{\Delta Power} \qquad \text{(Eq. 2.4)}$$

The power deficit is the total reactivity change associated with a large change in power level such as from HFP to HZP. The power deficit is also calculated using SIMULATE-3 to determine the reactivity difference between the reference and perturbed cases. Both moderator and the fuel temperatures are to be changed based on the perturbed power. The power deficit is used in the shutdown margin calculation.

2.3.6 Boron Related Parameters

Critical boron concentrations at BOC and EOC for HFP and HZP are calculated using SIMULATE-3 for the all rods out condition. A boron letdown curve using a nominal control rod index is also determined. The boron coefficient, η_B, is the change of reactivity per ppm change in the boron concentration. The value of this parameter depends on the core conditions; such as, boron concentration, moderator temperature, the presence of control rods and the number of burnable poison rods.

The boron coefficient is calculated using SIMULATE-3 at various plant conditions. The reactivity of the reference condition is first determined. While all other conditions remain the same, the reference condition is perturbed by a uniform change in the soluble boron concentration. The reactivity of the perturbed case is then determined and the boron

coefficient is calculated as

$$\eta_B = \frac{\Delta\rho}{\Delta PPM} \qquad \text{(Eq. 25)}$$

Also, to ensure the required shutdown margin be met using boron injection, the minimum BWST boron concentration is calculated based on the refueling boron concentration, assuming the two highest worth control rods are stuck out and a full dilution of the boron makeup tank occurs.

2.3.7 Xenon Worth

The difference in reactivity between a no xenon and an equilibrium xenon case is the xenon worth. This is calculated at various times in the cycle using SIMULATE-3. The reactivity of the equilibrium xenon condition is determined at HFP with a nominal control rod position. The xenon concentration is then set to zero and the reactivity of the perturbed case is determined.

2.3.8 Kinetics Parameters

Kinetics parameters determine the dynamic response of the core. Neutronic excursions in calculations of the accidents in the safety analysis are directly related to the change in reactivity and the inverse of the delayed neutron fraction. The rate of change in power from a given reactivity insertion can be calculated by solving the kinetics equations if the

six group effective delayed neutron fraction, the six group precursor decay constants and the prompt neutron lifetime are known. For a given core condition, SIMULATE-3 collapses the 3-dimensional physics data to obtain the above core averaged kinetics parameters using adjoint flux weighting. There is no bounding value established in the safety analysis for the kinetics parameters. Since these parameters are entered into the reactimeter during startup testing, good agreements between the measurements and calculations indirectly verify the accuracy of the kinetics parameters.

2.4 Startup Tests

The purpose of the design analyses of the reload cycle is to ensure that the reference safety analyses remain applicable. This would normally be the case using the established methods and acceptance criteria if there are no design changes or changes in the manufacturing specifications. Since the calculation uncertainties applied are derived statistically, the startup test program confirms the applicability of the methodology and the safety analysis results.

Prior to plant startup of the new cycle, the HZP kinetics parameters, critical boron concentration, isothermal and moderator temperature coefficients, boron coefficient, control rod group worths and HFP fuel assembly power distribution, moderator and power coefficients are calculated using SIMULATE-3. Acceptance criterion for each parameter is established based on the calculation bias and uncertainty determined in Reference 5.

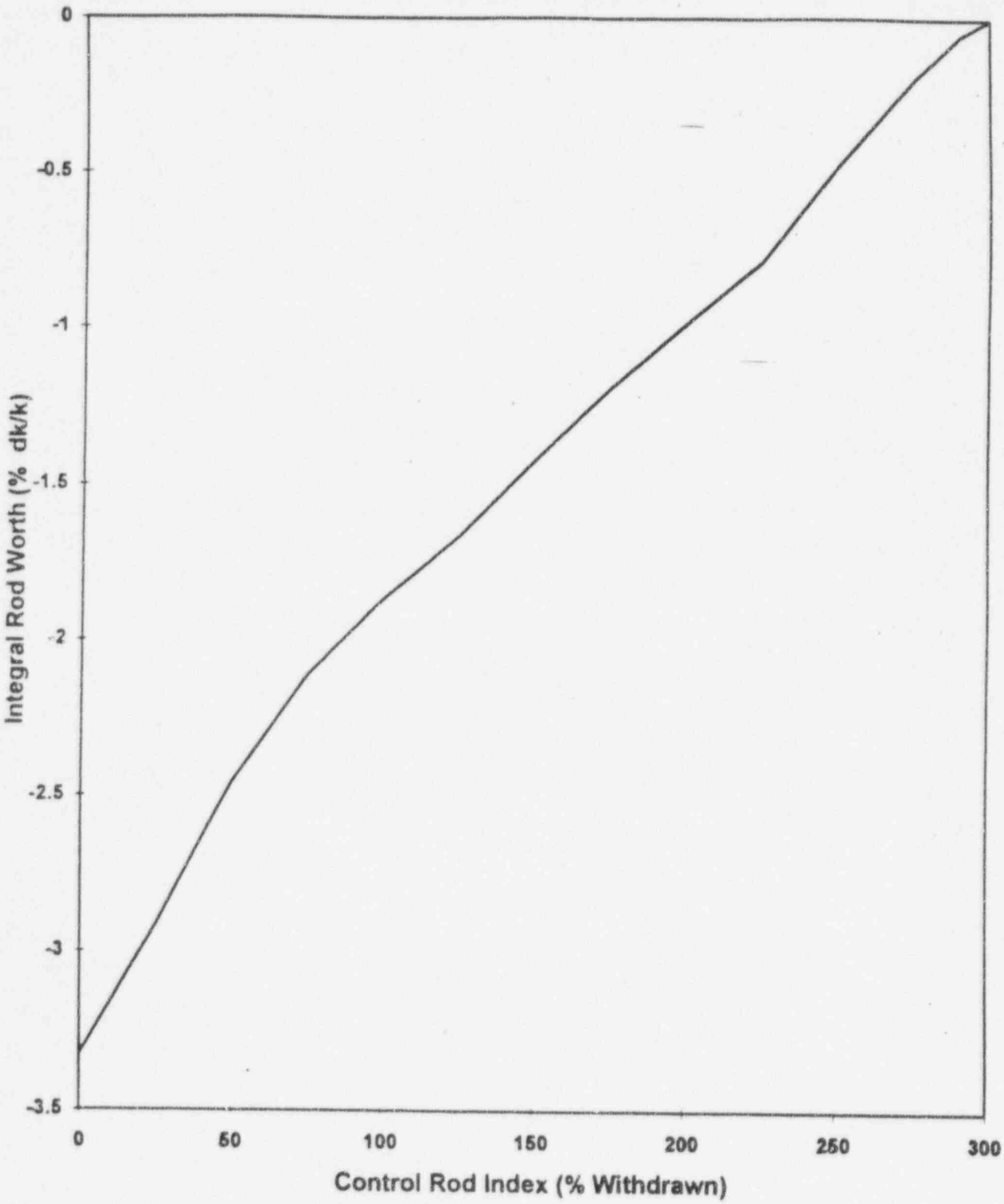
During startup, the calculated kinetics parameters are entered into the reactimeter. When the reactor reaches HZP, the critical boron concentration, the temperature coefficients, control rod

group worths and boron worth are measured and compared with the corresponding predicted values. Each test must show that the comparison of the calculation and measurement meets the appropriate acceptance criteria before operation may continue. As power operation commences, additional tests of the fuel assembly relative power distribution and critical boron must continue to show that comparisons of the predictions and measurements meet the appropriate acceptance criteria.

Figure 2.1
Typical Control Rod Group Configuration

					1		6		1					
				3		5		5		3				
			7		8		7		8		7			
		3		5		4		4		5		3		
	1		8		6		2		6		8		1	
		5		4		2		2		4		5		
	6		7		2		7		2		7		6	
		5		4		2		2		4		5		
	1		8		6		2		6		8		1	
		3		5		4		4		5		3		
			7		8		7		8		7			
				3		5		5		3				
					1		6		1					

Figure 2.2
Typical Integral Rod Worth Curve



3.0 CORE THERMAL-HYDRAULIC DESIGN

This section discusses the thermal-hydraulic (T-H) analysis methodology for the TMI-1 reload design. Specific items described in this section are the design criterion, determination of design T-H parameters, generic DNBR analysis, methodology for the pressure-temperature limit, flux/flow protection limits, and maximum allowable peaking limits. Analyses are performed by using the VIPRE-01 1/8th-core hot-channel model which is described in detail in Reference 7.

3.1 Design Criterion and Design DNB Limit

The heat transfer regime where small bubbles are forming on the fuel clad surface is defined as the nucleate boiling. At this heat transfer regime, or at the nucleate boiling condition, the temperature difference between clad and reactor coolant is small due to efficient heat transfer. As fuel rod power is increased, the bubble generation increases to a point where the bubbles form an insulating blanket over the heating surface, causing a large increase in clad temperature. This point is called either the critical heat flux (CHF), or departure from nucleate boiling (DNB). The DNB ratio (DNBR) is the ratio of this CHF at a given point on a fuel rod to the actual heat flux at the same location.

The design criterion for thermal-hydraulic (T-H) performance is that there shall be at least a 95% probability at a 95% confidence level that the hot fuel rod does not experience a departure from nucleate boiling (DNB) during normal operation and anticipated transients. An alternate statement of this 95/95 protection criterion is that 95% of fuel rods at the DNB condition will not experience DNB with 95% confidence. The design DNB limit is defined as the DNBR value for which there is

a 95% probability at the 95% confidence level that DNB will not occur.

The design DNB limit for the BWFC Mark-B fuel design with the zircaloy spacer grids is 1.18 (Reference 7). The CHF data base analysis is performed by using VIPRE-01 computer model, the BWC CHF correlation and CHF test data (References 7, 14 and 15). DNBR values greater than the 1.18 design DNB 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.

3.2 Reference Design Thermal-Hydraulic Parameters

The fresh fuel and corresponding core loading pattern of the reload cycle, including possible plant modifications, could impact the core thermal-hydraulic (T-H) condition. Therefore, it is necessary to show the design DNBR limit of 1.18 is met for each reload design. The DNBR analyses, both steady-state and transient, are based on the conservative T-H design parameters: coolant flow, core bypass flow, power level, core inlet temperature, system pressure, and radial and axial power distributions. The design T-H parameters are listed in Table 3.1. Described below are the design conditions of thermal-hydraulic parameters.

3.2.1 Core Inlet Conditions (Core Flow and Inlet Temperature)

The original TMI-1 design reactor coolant system (RCS) flow rate is 88,000 gpm per pump. Flow measurements during the plant operation indicate that the actual RCS flow rate averages approximately 110% of the design flow. Therefore, 106.5% of the original design

value has been conservatively chosen for the core T-H analysis for each reload design.

The flow available for heat transfer is called reactor core coolant flow and is equal to total RCS flow less the maximum bypass flow, which is defined as the flow that does not contact the active heat transfer surface area. The design (or maximum) bypass flow typically ranges from 7% to 10% depending upon reload cycle design, which is described in Section 3.2.2.

There are four TMI-1 constant volumetric flow reactor coolant pumps; thus the RCS mass flow rate (in lbm/hr) is a function of RCS cold leg temperature or core inlet temperature (T_{in}). Furthermore, the Integrated Control System (ICS) maintains a constant average temperature (defined as $T_{av} = (T_{out} + T_{in})/2$) at 579°F over the power range of 15-to-100 percent, which requires that the core inlet temperature decrease with increasing core power. As the cold leg temperature decreases the core inlet mass flow rate increases caused by the increase in the coolant density. These two factors, i.e., constant volumetric flow and the role of the ICS, are taken into account in determining the core coolant mass flow rate and core inlet temperature. A temperature instrument uncertainty of ± 2°F is also applied in determining both core flow and inlet temperature.

The TMI-1 reactor vessel internals are designed to provide a relatively uniform inlet flow distribution to the core. However, the fuel assemblies in the interior region of the core have a slightly higher flow than the flow in the peripheral region (Reference 4). Thermal-hydraulic analysis using the VIPRE-01 code assumes a uniform flow distribution core-wide with the hot assembly inlet flow reduced to 95% for interior locations and 87% when the

hot assembly is in a peripheral location (Reference 4). The inlet flow distribution factors for different pump conditions are given in Table 3.2.

3.2.2 Core Bypass Flow

As mentioned earlier the core bypass flow reduces the core flow available for heat transfer. Therefore, as bypass flow increases, DNBR results decrease (i.e., become worse). The main paths of the bypass flow are: (1) core barrel annulus, (2) interfaces separating the core inlet and outlet nozzle, (3) core shroud gap, and (4) control component and instrument guide tubes. Items (1) and (2) above are fixed bypass components by the NSSS design; however, Items (3) and (4) are fuel assembly design and reload core dependent.

The bypass flow through the core shroud gap in Item (3) is the flow between the outer edges of the peripheral assemblies and core baffle wall and is fuel design dependent. Item (4) above is the bypass flow through the unplugged control component and incore instrument guide tubes of fuel assemblies having neither a control rod assembly (CRA) nor burnable poison rod assembly (BPRA). The number of BPRAs varies from one fuel cycle to another depending upon nuclear design requirements during the reload design process. This means the number of unplugged guide tubes varies from cycle to cycle and is, therefore, reload-design dependent.

In general, reference design DNBR analyses are performed with an assumption of a conservatively high bypass flow that will bound actual reload cycle values.

3.2.3 System Pressure

The nominal TMI-1 system pressure is 2200 psia. The VIPRE-01 input for the DNBR analyses is 2135 psia incorporating a 65 psi uncertainty.

3.2.4 Power Distribution Factors

The VIPRE-01 DNBR analyses are performed with the design radial peaking factor (or maximum radial-local peaking factor), $F_{\Delta H}^{N}$, of 1.714 at the hot channel. The value of the design radial peaking factor, $F_{\Delta H}^{N}$, increases with decreasing power level and is given by:

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The generic fuel rod power distribution is conservatively modeled for the hot assembly in the VIPRE-01 model with a relatively flat local peaking gradient to minimize beneficial energy mixing effects. The design axial power shape is a 1.65 symmetric cosine shape with tails as shown in Figure 3.1.

Both design values of the fuel rod and axial power distributions are given in Table 4.3 and Figure 4.2 of Reference 7.

3.2.5 Engineering Hot Channel Factors

Engineering hot channel factors (Reference 4) consist of the local heat flux factor, F_q^*, average fuel rod power factor (also called enthalpy rise factor), F_q, spacer grid hot channel factor, F_{sp}, and flow area reduction factor, F_A. These factors are used in the VIPRE-01 DNBR analysis to account for the effects of manufacturing variations.

The local heat flux factor of 1.014 is used to account for the manufacturing tolerances in pellet density, pellet cross-sectional area, weight per unit length, local enrichment, and local outer clad diameter. The enthalpy rise factor of 1.0132 (Reference 16) accounts for variations in average fuel rod power caused by differences in the absolute number of grams of U-235 per fuel rod. The spacer grid factor of 1.007 is used to account for the flux depression due to the spacer grids. The flow area reduction factor is 0.98 for the hot fuel rod unit cells, and is 0.97 for the instrument guide tube and corner cells. This factor accounts for the effects of variations in fuel rod pitch and diameter.

3.2.6 Fuel Rod Bowing and Densification Effects

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in Section 3.2.5 and, therefore, is not applied.

The main concern of the fuel densification effect is a local power spike due to the possible fuel stack height shrinkage and the effects of inter-pellet gaps caused by the shrinkage.

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3.3 Reference Design DNBR Analysis

The reference design DNBR is the hot channel minimum DNBR based on the 112% overpower analysis incorporating other limiting T-H design conditions of coolant flow, pressure, core inlet temperature, and power distribution (see Table 3.1 for design T-H conditions and Figure 3.1 for design axial power shape). This analysis based on design T-H conditions not only demonstrates that the design overpower condition satisfies the T-H criterion of a 1.18 design DNBR limit but also defines the reference points for other T-H analyses. These analyses include the determination of pressure-temperature (P-T) core protection limits and maximum allowable peaking (MAP) limits. These are described in detail in the following sections.

3.4 Determination of Pressure-Temperature Core Protection Limits

The curves given in Figures 3.2 and 3.3 show Core Protection Safety Limits and Core Protection Safety Bases. These curves place limits on the observable (and controllable) RCS parameters such as the core outlet temperature and pressure for DNBR protection. In other words, the curves represent a series of values of RCS temperature and pressure where the T-H conditions result in the design DNBR limit of 1.18.

The Safety Limits in Figure 3.2 are determined as the most conservative (or restrictive) values from the Safety Bases in Figure 3.3. Operation above and to the left of the Safety Limits in Figure 3.2 at any power level up to the design overpower (112% of rated power) will ensure DNBR values greater than the design limit of 1.18 and, consequently, will keep the RCS in the nucleate boiling regime.

The overall analysis procedure for the determination of the Core Protection Safety Limit curve is:

The method on how to determine the reactor trip setpoints based on the above safety limit is described in Section 5.2.1.

3.5 Transient Core Thermal-Hydraulic Analysis

During a loss of one or more reactor coolant pumps, the core is prevented from violating the 1.18 design DNBR limit through a reactor trip initiated by either a flux/flow trip or power/pump monitor trip, which are discussed in Sections 3.5.1 and 5.4.1. The VIPRE-01 transient DNBR analysis is performed by using either a transient model or a quasi-steady state model, depending on which produces more conservative DNBR results. The reload safety evaluation determines whether any changes introduced by the reload core affect the validity of the existing (or reference) transient analyses.

The key parameters reviewed to assess the need for reanalysis of the DNB-limited transients are

given in Table 3.5. Parameters whose minimum values cause a limiting T-H condition are the RCS flow rate and pressure. Parameters whose maximum values lead to a limiting condition are the core power, RCS inlet temperature, bypass flow, peaking factors, and engineering hot channel factors.

3.5.1 Flux/Flow Protective Limits

As mentioned in the previous section, the DNB protection for loss-of-coolant flow (LOCF) transients is provided by either a power/imbalance/flow trip (also referred to as the protection system maximum allowable setpoints) or a power/pump status trip depending upon the transient initiating conditions. In general, LOCF transients are the locked rotor accident and RCS pump coastdowns (single or multiple pump trips).

The power/pump status monitor produces a reactor trip if (Reference 19):

1. Two RCS pumps are lost in one coolant loop (i.e., 2/2-to-2/0, 2/1-to-2/0, 2/1-to-0/0, 1/1-to-0/0 and 2/2-to-0/0 pump coastdowns), or

2. Power level is higher than or equal to 55% of rated power when one pump is operating in each coolant loop (i.e., 2/2-to-1/1 and 2/1-to-1/1 pump coastdowns with power level no lower than 55%).

The power/pump status trip therefore provides protection against multiple pump coastdowns and complete loss of coolant flow.

The flux/flow limit is the flow-dependent portion of the power/imbalance/flow limits. The flux/flow protective limit is established for DNB protection when a flow reduction occurs due to single or multiple pump trips and also provides a high flux trip for partial pump operation (i.e., 3-pump or 2-pump). An example of the overpower protection event during the partial pump condition would be a control rod withdrawal accident.

When the RCS flow is reduced due to a pump coastdown (single or multiple) and the flux-to-flow ratio reaches the flux/flow trip setpoint (current value = 1.08), the reactor will trip. This means that when a multiple pump coastdown occurs, the flux/flow trip and pump status trip compete with each other and the reactor will be tripped by the pump status monitor. This is because the pump status monitor detects the trip condition faster than the flux/flow monitor by immediately recognizing the pump status in each RCS loop, instead of tracing the flux/flow ratio as a function of time.

Therefore, the design basis event (DBE) or limiting flow coastdown of the flux/flow setpoint is a single pump coastdown, i.e., 4-to-3 pump trip. This transient bounds the loss of one pump from a 3-pump operating condition, (i.e., 3-to-2 pump coastdown).

As shown in Table 3.5, the VIPRE-01 DNB analysis for the flux/flow event (Reference 7) assumes initial power of 108% (= 102% + 6% of neutron measurement error) with 1.714 reference design radial-local power peaking factor and an axial power shape of 1.65 cosine with tails.

The neutron measurement error breakdown follows with each component discussed below:

Heat Balance Error	=	2 %FP
Steady State Flux Calibration Error	=	2 %FP
Transient Induced Error	=	2 %FP
Total Neutron Measurement Error	=	6 %FP

The heat balance error is required by Regulatory Guide 1.49 (Reference 20), which states that the safety analysis should be performed at a power level 2 %FP greater than rated power to account for uncertainties in the determination of power level through the heat balance calculation. The steady state flux calibration error accounts for the neutron flux

The VIPRE-01 flux/flow transient DNBR analysis is performed based on the limiting T-H conditions in Table 3.5 and a conservative flow coastdown curve (Figure 3.4) for a 4-to-3 pump trip. As shown in Figure 3.4, the flux/flow limit is determined as a ratio of reactor power to the RCS flow at which the design DNBR occurs.

3.6 Maximum Allowable Peaking (MAP) Limits

As described in previous sections, the DNBR analyses for the P-T core protection limits, overpower limits, and flux/flow protective limits, including all FSAR events, utilize a design radial-local power peaking factor of 1.714 and a design axial power shape of 1.65 cosine with tails. The design axial power shape is center-peaked and produces a zero axial power imbalance and offset, which are defined as:

$$\theta_{IMB} = \theta_{OFF} * P \qquad \textbf{(Eq. 3.3.a)}$$

$$\theta_{off} = (P_T - P_B)/(P_T + P_B) \qquad \textbf{(Eq. 3.3.b)}$$

where:

θ_{IMB} = axial power imbalance

θ_{off} = axial core power offset

P = fraction of rated power

P_T = power in the top half of the core

P_B = power in the bottom half of the core

In actual power operation, the power imbalance can be either positive or negative. For example, the axial power shape could become bottom-peaked (negative imbalance) with a certain degree of control rod insertion, or a top-peaked shape (positive imbalance) when a xenon oscillation occurs during a reactor power load swing. There is a need to establish power/imbalance limits for both the RPS trip setpoints and LCO alarm limits when the axial power shape is non-zero imbalance. The Maximum Allowable Peaking (MAP) limits provide information to determine these

power/imbalance limits.

MAP limits are a series of curves, typically plotted as maximum allowable total peak versus the axial location of the peak as shown in Figure 3.5, with the axial peaking factor as the variable parameter. The family of curves is a locus of points for which the minimum DNBR is equal to an analysis target value. The MAP limits are compared with calculated power distribution data for the determination of trip setpoints and alarm limits. MAP limits ensure the design radial-local power peaking factor of 1.714 with the center-peaked design axial power shape used for the core protection limits and transients in the FSAR bounds the various non-center peaked axial power shapes/peaking factors.

There are two kinds of MAP limits depending upon where they are used:

1. RPS MAP limits are used for the determination of RPS power/imbalance trip setpoints

2. LCO MAP limits are used for the determination of power/imbalance alarm limits for normal operating control.

These MAP limits are discussed in the following sections.

3.6.1 Determination of RPS MAP Limits

The RPS MAP limits define the limiting total peaking factors for various axial power shapes/peaking factor, which result in the same minimum DNBR value as that at the worst state points of the core P-T protection limits; that is, the low-pressure and high temperature

points in Figure 3.6.

3.6.2 Determination of LCO MAP Limits

The LCO MAP limits are developed to ensure that the DNBR performance of the core with various axial power shapes is bounded by the design power shape for the loss of coolant flow (LOCF). LCO MAP limits are used to determine the control rod insertion limits and axial power imbalance limits for normal operating control. Analysis procedures are identical to those for the RPS MAP limits described in the previous section except:

3.6.3 Verification of MAP Margins

In order to determine either trip setpoints or alarm limits, the power peaking factors actually calculated in the maneuvering analysis (Section 4) are compared with MAP limits in the following manner:

$$\Delta(MAP) = (MAP_{IPX} - PF_{IPX}) / MAP_{IPX} \qquad \textbf{(Eq. 3.6)}$$

where:

$\Delta(MAP)$ = MAP margin

MAP_{IPX} = MAP limit (LCO MAP or RPS MAP) for given power imbalance (I), axial peak (P), and location of axial peak (X)

PF_{IPX} = Calculated peaking factor (uncertainty adjusted) for given power imbalance (I), axial peak (P), and location of axial peak (X).

As previously described, the MAP limits are derived by the use of a series of smooth, mathematically-derived axial power shapes (Section 3.6.1); therefore, they are considered to be approximations subject to verification. Verification analysis is performed by

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Table 3.1
Design Thermal-Hydraulic Conditions

KEY PARAMETERS:	VALUE
Design Overpower(% of 2568 MWth)	112
Average Linear Heat Rate (kw/ft)	5.76
Active fuel Length (Inches)	140.6
Design Radial-Local Peaking Factor	1.714
Axial Power Shape	1.65 Cosine (With Tails)
Engineering Hot Channel Factors:	
Enthalpy Rise Factor (F_q)	1.0132
Local Heat Flux ($F_{q''}$)	1.014
Flow Area Reduction Factor	0.97
RCS Pressure (psia)	2135
Design Flow per RC Pump (gpm)	88000
RCS Flow (% Design)	106.5
Bypass Flow (%)	Varies from 7% to 10%
Effective Flow Area for Heat Transfer (ft^2)	49.65
CHF Correlation	BWC
Design DNBR Limit	1.18

Table 3.2
Core Inlet Flow Factors

Table 3.3
Typical Limiting Thermal-Hydraulic Conditions for P-T Analysis

PARAMETER DESCRIPTION	4-Pump OPERATION	3-Pump OPERATION	2-Pump OPERATION
Design Overpower (% of 2568 MWth)	112	(1)	(1)
Design Radial-Local Peaking Factor	1.714	1.714	1.714
Axial Power Shape	1.65 Cosine (With Tails)	1.65 Cosine (With Tails)	1.65 Cosine (With Tails)
ENGINEERING HOT CHANNEL FACTOR:			
Enthalpy Rise Factor (F_q)	1.0132	1.0132	1.0132
Local Heat Flux ($F_{q'}$)	1.014	1.014	1.014
Flow Area Reduction Factor	0.97	0.97	0.97
RCS Pressure (psia)	2135	2135	2135
RCS Flow (% Design)	106.5	[ ]	[ ]
Bypass Flow (%)	Maximum	Maximum	Maximum
RCS Inlet Temperature (°F)	Nominal + [ ]	Nominal + [ ]	Nominal + [ ]

NOTES:

(1) = See Equation 3.2 in Section 3.4 for the determination of partial pump design overpower.

Table 3.4
Typical Values of Total Flux Measurement Errors

for Flux/Flow Instrument String

[ ]

Table 3.5
Initial Conditions for Transient DNBR Analysis

PARAMETER DESCRIPTION	VALUE
Initial Reactor Power (% of 2568 MWth)	102[1]
Design Radial-Local Peaking Factor	1.714
Axial Power shape	1.65 Cosine (With Tails)
Engineering Hot Channel Factors:	
Enthalpy Rise Factor (F_q)	1.0132
Local Heat Flux ($F_{q''}$)	1.014
Flow Area Reduction	0.97
RCS Pressure (psia)	Nominal - 65
RCS Flow (% Design)	106.5
Bypass Flow (%)	Maximum
RCS Inlet Temperature (°F)	Nominal + [ ]

NOTES:
(1) = Initial power for flux/flow event is 108% (see Section 3.5.1).

Table 3.6
Limiting Thermal-Hydraulic Conditions for MAP Analysis

PARAMETER DESCRIPTION	RPS MAP	LCO MAP
Target DNBR Limit State Point[1]	[ ]	[ ]
Axial Power Shape	Varied	Varied
Maximum Axial Power Peaking	Varied	Varied
Reactor Power (% of 2568 MWth)	112	108
Hot Assembly Power Peaking	Varied	Varied
Design Radial-Local Peaking Factor	1.714	1.714
RCS Pressure (psia)	Nominal - 65	Nominal - 65
RCS Flow (% Design)	106.5	[ ]
Bypass Flow (%)	Maximum	Maximum
RCS Inlet Temperature (°F)	Nominal + [ ]	Nominal + [ ]
Engineering Hot Channel Factors:		
Enthalpy Rise Factor (F_q)	1.0132	1.0132
Local Heat Flux ($F_{q''}$)	1.014	1.014
Flow Area Reduction Factor	0.97	0.97

NOTES:

(1) = See Section 3.6.1 and 3.6.2.

Figure 3.1 Reference Design Axial Power Shape
(1.65 Cosine with Tails)

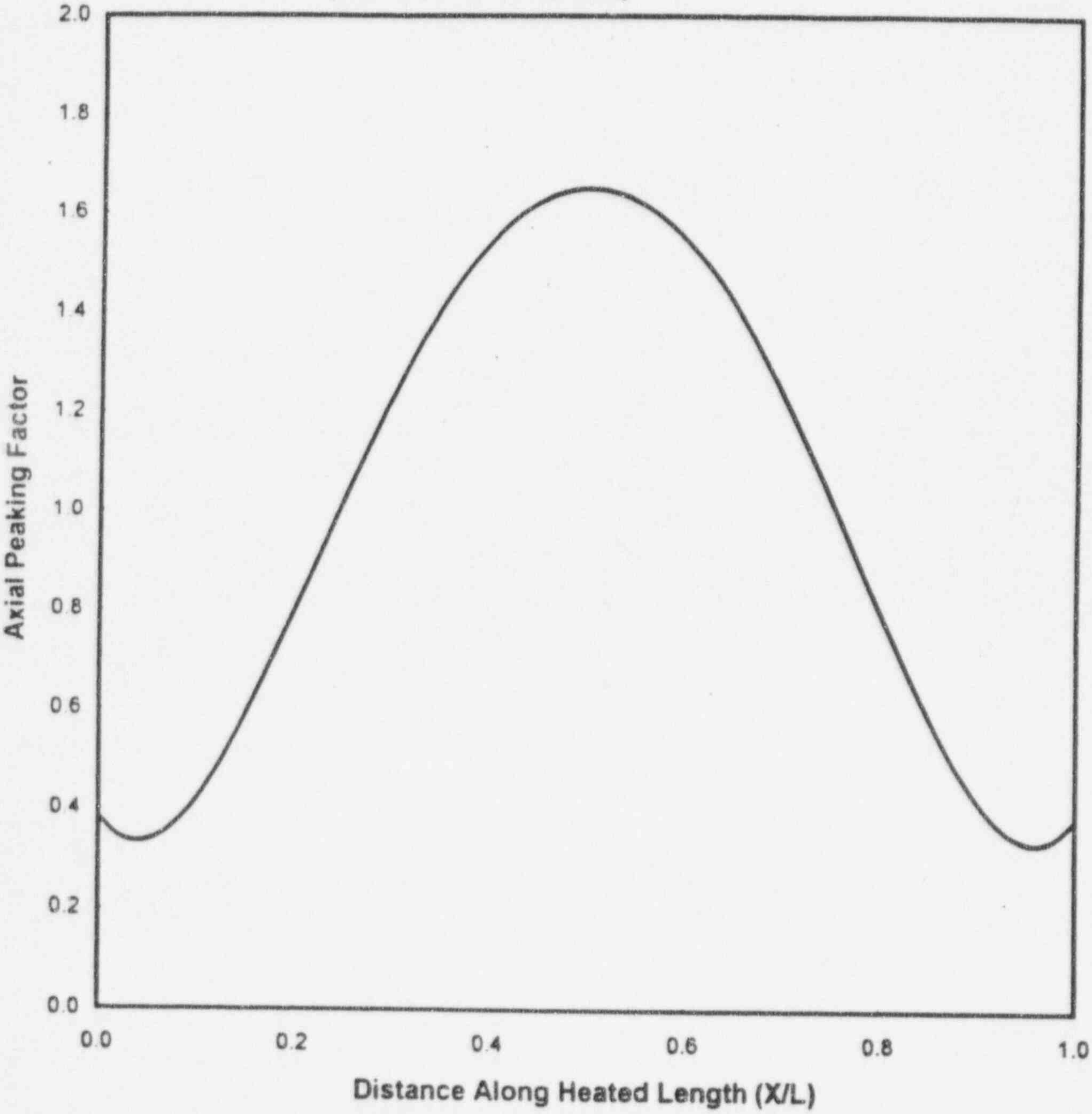


Figure 3.2. Typical Core Protection Safety Limit

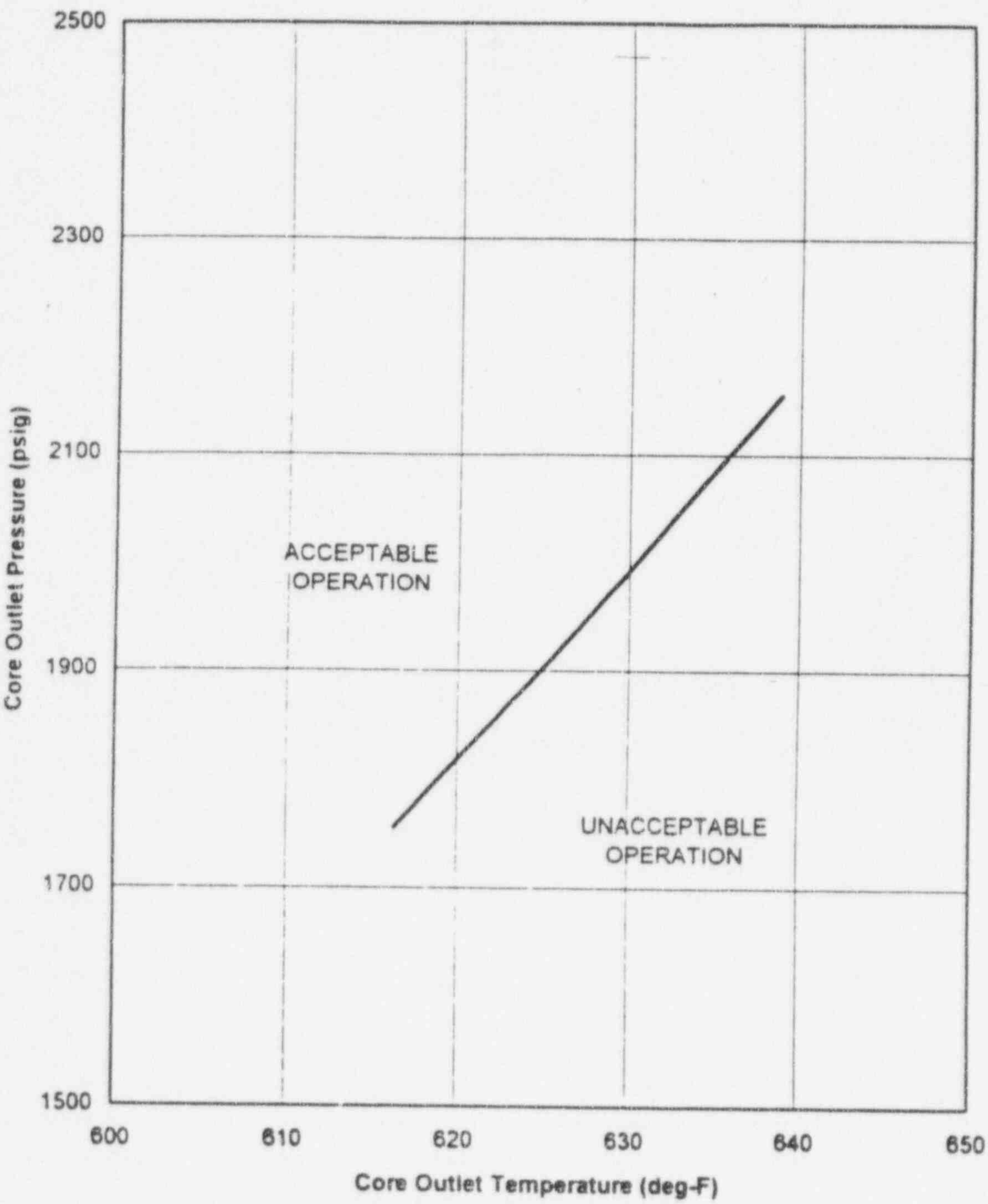


Figure 3.3 Typical Core Protection Safety Bases

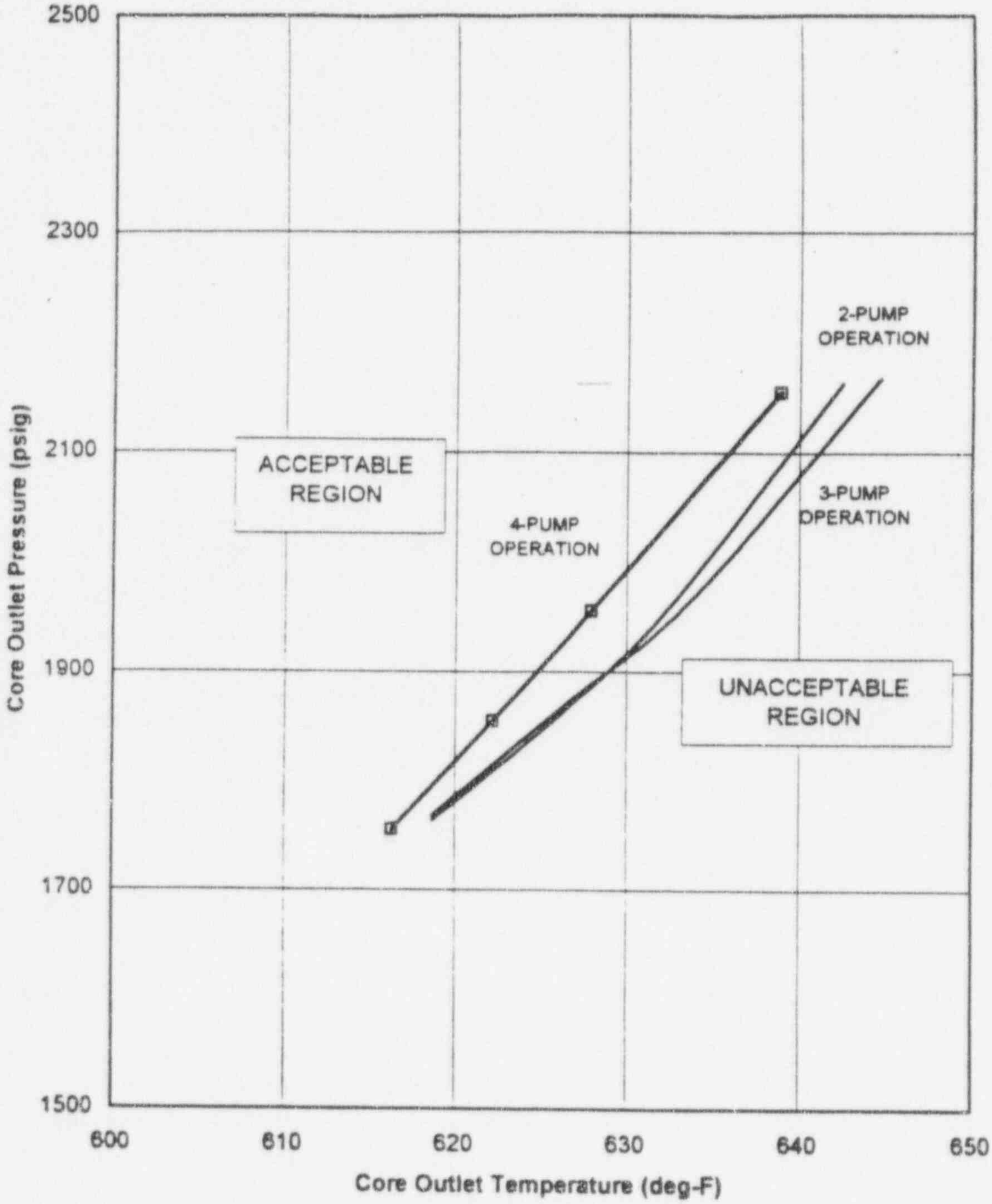


Figure 3.4 Determination of Flux/Flow Protective Limits

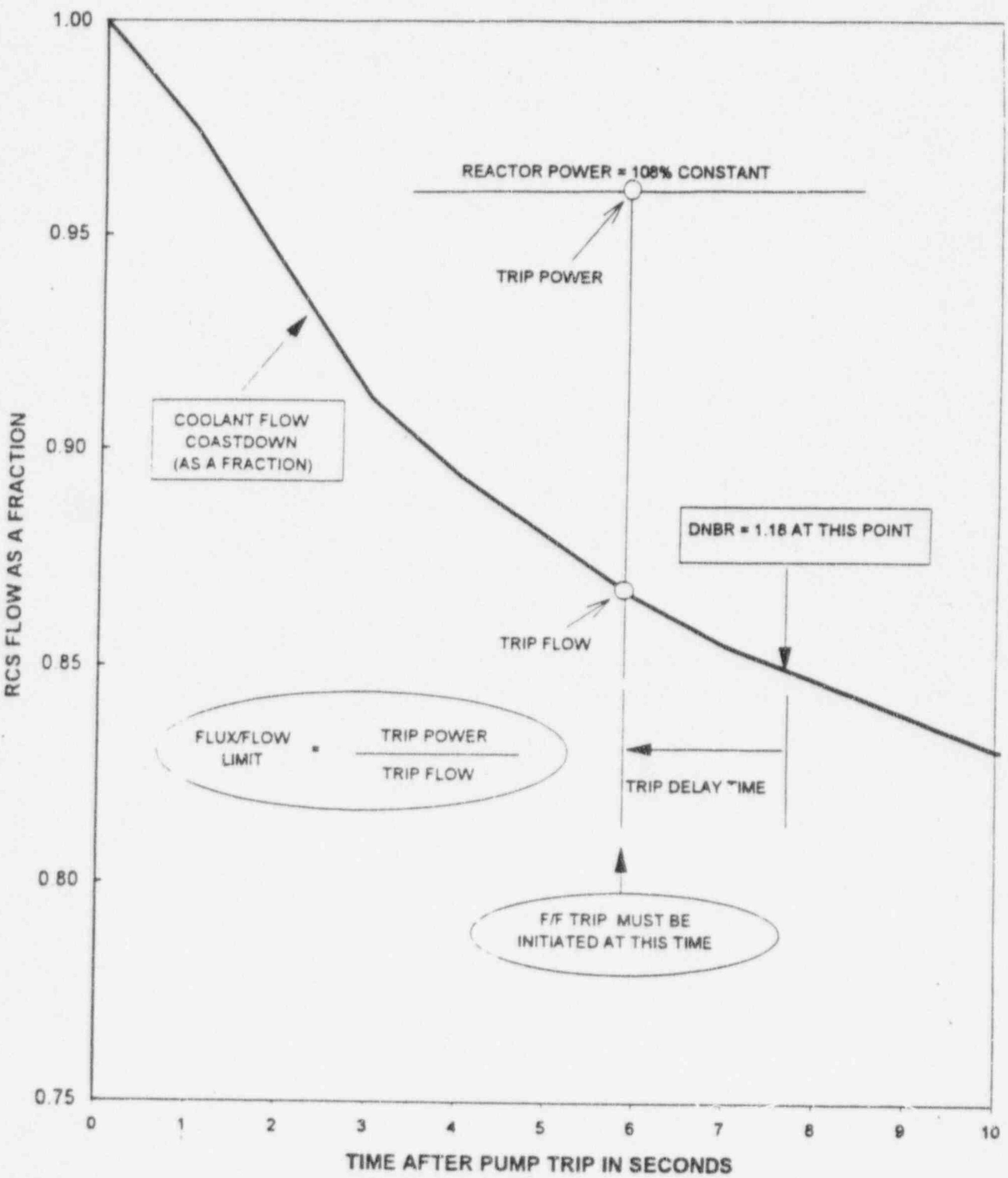


Figure 3.5 A Typical Set of MAP Limits

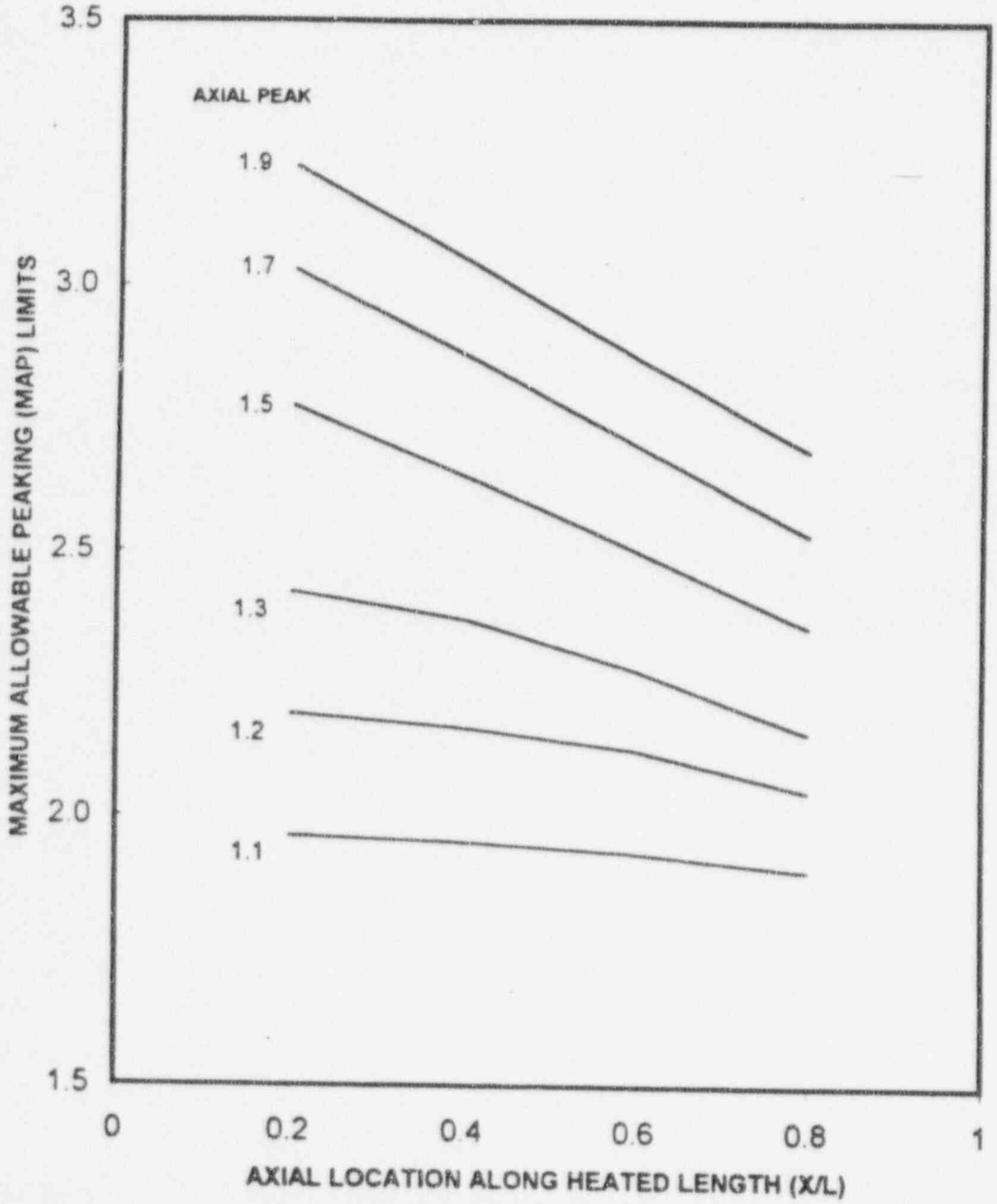
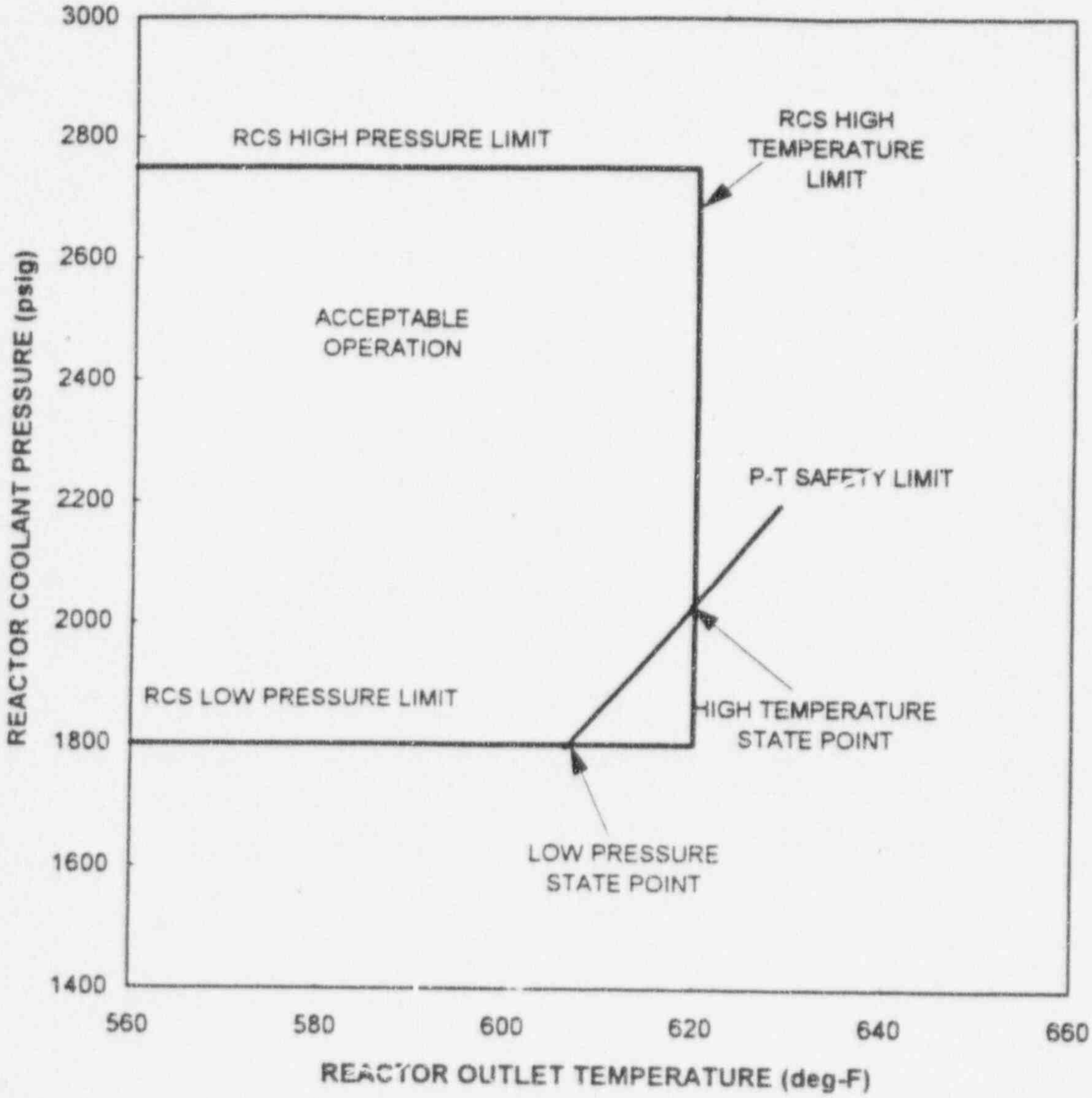


Figure 3.6 P-T State Points for the RPS MAP Analysis



4.0 MANEUVERING ANALYSIS

The main purpose of the maneuvering analysis is to determine the power distributions during all modes of power operation allowed by Technical Specifications. The power distribution data consists of two and three dimensional power peaking factors, linear heat rates (LHR), and axial power profiles; all being functions of operating parameters (or process variables). The axial power profile is represented by either the axial core imbalance or axial power offset, which are defined in Section 3.6.

The process variables characterize and control the radial and axial power distributions in the reactor core. The process variables consist of: control rod assemblies (CRAs) and APSRAs positions, xenon distribution, fuel burnup, axial power imbalance, power level, and core quadrant power tilt. According to a requirement of 10CFR50.36 (Reference 10a), the safety limits, reactor trip setpoints, and LCO alarm limits must be imposed on these process variables for the safe operation of the reactor. The power-distribution related safety limits and setpoints are determined by comparing the power distribution data against numerically quantified acceptance limits. They are the centerline fuel melting (CFM) limit, LOCA kw/ft limit, and DNBR limit. This comparison of the local power distribution data against the acceptance limits is called the "margin evaluation," which is also included in this section. However, the current core monitoring system at TMI-1 is not capable of monitoring local parameters such as 3D pin powers. Hence, RPS trips and LCO alarms monitor the core global parameters of power imbalance, core quadrant power tilt and control rod index. Therefore, maneuvering analysis must also establish a relationship between the bounding acceptable local power distribution data and the limiting global parameters.

4.1 Power Distribution Analysis

The operating parameters affecting the power distribution are:

- fuel assembly burnup
- CRA and APSRA positions
- core power transients and corresponding CRA and APSRA movement causing a xenon oscillation
- power level
- RCS pump operating mode (four-pump or partial pump operation)

The power distribution maneuvering analysis is performed using the SIMULATE-3 code (Reference 13). All of the above operating parameters are varied in this analysis as described below.

4.1.1 Control Rod Scan at Various Cycle Depletion Points

The control rod scan at various depletion points generates local and global power distribution data. The CRAs (i.e., regulating CRA Groups 5, 6, and 7) and APSRAs are positioned at various points between fully inserted and total withdrawal. In this manner, power distributions for all possible control rod combinations are obtained.

[ ]

4.1.2 Control Rod Scan in the Design Power Maneuver

The purpose of the control rod scan in the design power maneuver is to obtain the worst power distribution data by scanning control rods (regulating groups and APSRAs) at the worst xenon distribution conditions (at the "maximum" and "minimum" xenon conditions; described below).

4.2 Margin Evaluation for the Setpoint Determination

4.2.1 Peaking Augmentation Factors

As mentioned in Section 4.0, the setpoints and alarm limits are determined by comparing the power distribution data against various acceptance limits. However, the power peaking factors and LHRs (in kw/ft) from the power distribution analysis are not "real" values. This data is generated using SIMULATE-3 which has model uncertainties. These uncertainties are determined statistically, based on a 95/95 protection criteria, and are defined as nuclear reliability factors (Reference 5). For GPU Nuclear, values of the nuclear reliability factors for the radial pin power and for the total (3-dimensional) pin power are 1.038 and 1.055, respectively. The peaks in power distribution data are increased using these model uncertainties to obtain the "real" power distribution data to determine and compare to the limits and setpoints.

The other types of uncertainties applied to the power distribution data are: (1) power spike factors (discussed in Section 3.2.6), and (2) engineering hot channel factors. The engineering hot channel factors are described in Section 3.2.5 and consist of the spacer grid factor and local heat flux factor.

In addition to the above uncertainty factors, the power peaking factors in the margin evaluation are adjusted for the core quadrant power tilt factor (typically 1.07 to 1.08), which is discussed in Section 5.5.3. The application of the above peaking augmentation factors

[ ]

4.2.2 Margin Analysis Against CFM, DNBR and LOCA Limits

A safety limit for a process variable is determined at the value where the margin to the acceptance limit is zero, or a predetermined target margin value. The target margin is set to some value greater than zero (typically 1% to 5%) to account for such uncertainties as the actual cycle length and/or possible changes in the reload design.

The RPS safety limits are determined based on both the CFM and DNBR margins, while the LCO alarm limits are determined based on LOCA and DNBR margins in addition to shutdown margin (SDM) and ejected rod worth (ERW) margins, of which both SDM and ERW are described in Sections 2.3.3, 2.3.4 and 5.5.2. As described in Section 3.6, the DNBR limit is used in two areas: (1) RPS MAP limits for the RPS setpoints, and (2) LCO MAP limits for LCO alarm limits.

The equations for the margin analyses are given below:

[ ]

5.0 SETPOINT METHODOLOGY

5.1 Introduction

To satisfy the requirements given in 10CFR50.36 (Reference 10), the operation of power reactor should be in accordance with the Technical Specifications (TS), which are established from applicable design evaluations, safety analyses, and other considerations. Included in the Technical Specifications (Reference 19) are safety limits, RPS trip setpoints (or limiting safety system settings), LCO alarm limits (or limiting conditions for operation), surveillance requirements, and identification of design features and administrative controls. Technical Specifications generally affected by a typical reload design are the (1) core safety limits, (2) RPS trip setpoints, and (3) LCO alarm limits. As mentioned in Section 1, many of the reload-dependent items are removed from TS and included in the COLR (Reference 2).

The core safety limits define limits on the values of important process parameters to protect the integrity of fission product barriers such as fuel cladding and the reactor vessel. The important process parameters consist of the RCS pressure, RCS temperature, RCS flow rate, reactor power, and core power distributions. The RPS trip setpoints are settings on the measurable (and controllable) parameters, which include the measuring instrument's detection uncertainties, that allow an automatic protective action (i.e., tripping the reactor) before an operating parameter exceeds the safety limit. The typical RPS trip instrument errors and trip delay times are given in Table 5.1.

In TMI-1, the measurable parameters for the RPS trip function consist of following:

1. Reactor outlet pressure measured at the pressure tap located in the RCS hot leg
2. Reactor outlet temperature measured at the temperature gauge located in the RCS hot leg
3. RCS flow rate
4. Reactor power measured by the out-of-core detectors (OCDs)
5. Core axial power imbalance measured by OCDs.

The LCO alarm limits are established to assure that transients or accidents which are initiated from the limiting conditions do not violate appropriate acceptable limits. In other words, the LCO limits define the limiting operation parameters to assure that initial conditions assumed for accident analyses are not breached. The LCO alarm limits for TMI-1 consist of control rod assembly (CRA) insertion limits, core quadrant power tilt limits, and power imbalance limits.

The following sections describe the safety criteria and methodology of how these limits and setpoints are determined.

5.2 Protection System Maximum Allowable (PSMA) Trip Setpoints (Pressure-Temperature Trip Setpoints)

The PSMA trip setpoints (or pressure-temperature trip), as shown in Figure 5.1 and also given in Figure 2.3-1 of TMI-1 TS, define the P-T trip envelope consisting of the high and low reactor pressure trip setpoints, high reactor outlet temperature trip, and variable low pressure (VLP) setpoints.

5.2.1 Variable Low Pressure (VLP) Trip

The VLP trip provides primary steady-state DNB protection. The VLP trip also provides DNB protection for transients resulting in RCS depressurization, specifically, a steam line break, a letdown line rupture, and the loss of coolant accident (LOCA). This protection is provided through monitoring hot leg pressure and temperature conditions and tripping the reactor when the P-T conditions reach the design DNBR of 1.18. As shown in Figure 5.1, the VLP trip setpoints are a functional dependency: low pressure as a function of the measured core outlet temperature.

The VLP setpoints are determined by applying instrumentation uncertainty and other correction factors to the P-T core protection limits, whose derivation is discussed in Section 3.4. The equation is given below:

The PSMA trip setpoints in the current TMI-1 TS (Reference 19) do not include the VLP trip setpoints. This is because, with the application of cross flow model in the T-H analysis (Reference 18), these VLP setpoints are located outside of the acceptable operating region formed by high and low pressure, and high outlet temperature trip setpoints. This removal of the VLP setpoints from the PSMA setpoint envelope is also demonstrated by using the VIPRE-01 analysis as given in Appendix A of this report. However, the removal of the VLP setpoints should be confirmed by analysis for every reload fuel cycle since the VLP setpoints along with the core protection safety limit in Figure 3.2 are dependent on the reload design.

5.2.2 High RCS Pressure Trip

The reactor coolant system (RCS) serves as a barrier to prevent radionuclides in the reactor coolant from reaching the environment. The high RCS pressure safety limit is defined as 110% of the RCS design pressure in accordance with ASME Boiler and Pressure Vessel Code, Section III. With a design pressure of 2500 psig, the safety limit is 2750 psig (References 1 and 19). The initial hydrostatic test conducted at 3125 psig (125% of design pressure) verified that no failure of the primary system pressure boundary occurred. This indicates that the safety limit is chosen with a large margin to the point of actual system failure (Reference 19).

The high RCS pressure trip provides protection for the high pressure safety limit of 2750 psig and maintains the integrity of the RCS by initiating a reactor trip during pressure-increasing events. The FSAR transients which use this trip are the startup event, control rod withdrawal accident at rated power, moderator dilution event, control rod ejection accident, and loss of electric power.

The analysis criteria for the determination of the high RCS pressure setpoint are:

1. The setpoint shall lie within the detection range of instrumentation.

2. The setpoint shall assure a reactor trip 95% of the time at a 95% confidence level, and shall account for instrumentation uncertainties.

3. The setpoint shall be below the pressurizer safety valve setpoints to avoid the lifting of the pressurizer safety valves prior to a reactor trip.

4. The setpoint shall ensure the peak RCS pressure during reactor transients remain below the high RCS pressure safety limit (2750 psig).

The trip setpoint determination process involves accident analyses and instrument error adjustments, which are illustrated in Figure 5.2. For a given accident analysis result (or peak pressure in Figure 5.2), the difference between the peak pressure and the safety limit represents a quantifiable margin, while the difference between the safety limit and actual barrier integrity can be considered as unquantifiable margin. The accident overshoot represents the continued pressure rise toward the safety limit even after a reactor trip due to the inertia of the system.

The accident analysis setpoint differs from the TS trip setpoint by the instrument uncertainties. The equation to determine the TS setpoints based on the accident analysis is given below:

[ ]

5.2.3 Low RCS Pressure Trip

The low RCS pressure trip provides protection against DNB and CFM during steady state operation and pressure-decreasing transients. The transients for this trip are the small break loss of coolant accident (SBLOCA) and steam generator tube rupture (SGTR) accident.

[ ]

The trip setpoint is determined based on the following equation:

[ ]

5.2.4 High RCS Temperature Trip

The high RCS temperature trip setpoint is established to prevent an excessive reactor coolant temperature during both steady state operation and transients. The trip is not used as the primary trip function for any accident in FSAR accident analysis. However, this trip provides a backup protection for RCS overheating events.

The trip setpoint is determined based on the following equation:

[ ]

The calibrated detection range of the RPS temperature channels is 520°F to 620°F (Reference 19). Since the high RCS temperature trip is not used for the FSAR accident analysis, the accident analysis setpoint, XT_R, is assumed to be at the upper limit of the temperature measurement instrumentation (620°F).

5.3 High Flux Trip

The high flux trip provides DNB and CFM protection during both steady state operation and transient conditions. The transients for which the high flux trip serves are the startup accident, the control rod withdrawal accident (RWA), the control rod ejection accident (REA), and the steam line break accident (SLBA).

The high flux trip setpoint is determined based on the accident analysis setpoint and by accounting for the instrumentation errors. The original value of the accident analysis setpoint was 114% FP. In order to incorporate the fuel densification effect in the early 1970's, the accident analysis setpoint was reduced to 112% FP from 114% FP (Reference 22).

The equation for the determination of the trip is given below:

[ ]

[ ]

5.4 Protection System Maximum Allowable (PSMA) Setpoints for Axial Power Imbalance

The PSMA setpoints for axial power imbalance, shown in Figure 5.3 is also called the power-imbalance-flow (PIF) trip setpoints. The PIF trip setpoints are a combination of the flux/flow (F/F) trip and the axial power imbalance trip (Figure 5.3). The analysis process for the PIF setpoints is broken down into two main tasks: (1) calculation of flux/flow setpoint, and (2) calculation of power-imbalance envelope.

5.4.1 Flux/Flow Trip

As described in Section 3.5.1, the F/F trip, during four pump operation, provides DNB protection for both steady state operation and decreasing flow events. These events include RC pump coastdowns and the locked rotor accident. However, while in partial pump operation the F/F provides overpower power protection for increasing power events.

An example of this type event is a control rod withdrawal accident (RWA).

The F/F trip setpoint is derived by adjusting instrument errors to the F/F protective limits, which are discussed in Section 3.5.1. The relationship between these two is given by:

[ ]

5.4.2 Axial Power Imbalance Core Protection Safety Limits

The axial power imbalance (API) core protection safety limits, shown in Figure 5.4, establishes the maximum allowable overpower as a function of the axial power imbalance for various pump operating conditions. The governing design criteria is from General

Design Criterion 10 (Reference 10), which requires that the fuel must not sustain damage as a result of normal operation or anticipated operational occurrences. Determining the API safety limits, this criterion is represented by the centerline fuel melting (CFM) kw/ft limit to protect the fuel and the design DNBR limit of 1.18 for the protection of fuel cladding. Therefore, operation within the API protection limits ensures that both the maximum allowable linear heat rate (LHR) based on CFM limit and maximum power peaking based on DNB criterion will not be exceeded.

The analysis methods for both CFM and DNB margins are described in Section 4. The maneuvering analysis provides the power distribution (power peaking, LHRs, and axial power imbalance and offset) at overpower conditions as a function of core burnup, control rod positions, and xenon distribution. Applying both uncertainty factors and peaking augmentation factors (Section 4.2), the margins are determined by comparing LHRs with the CFM limit (typically 20.5 kw/ft), and power peaking factors with the RPS MAP limits (Section 3.6).

The core offset limits are determined by plotting the margin data on the margin-to-core offset plane as illustrated in Figure 5.5. This analysis procedure is repeated for different power levels, core burnup statepoints from BOC to EOC , and RC pump combinations. Typically, the core offset limits are specified by pairs of points (negative and positive limits as shown in Figure 5.5) at power levels of 112, 100, 80, and 50 %FP.

These core offset limits are then converted to axial power imbalance limits (Figure 5.4) by using the relationship:

$$\theta_{IMB} = \theta_{OFF} * P \tag{Eq. 5.7}$$

where:

P = fraction of rated power,

θ_{IMB} = axial power imbalance

θ_{OFF} = core power offset

5.4.3 PSMA Setpoints for Axial Power Imbalance

The reactor trip based on the PSMA setpoints for axial power imbalance (to be referred as the PIF trip setpoints hereafter) are initiated by combining the signals from the out-of-core detector (OCD) and flux/flow (F/F) instrumentation. The OCD determines both power level and power imbalance while the F/F instrumentation compares power level against RCS flow. The PIF trip setpoints are derived from the PIF safety limits described in the previous subsection by adjusting uncertainties associated with both OCD and F/F instrumentation, that is, instrument uncertainties for the measurement of power level, imbalance, and flow.

The power adjustment is as follows:

[ ]

The detectability envelope shown in Figure 5.6 represents a saturation of the equipment signal. The setpoints outside of this envelope cannot be detected due to equipment limitations. To provide a reactor trip, the RPS must have the capability of detecting the trip condition. Therefore, the PIF trip setpoint must lie inside the region formed by the flux/flow setpoint and detectability curve as shown in Figure 5.6.

5.5 Limiting Conditions for Operation (LCO) Alarm Limits

A requirement of 10CFR50.36 (Reference 10a) is that LCOs be placed on process variables required for safe operation of the plant. Regulating control rod position, APSRA position, axial power imbalance, and quadrant power tilt are process variables that characterize and control the radial and axial power distribution of the reactor core. This section describes power-distribution related LCOs in the COLR, which consist of:

1. Power imbalance alarm limits
2. Control rod insertion (CRI) limits
3. Core quadrant power tilt limits.

The process variables reflected in the above administrative limits are monitored and controlled during power operation to ensure that the power distribution does not violate these limits. If the LCO limits are violated, a short time is allowed for corrective action (typically four hours). If the LCO limits are not restored within the required time, a significant power reduction is required. Operation beyond the shutdown margin CRI limit is not permitted as it is a violation of the TS requiring adequate shutdown capability. If shutdown margin CRI limit is exceeded, the reactor must be shut down immediately.

The safety criteria to set the LCOs are as follows:

1. The peak cladding temperature during a LOCA must not exceed a limit of 2200°F (10CFR 50.46, Reference 10b).

2. DNB protection during a loss of forced reactor coolant (LOCF) flow accident such as an RC pump trip (GDC 10, Reference 10c)

3. During a control rod ejection accident (REA), the fission energy in the fuel must not exceed 280 cal/g (GDC 28, Reference 10f).

4. The control rods must be capable of shutting down the reactor with a minimum required shutdown margin (SDM) with the highest worth control rod assembly stuck fully withdrawn (GDC 26, Reference 10e).

These safety criteria preserve the accident initial conditions assumed in the safety analysis related to the core power distribution and reactivity conditions. As described in Section 1.4, the above GDC requirements do not provide the numerically quantified acceptance limits. The acceptance limit of the LOCA criterion in Item 1 above is numerically defined, as shown in Figure 5.7, by the maximum allowable linear heat rate (kw/ft) versus axial position along a fuel assembly. The DNB protection during a LOCF is represented by the LCO MAP limits discussed in Section 3.6.2. The REA criterion in Item 3 is represented by a maximum ejected control rod worth of 0.65% $\Delta k/k$ at hot full power (HFP), and 1.0% $\Delta k/k$ at hot zero power (HZP). In Table 5.2, the ejected control rod worth (ERW) limits are given as a function of power level. The SDM criterion is numerically interpreted as 1% $\Delta k/k$ shutdown margin at HZP with the maximum worth control rod assembly fully withdrawn (References 4 and 23).

5.5.1 Power Imbalance Alarm Limits

The power imbalance limits for the normal operating control (or LCO) are based on LOCA kw/ft limits (Figure 5.7) and LOCF DNBR criteria as represented by the LCO MAP limits (Section 3.6.2). The LOCA and LCO MAP margin analyses are discussed in Section 4.2.

The core offset limits are determined in a similar manner as described in Section 5.4.2 and as illustrated in Figure 5.5. Typically, the positive offset limits are determined by the MAP limits, while the negative offset limits by the LOCA kw/ft limits. This is because the DNBR condition (or LCO MAP) becomes worse when the axial power profile is a top-peaked shape (positive offset) which occurs at the minimum xenon condition during the recovery of the design power transient (Section 4.1.2).

The core offset limits are determined at different power levels and different core burnup intervals. The offset limits are then adjusted by incorporating the measurement instrument errors and are converted to the imbalance alarm limits (Figure 5.8) by using the equation given in Section 5.4.2. Three sets of alarm limits are determined based on three different detector systems: full incore system (FIS), minimum incore system (MIS), and out-of-core detector (OCD) system, which are categorized depending upon the number of operable incore detectors (Reference 19). The instrument error is also a function of the core offset and the depletion of the incore detector. The method to determine the offset measurement errors for these instrumentation system conditions is discussed in detail in Reference 23.

5.5.2 Control Rod Insertion Limits

As described in Section 2.3.3, the TMI-1 control rod assemblies are divided into seven groups. Each control rod group position is measured in terms of percent withdrawn (%WD). Groups 1 through 4 are the safety control rod assemblies and are always 100 %WD above the HZP condition. Groups 5, 6, and 7 (the regulating groups) are used to maneuver the plant between 0 to 100% full power. To increase or decrease core power the regulating groups are sequentially withdrawn or inserted (5, 6, and 7 for withdrawal or 7, 6, and 5 for insertion) with an overlap of 25 %WD between each group.

The bases of the control rod index (CRI) limits are the LOCA kw/ft limits, ejected control rod worth (ERW) limits (Table 5.2), and the 1% shutdown margin (SDM). The CRI limits in Figure 5.9 establish the allowable operating range of the full-length control rod assemblies (regulating groups) positions regardless of the gray APSRA positions.

LOCA margin (Section 4.2.2) and core offset contour curves are given in Figures 5.10 and 5.11, respectively, as a function of APSRA and CRA positions. As illustrated in Figures 5.10 and 5.11, the CRI limits based on LOCA limits in conjunction with the core offset limit, are determined to provide complete protection of the target margin.

The ejected control rod worths are determined based on the methodology given in Section 2.3.3.3 and an example is given in Figure 5.12 as a function of the inserted control rod worth. The ERW limits based on accident analysis are given in Table 5.2 as a function of reactor power. The analysis limits of ERW are then chosen by applying a 15%

conservatism factor. With the ERW analysis limits, the allowable inserted control rod worth is determined as shown in Figure 5.12. The allowable inserted control rod worth is then converted to the control rod position limits by using the integral control rod worth curve given in Figure 5.13.

The allowable inserted control rod worth, preserving a 1 %Δk/k SDM, is calculated in accordance with the methodology given in Section 2.3.4 and is summarized in Table 5.3. The CRI limits based on the shutdown margin are then determined by the relationship between the allowable inserted control rod worth and an integral control rod worth curve as illustrated in Figure 5.13.

Once the CRI limits are determined based on the above three criteria, the limits are then adjusted by applying instrument uncertainties of both power level measurement [ ] [ ] and control rod position measurement [ ].

5.5.3 Core Quadrant Power Tilt Limit

The core quadrant power tilt is defined as:

$$\Omega = 100 \times (\phi_Q/\phi_{AV} - 1) \qquad \textbf{(Eq. 5.9)}$$

where:

Ω = core quadrant power tilt (%)

ϕ_Q = power in any core quadrant

ϕ_{AV} = average power in all quadrants.

Quadrant tilts indicate deviations from core radial symmetry, therefore, disruptions in power distributions. The quadrant tilt can be caused by various operating parameter changes such as control rod(s) out of sequence, a dropped control rod, non-symmetric core burnup gradient, and reactor coolant temperature mismatch. As quadrant tilt increases the power peaking factor increases. The relationship between the tilt and peaking is typically [ ] [ ] peaking increase per 1% tilt increase, which is determined assuming the above-mentioned operating parameter changes.

[ ]

The quadrant tilt alarm limits are determined by applying instrument errors for three different quadrant tilt measurement systems: (1) full incore detector system, (2) minimum incore detector system, and (3) out-of-core detector system. The instrument error adjustments for the tilt alarm limits are illustrated in Table 5.4.

Table 5.1
Typical RPS Trip Instrument Errors and Response Time

Table 5.2
Ejected Control Rod Worth Limits

POWER (% FP)	ACCIDENT ANALYSIS ERW LIMIT (% $\Delta k/k$)	ADJUSTED ERW LIMIT(1) (% $\Delta k/k$)
100	0.65	0.55
50	0.82	0.70
15	0.95	0.81
0	1.00	0.85

NOTES:
(1) 15 % conservatism factor is applied to the accident analysis ERW limit.

Table 5.3
Calculation of Shutdown Margin and Allowable Inserted Control Rod Worth

[ ]

Table 5.4
Determination of Error-Adjusted Quadrant Tilt Alarm Setpoints

[ ]

Figure 5.1 Determination of Protection System Maximum Allowable Setpoints

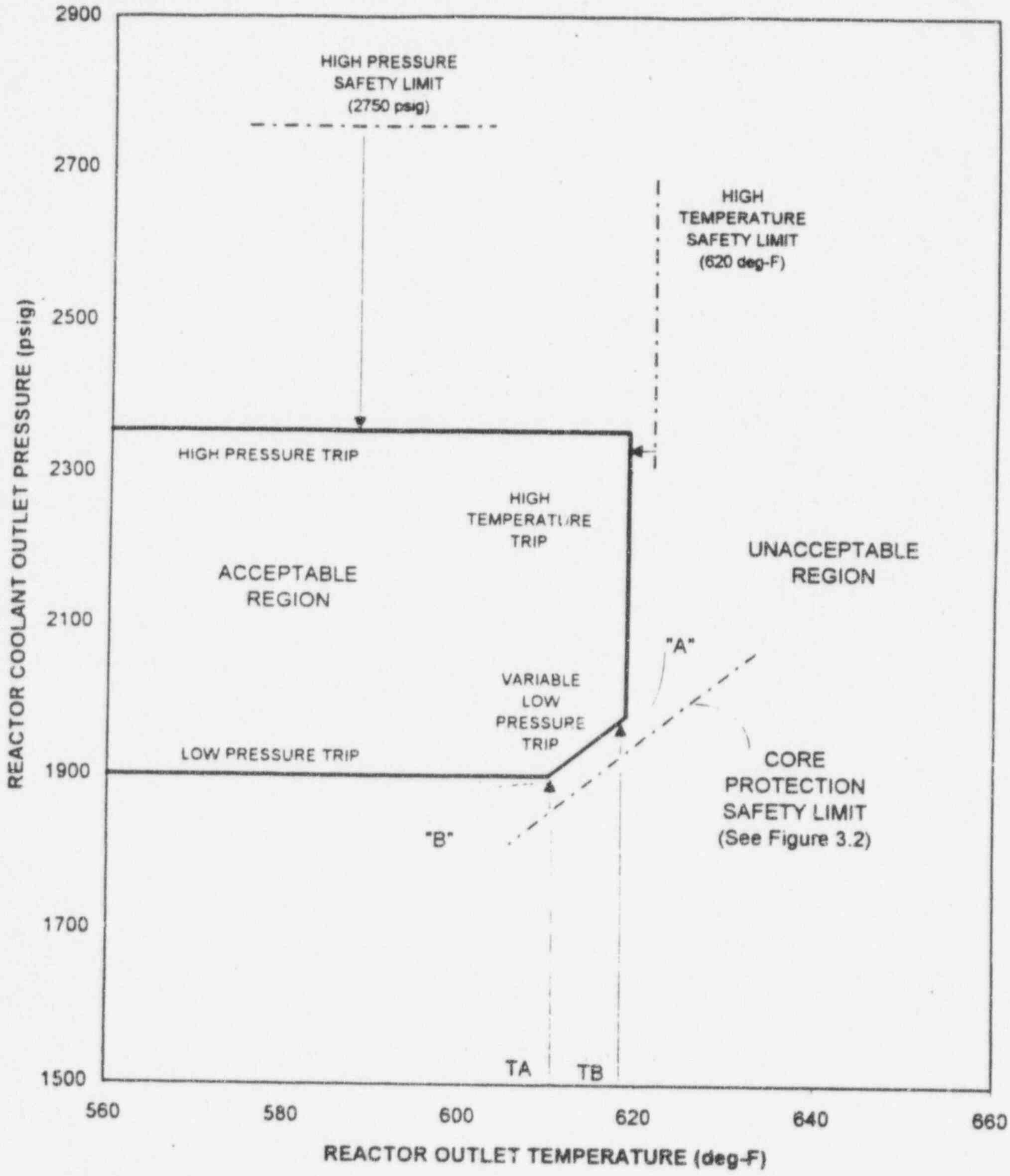


Figure 5.2 Relationship of Trip Setpoint to Accident Analyses and Instrument Errors

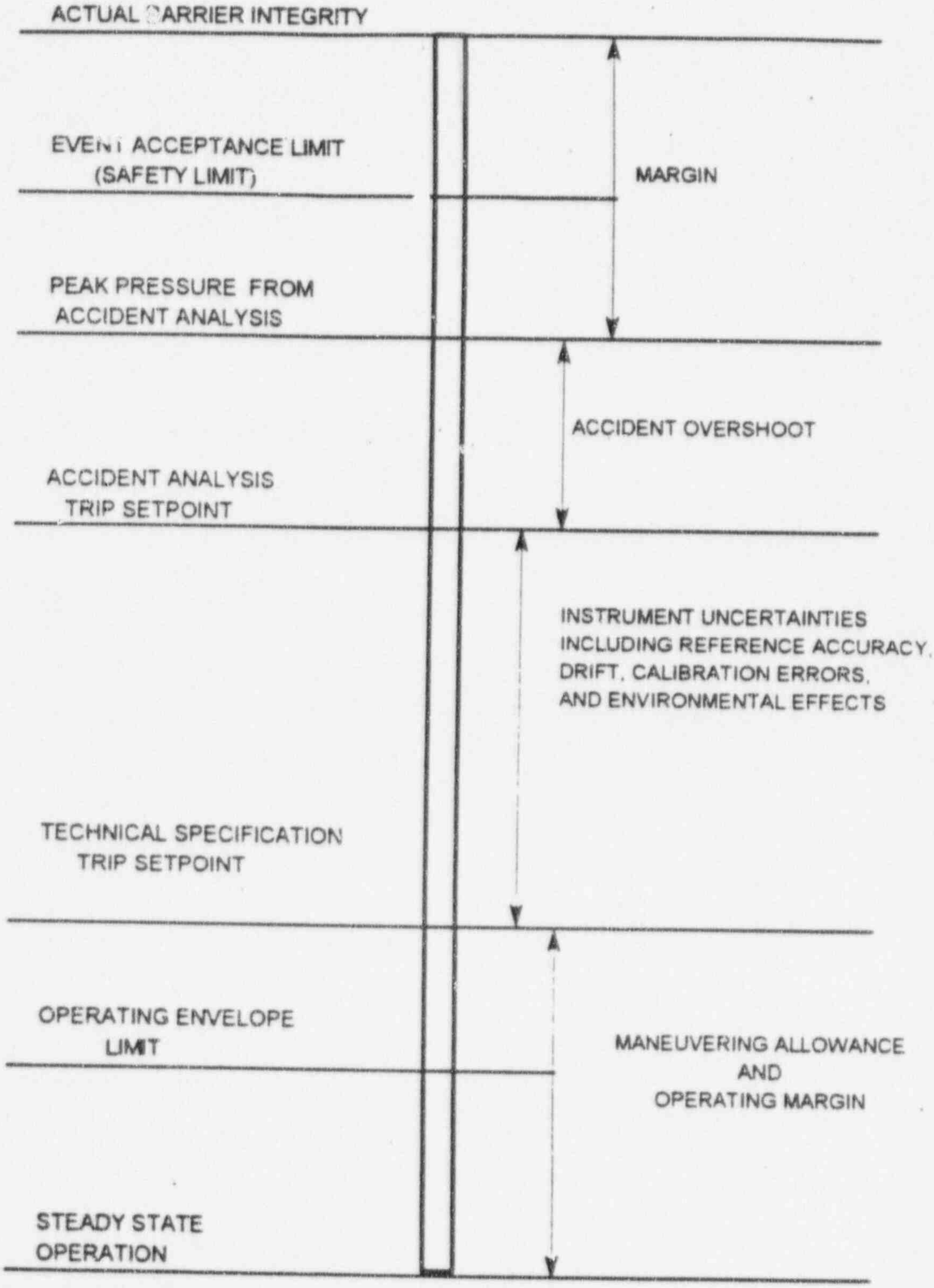


Figure 5.3 Typical Power Imbalance Flow Trip Setpoints

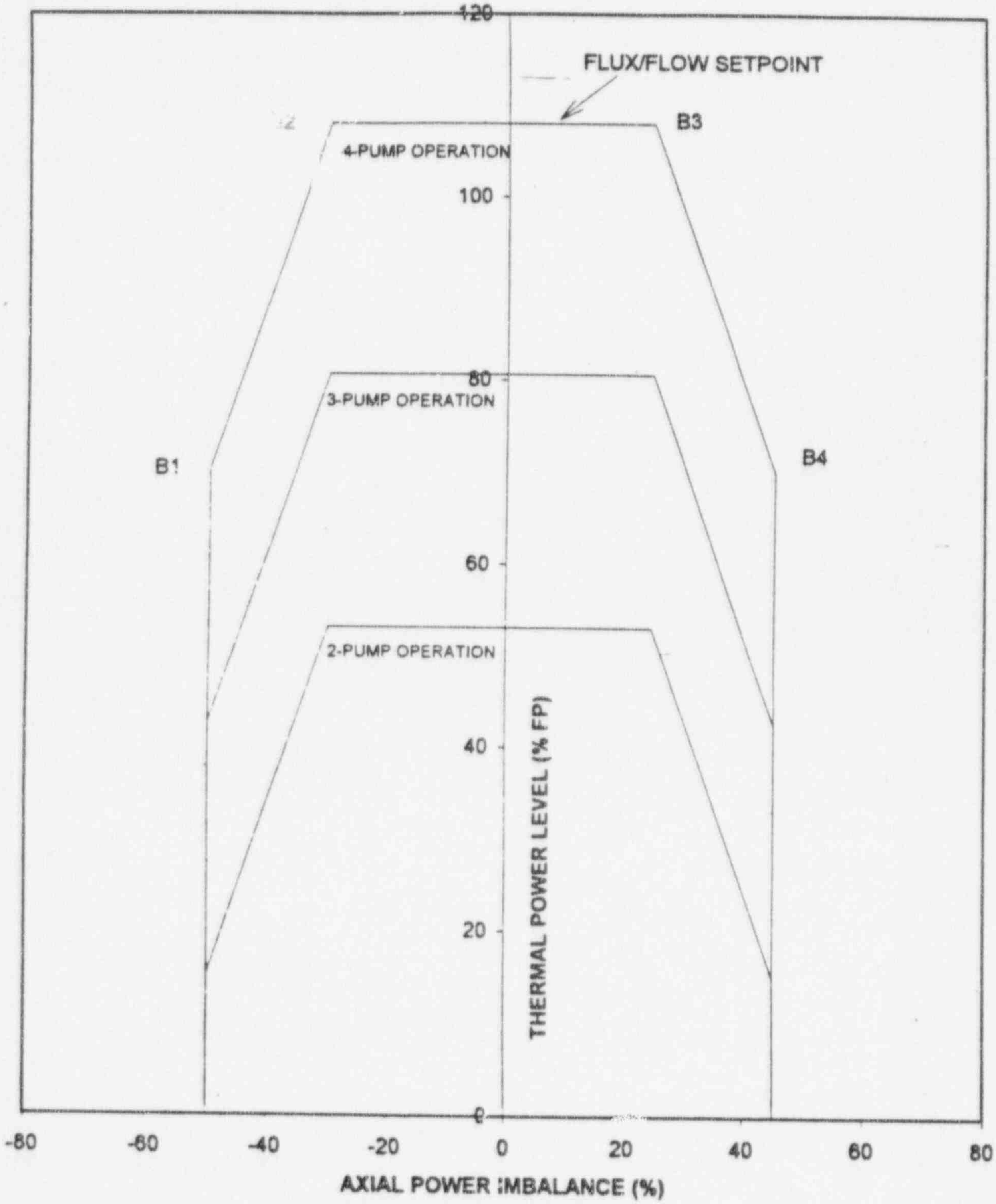


Figure 5.4 Typical Power-Imbalance-Flow Safety Limits

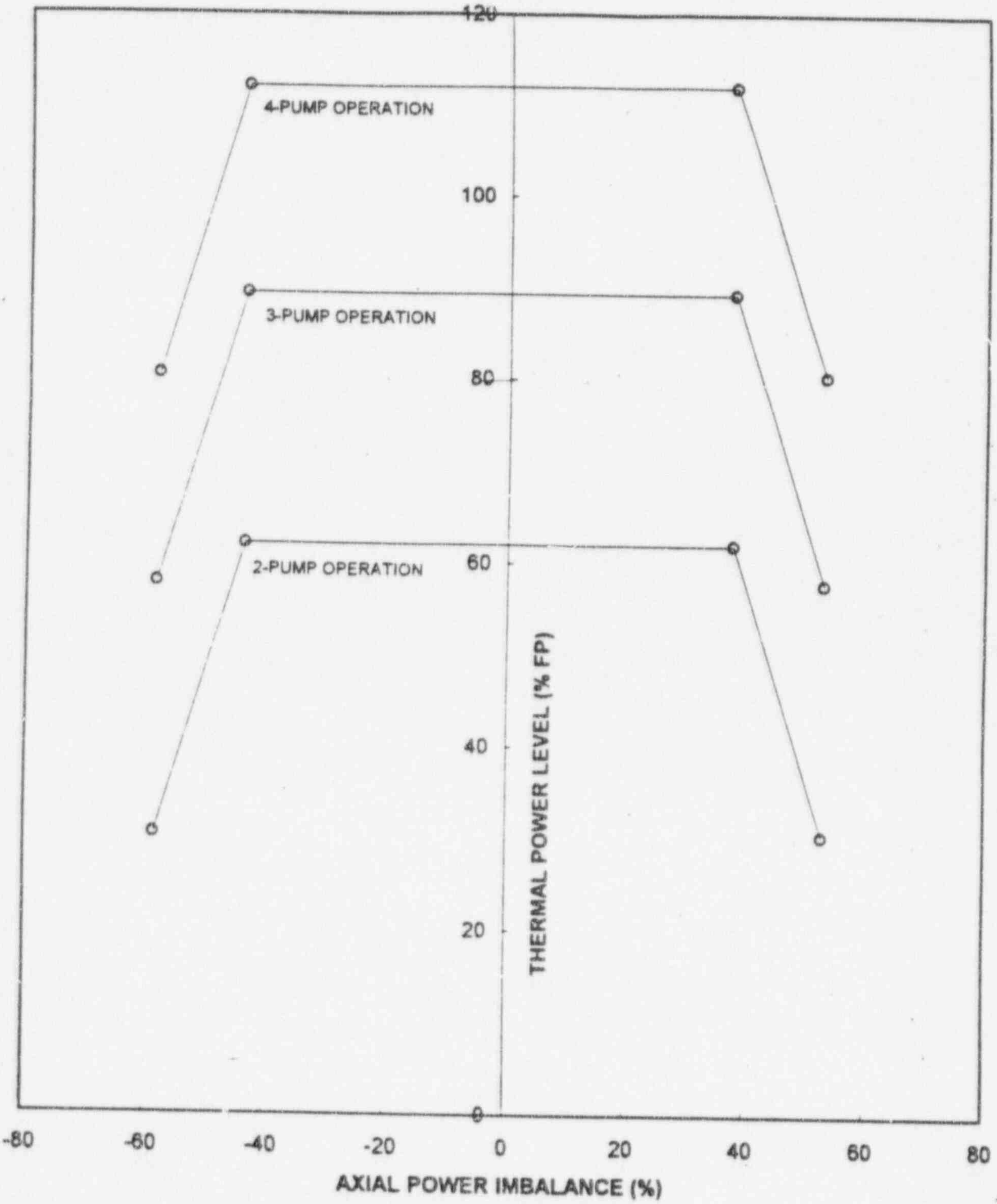


Figure 5.5 Determination of Axial Offset Limit Based on CFM and DNB Margins

Figure 5.6 Power-Imbalance Detectability Envelope

Figure 5.7 Typical LOCA Limited Linear Heat Rate

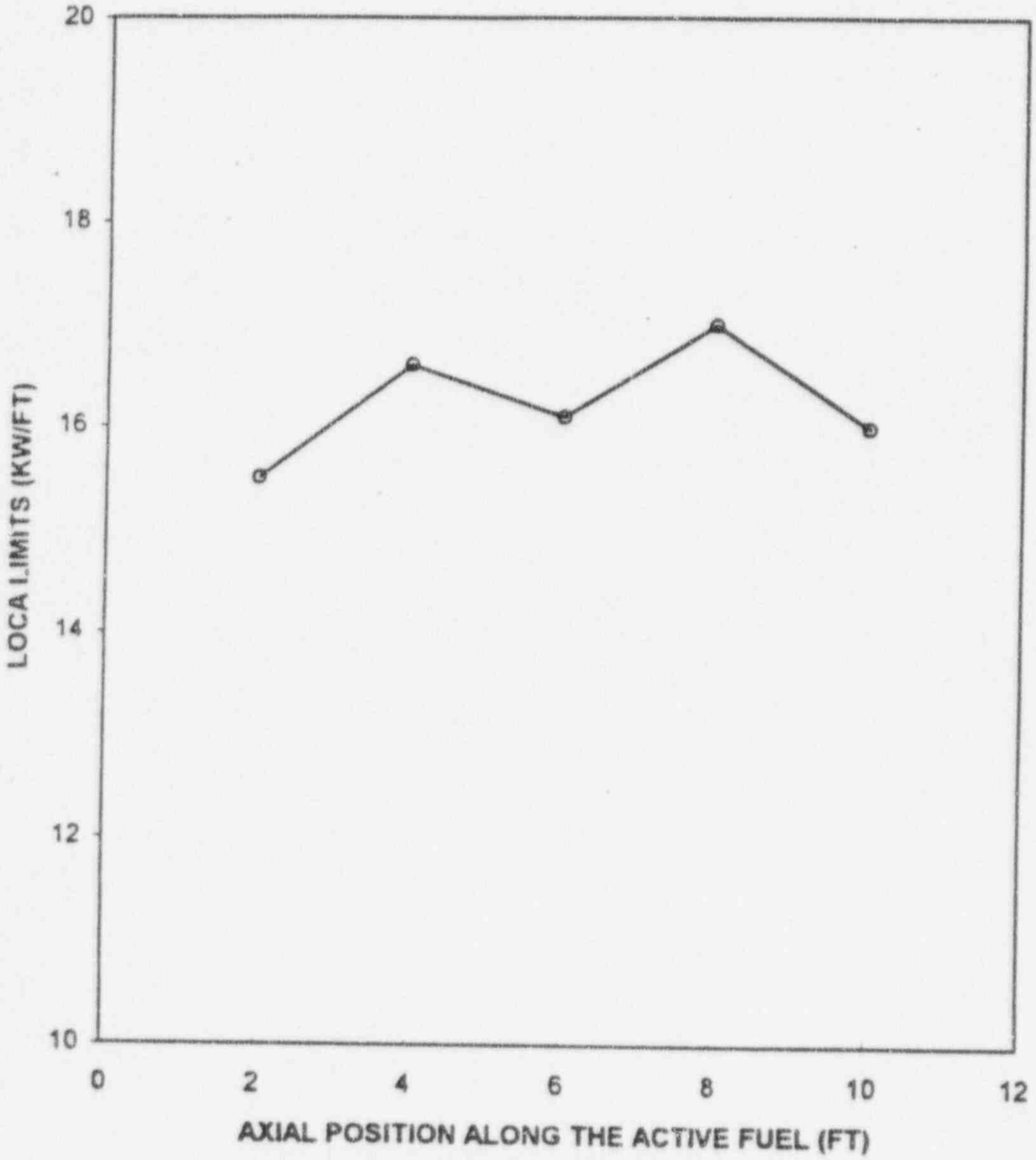


Figure 5.8 Typical Power Imbalance Alarm Limits

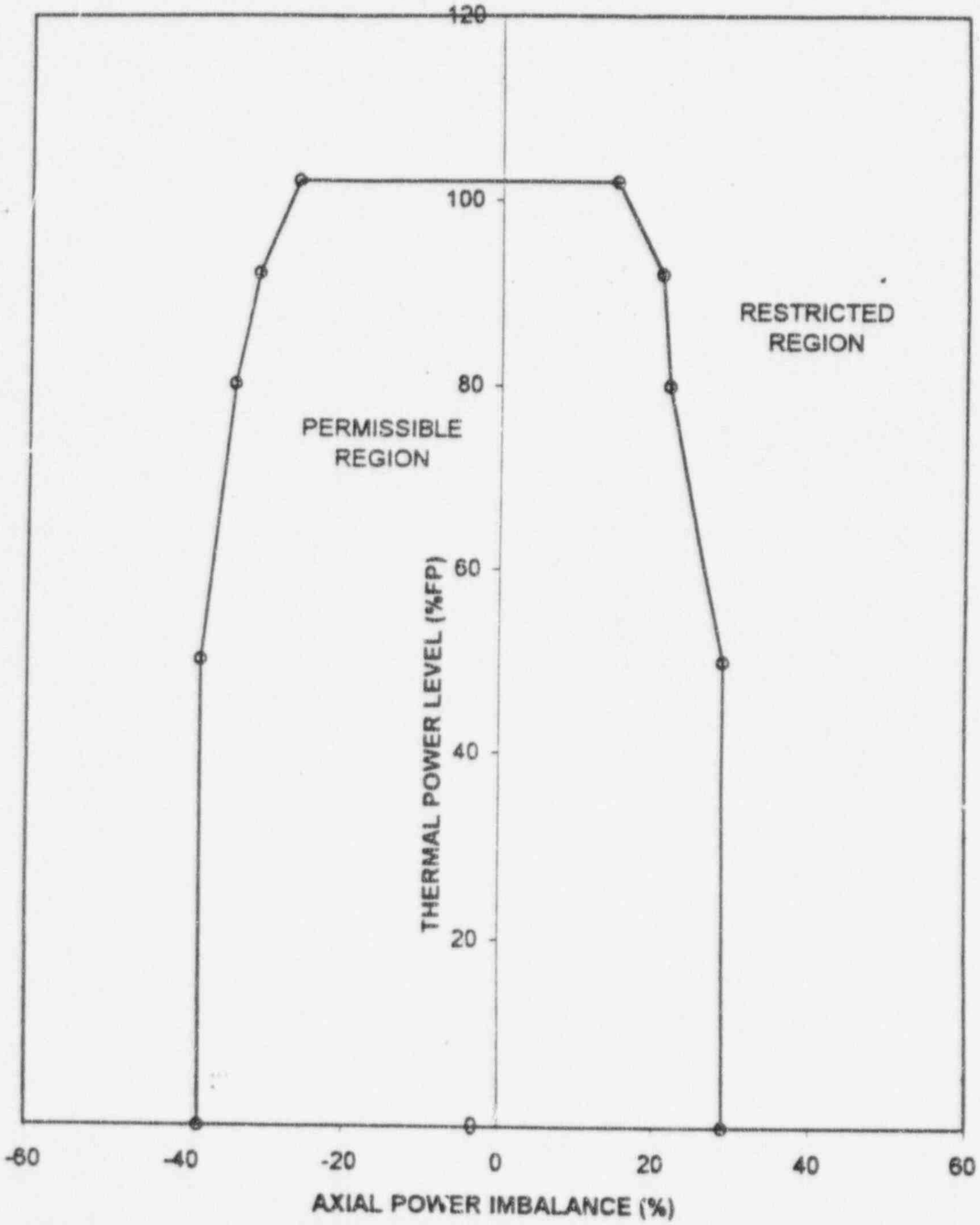


Figure 5.9 Typical Control Rod Insertion Limits

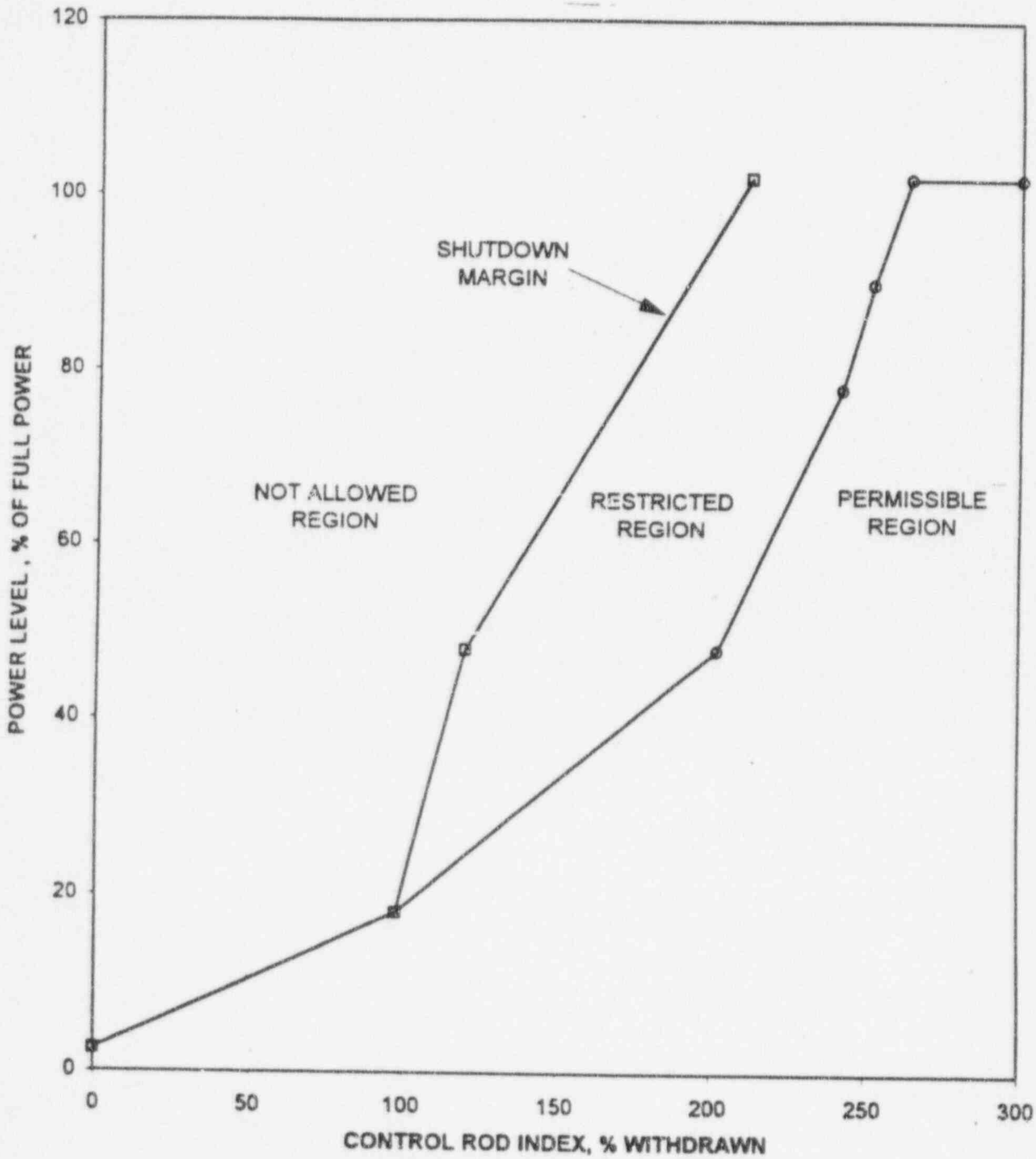


Figure 5.10 LOCA Margin Contour Curve

Figure 5.11 Core Offset Contour Curve

Figure 5.12 Determination of Allowable Inserted Control Rod Worth
Based on Ejected Control Rod Worth Limit

Figure 5.13 Integral Control Rod Worth Curve and Determination of CRI Limits

6.0 ACCIDENT ANALYSIS REVIEW

6.1 Overview

The reference safety analyses in Chapter 14 of the TMI-1 FSAR (Reference 1) are performed by employing conservative plant parameters and are supposed to be bounding all reload cycles. The accident analyses contained in the licensing basis safety analyses remain valid if a reload design predicts steady-state and transient parameters including RPS trip setpoints that lie within the ranges of the values assumed in the reference safety analyses. Therefore, it is important to verify that the predicted parameter values of a reload fuel cycle lie within the current licensing base of the plant. Should there be a change in the RPS trip setpoints due to either reload fuel or a plant modification, the impact of this change to the accident analyses must be evaluated to see the safety margin in the reference analyses is preserved.

The key parameters that have the greatest effect on the outcome of either a transient or an accident can typically be classified into three major areas: fuel thermal parameters, thermal-hydraulic parameters, kinetic parameters, including the reactivity feedback coefficients and control rod worth. The fuel thermal parameters consist of fuel design characteristics including the minimum LHR to cause fuel melt. The core thermal-hydraulic design parameters are given in Table 3.1. The key kinetic parameters are given in Table 6.1. These include Doppler and moderator temperature coefficients, ejected and dropped control rod worths, and boron worth. These parameters directly affect the core power response if an accident occurs during the fuel cycle. Those parameters used in the reference safety analyses are compared with the typical values for a reload core design.

In general, as can be seen in Table 6.1, both Doppler and moderator temperature coefficients become more negative as the fuel cycle operation proceeds from BOC to EOC. The moderator coefficient becomes more negative as the soluble boron concentration in the RCS is reduced at EOC and Doppler coefficient become more negative because of plutonium buildup. In the analyses of events that cause reactor coolant (RC) heatup, the use of the BOC coefficients would result in more severe consequences, while the same is true for the use of the EOC coefficients in the events resulting in a RC temperature decrease.

The RC heatup events where the BOC Doppler and moderator coefficients are used in the analysis are: (1) moderator dilution event, (2) control rod withdrawal accident, (3) startup event, (4) control rod ejection accident, (5) loss of coolant flow, (6) loss of main feedwater, (7) uncompensated operating reactivity change, and (8) loss of electric power. The RC temperature decreasing accidents include: (1) steam line break, (2) cold water event, and (3) dropped or stuck control rod. The relationship of these accidents with the key parameters as well as the corresponding RPS trip functions are given in Table 6.2.

6.2 Discussion of Individual Accidents

6.2.1 Moderator Dilution Accident

During normal operation, the boron concentration is reduced through the makeup system by gradually injecting fluid with a lower boron concentration than that in the RCS. A moderator dilution accident occurs when the process continues for a long period of time at excessive makeup flow rates. The positive reactivity insertion caused by the decrease

in boron concentration would cause an increase in reactor power, which, in turn, leads to increased RC and fuel temperatures.

The reactor protection criteria for this accident are: (1) reactor thermal power remains less than 112 %FP, (2) RCS pressure remains below 110% of the design pressure (1.10 x 2500 = 2750 psig), and (3) the preservation of 1 %Δk/k shutdown margin.

The key parameters in this accident analysis are the dilution flow rate (maximum 500 gpm), initial power of 100 %FP, initial boron concentration, inverse boron worth, and BOC values of Doppler and moderator coefficients. The 500 gpm dilution flow rate (nominal value = 45 to 70 gpm) is chosen to bound all operating power conditions. This flow rate can only be achieved at an RCS pressure less than 1000 psi. The initial boron concentration in combination with the inverse boron worth determine the positive reactivity insertion rate due to boron dilution. A larger initial boron concentration causes a higher reactivity held by the boron, while the higher inverse worth results in a lower reactivity insertion rate. The accident is terminated by reactor trip on either high pressure or high temperature.

6.2.2 Rod Withdrawal Accident

An accidental and uncontrolled control rod withdrawal results in a positive reactivity addition to the core. As a result, the power level increases, the RC and fuel temperatures increase. The acceptance criteria for this accident are that the following parameters are not to exceed the following: reactor power less than 112 %FP, RC temperature less than the limit (620 °F), and high RC pressure less than the safety limit (2750 psig).

The key parameters for this accident are initial power level, BOC Doppler and moderator coefficients, total control rod worth, and control rod withdrawal rate. The use of the least negative Doppler and moderator coefficients causes more severe results. The total control rod worth defines a maximum control rod worth that provides the largest positive reactivity insertion. The reactivity insertion rates assumed in the reference safety analysis (Reference 1) range from a minimum of 1.09×10^{-4} to a maximum of 7.25×10^{-4} %Δk/k/sec. The minimum rate represents the single control rod group withdrawn at maximum speed, while the maximum insertion rate is based on all control rod groups withdrawn at the maximum speed. The accident is terminated by either an overpower trip or high pressure trip depending upon the reactivity insertion rate.

6.2.3 Startup Accident

The uncontrolled reactivity insertion during reactor startup at zero power is a startup accident. This accident is terminated by the negative Doppler effect in the absence of all other protective actions.

The criteria for this accident analysis is that the RPS shall be designed to limit (1) the reactor thermal power to less than the design overpower condition (112 %FP), and (2) the RCS pressure shall not exceed the safety limit of 2750 psig. The key parameters for this accident and the RPS trip functions are same as those for the control rod withdrawal accident and are given in Table 6.2.

6.2.4 Uncompensated Operating Reactivity Change

During normal operation of the reactor, the overall reactivity of the core changes because of fuel depletion and changes in the fission product poison concentration (mainly xenon buildup). These reactivity changes, if left uncompensated, can cause LCO limits to be exceeded. Depending upon the direction of the reactivity change, the reactor power increases or decreases, which, in turn, causes an increase or decrease of the RCS average temperature.

The acceptance criteria used in evaluating this event are that (1) the reactivity insertion rate is less than the rate at which the operator can compensate, and (2) the rate of the RCS temperature change is less than the rate at which the automatic control system can compensate for the change. The key parameters for this event are the BOC Doppler and moderator coefficients and the rate of the reactivity change. The reactivity changes in this event are extremely slow and allow the operator to detect and compensate for the changes.

6.2.5 Rod Ejection Accident

A control rod assembly ejection is the rapid ejection of a single control rod assembly from the core region during plant operation. The ejection is due to a pressure differential between system pressure and atmospheric pressure which acts on the control rod following a breach of the RCS pressure boundary in the control rod drive housing. As the control rod is ejected, positive reactivity is added to the core. The actual amount of

positive reactivity inserted into the core depends on the ejected control rod worth. The neutron power rises extremely rapidly. The speed of the neutron power rise does not allow heat transfer out of the fuel to the coolant. This results in the process being nearly adiabatic. The large power rise heats the fuel and the Doppler feedback causes a power reduction prior to the reactor trip. As the energy in the fuel is transferred to the coolant, the RCS pressure increases. The rapid increase in power can lead to fuel failures and RCS pressure increases to challenge the pressure boundary.

The acceptance criteria for this accident are that fuel enthalpy shall remain below the 280 cal/gm threshold point at which the beginning of fuel fragmentation occurs (Reference 25) and that the RCS pressure shall remain less than the design pressure. The percent of the core experiencing DNB is also a concern.

The key parameters are the BOC Doppler and moderator coefficients, 0.65 %Δk/k ejected control rod worth at the rated power (also 1.0 %Δk/k at zero power), and control rod ejection time [ ]. The accident analyses are also performed based on the delayed neutron fraction values at both BOC and EOC to see the effect on the rate of power increase.

The reactivity transient resulting from this accident is limited by the Doppler feedback and terminated by the RPS trip with no serious core damage. The accident can be terminated by either the high pressure trip or the high flux trip. For example, if the transient occurs at zero power and the ejected control rod worth is small, the plant will trip on high pressure. If the transient occurs at full power and the ejected rod worth is large, the plant will trip on high flux.

6.2.6 Loss of Coolant Flow

A loss of coolant flow (LOCF) accident is the reduction in or loss of forced flow through the RCS. The LOCF accident could be either due to pump coastdown(s) (1 to 4 pump trips) or locked pump rotor. The locked rotor event occurs when one of the RC pump rotors seizes and no longer provides forced reactor coolant flow, which causes a rapid RC flow reduction to 75% of design flow. With the reactor at power, the LOCF results in an increase in the RCS temperature and, therefore, a decrease in heat transfer. The key parameters are BOC Doppler and moderator coefficients, and the flow reduction rate during the given transient. The BOC parameters are chosen to maximize the reactor power response to the RC temperature increase.

The safety criteria for these events are that the minimum DNBR must be greater than (1) the design limit of 1.18 for pump coastdowns and (2) 1.0 for the locked rotor accident. Reactor protection for these events is provided by either a flux/flow trip or a power/pump status monitor trip, which are described in Sections 3.5.1 and 5.4.1.

6.2.7 Loss of Electric Power

Loss of electric power accident addresses two separate events:

(1) loss of electric load (LOEL), and
(2) complete loss of all station power (station blackout).

The acceptance criteria for these events for reactor protection are: (1)fuel damage will not occur from an excessive flux-to-flow ratio, and (2) RCS pressure shall not exceed 110 percent of design pressure.

The LOEL event is initiated by faults within the turbine generator or an open switchyard circuit breaker. Thus, the LOEL may be caused by separation of the unit from the transmission system. During the LOEL event, the integrated control system (ICS) initiates a power reduction to 15% FP and controls main feedwater flow, secondary pressure and makeup flow. As a result of this control, reactor trip is avoided. Without ICS action, reactor will be tripped on high coolant pressure or temperature. The analyses of this event assumes the BOC parameters to maximize the power response to the coolant temperature and pressure increase.

The station blackout event is the hypothetical case where all unit power, except the unit batteries, is lost. The loss of station power results in gravity insertion of the control rods and trip of the turbine valves. The RC pump, main feedwater, and condensate booster pump trips follow. After the turbine stop valves trip, excessive temperature and pressure in the RCS are prevented by natural circulation with steam relief through the main steam line safety valves and the atmospheric dump valves. The station blackout does not result in any fuel damage or excessive pressure on the RCS.

The analyses use the BOC parameters to maximize the power response to the coolant temperature increase. However, the consequence of the station blackout is practically independent of reload design parameters.

6.2.8 Steam Line Break

The steam line break (SLB) accident assumes a break in the secondary system pressure boundary that results in a decrease in secondary system pressure. Under this condition, the rapidly decreasing secondary pressure results in an excessive primary system cooldown which, under the influence of a negative moderator temperature coefficient, produces a positive reactivity insertion. A reactor trip occurs due to low RC pressure or high neutron flux.

If feedwater flow continues to the affected steam generator, the excessive heat removal and concurrent RC cooldown will continue. Furthermore, the reactor may experience a recriticality even after the trip if the positive reactivity from the cooldown exceeds the shutdown margin.

EOC parameters are used in the SLB accident to maximize the positive reactivity insertion. The reference analysis (Reference 16) demonstrates that adequate margin is available throughout the accident (before and after the reactor trip) when the reactivity feedback during primary system cooldown below 532°F is limited to a reactivity temperature deficit of 0.6% $\Delta k/k$ (Table 6.1).

6.2.9 Cold Water Accident

This accident cannot occur in the TMI-1 reactor at full power operation because the RCS piping contains no check valves or isolation valves, thus eliminating the cause of the cold

water accident. However, the cold water accident is possible when the reactor is in the partial pump operation mode (e.g., two pump operation at 50% power). The remaining pumps not running can be started abruptly and thus the core average temperature would decrease. In the presence of negative reactivity coefficients (both Doppler and moderator), a reduction in the coolant and fuel temperature would yield a positive reactivity insertion and thus increase the power level.

The power increase response to this type of accident is inherently self-limiting due to the compensating reactivity feedback effects. Furthermore, the RPS provides a high neutron flux trip function.

The protection criteria for this accident are that the minimum DNBR be greater than 1.18 and the RCS pressure limits are not exceeded. The reference analysis is based on EOC parameters to promote positive reactivity insertion with more negative Doppler and moderator temperature coefficients.

6.2.10 Stuck-Out, Stuck-In, or Dropped-In Control Rod Accident

A stuck-out, stuck-in or dropped control rod are the three types of control rod misalignment events which can cause a decrease in shutdown margin, an increase in local power peaking, and a positive reactivity insertion.

The stuck-out accident is that, during a reactor trip, a control rod misalignment causes a control rod stuck-out at the fully withdrawn position. This condition requires an evaluation

to determine that sufficient negative reactivity is available for tripping the reactor with a maximum worth control rod stuck-out. The effect of this accident is mitigated by requiring a shutdown margin of 1 %Δk/k, with the control rod of maximum worth fully withdrawn from the core. The shutdown margin analysis is performed in every reload design and is incorporated into the control rod position limits as described in Section 5.5.2 (Figure 5.9).

A stuck-in control rod accident occurs during withdrawal of the control rods where one control rod assembly becomes stuck at some position as the other control rods continue in motion. This condition causes neutron flux distortions that could result in localized power densities and heat fluxes in excess of the design limits if the reactor is allowed to return to full power without corrective action by the reactor operator. This accident is less limiting because, if a control rod becomes stuck, the operator is informed by several alarms and has time to take corrective action. Nevertheless, a stuck-in control rod assembly implies a dropped control rod condition which is discussed below.

A dropped control rod assembly is defined as a deviation of a control rod from the average group position by more than an indicated five inches. This definition then covers both a stuck-in and a dropped control rod assembly. If a control rod is dropped while operating, a rapid decrease in neutron power would occur, accompanied by a decrease in core average coolant temperature. In addition, the power distribution might be distorted (both axially and radially) due to the new control rod pattern. Furthermore, the decrease in the RC temperature would result in a positive reactivity insertion due to the negative moderator and Doppler coefficients.

The protection criterion for this accident is that the minimum DNBR is greater than the design limit of 1.18. The EOC parameters are used in the accident analysis to maximize the positive reactivity insertion. The reference analysis demonstrates that the highest power levels resulting from the dropped control rod accident are well below the overpower condition.

6.3 Radiation Dose Evaluation

Radiation dose analyses for the FSAR events (Reference 1) assess the risk to the public accounting for the performance of each of the fission product barriers including meteorological conditions. The predicted radiation doses must be less than the limits stipulated in 10CFR100 (Reference 9).

In general, the radiation source term data are not reload dependent. And, as shown in the FSAR (Reference 1), the source term used in the reference analysis is very conservative and the predicted doses are well below the allowed limits. However, the source term could be slightly changed from cycle to cycle depending upon the length of fuel cycle and fuel enrichment. For this reason the radiation source term is calculated in each reload design by using the ORIGEN2 code (Reference 26) and is compared with the reference analysis.

Table 6.1
Key Safety Parameters for Accident Analysis Review

PARAMETERS	REF.ANALYSIS VALUES	CONSERVATIVE DIRECTION	TYPICAL RELOAD CYCLE VALUES
Doppler Coeff. BOC, $\Delta k/k/°F$	-1.17×10^{-5}	more negative	-1.55×10^{-5}
Doppler Coeff. EOC, $\Delta k/k/°F$	-1.33×10^{-5}	(1)	-1.74×10^{-5}
Moderator Coeff. BOC, $\Delta k/k/°F$	$+0.90 \times 10^{-4}$	less positive	-0.28×10^{-4}
Moderator Coeff. EOC, $\Delta k/k/°F$	-4.00×10^{-4}	less negative	-3.29×10^{-4}
Total Rod Worth % $\Delta k/k$	12.9	smaller	8.07
Maximum Ejected Rod Worth, % $\Delta k/k$	0.65	smaller	0.25
Maximum Dropped Rod Worth, % $\Delta k/k$	0.46	smaller	$\leq$ 0.20
Inverse Boron Worth, ppm/(% $\Delta k/k$)	75[2]	larger	167
Init. Boron Conc. BOC, HFP, ppm	1200[2]	smaller	2200
SLB Temperature Deficit, EOC, %$\Delta k/k$	0.6	smaller	0.25

NOTES:

(1) = Safety margin increases when the value is more negative for the steam line break accident, while the margin increases with a less negative value for the dropped control rod accident.

(2) = The initial boron concentration in combination with the inverse boron worth determines the positive reactivity insertion rate during the moderator dilution accident.

Table 6.2
Safety Parameter Matrix Against Affected Safety Analyses

ACCIDENT DESCRIPTIONS	KEY SAFETY PARAMETERS									
	BOC DOP COEF	EOC DOP COEF	BOC MOD TEMP COEF	EOC MOD TEMP COEF	EJECTED ROD WORTH	DROPPED ROD WORTH	BORON WORTH	INIT BORON CONC	TOTAL ROD WORTH	REACTOR TRIP
MODERATOR DILUTION EVENT	X		X				X	X		HT or HP
ROD WITHDRAWAL ACCIDENT	X		X						X	HF or HP
STARTUP EVENT	X		X						X	HF or HP
ROD EJECTION ACCIDENT	X		X		X					HF or HP
LOSS OF COOLANT FLOW	X		X							FF or PSM
LOSS OF ELECTRIC POWER	X		X							HT or HP
STEAM LINE BREAK		X		X						LT or HF
COLD WATER EVENT		X		X						HF
DROPPED OR STUCK ROD		X		X		X				NA
UNCOMP. REACT. CHANGE	X		X							NA

NOTES:

HT = High Temperature Trip
HP = High Pressure Trip
FF = Flux/Flow Trip
NA = Transients terminated without reactor trip.

LT = Low Temperature Trip
HF = High Flux Trip
PSM = Pump Status Monitor Trip

7.0 CONCLUSION

This report presents GPU Nuclear's (GPUNs) reload design methodology and corresponding safety criteria for the TMI-1 reactor. The methodology covers all the important aspects of the reload design such as nuclear design, thermal-hydraulic design, maneuvering analysis, setpoint methodology and accident analysis review. The setpoint methodology included in this report consists of RPS trip setpoints and LCO alarm limits for normal operating control.

Computer codes to be employed for the GPUN reload analyses (CASMO-3/SIMULATE-3, RETRAN-02, and VIPRE-01) have been qualified and topical reports on these computer programs were previously submitted to the NRC (References 5, 6, and 7). The CASMO-3/SIMULATE-3 codes (Reference 5) will be applied for all the nuclear-design related analyses including fuel cycle design, generation of the reload specific safety parameters, maneuvering analyses, and setpoint analyses. RETRAN-02 (Reference 6) will be employed for the system transient analyses including FSAR events and setpoint analyses. VIPRE-01 (Reference 7) will be used for core thermal-hydraulic analyses including MAP limit analyses and setpoint analyses.

Also included in this report are results from a demonstration analysis using the setpoint methodology discussed herein. In Appendix A the GPUN-predicted setpoints are compared with those of the fuel vendor for TMI-1 Cycle 10. The good agreement between the two results adds further confidence in the GPUN reload methodology.

In conclusion, the reload design methodology presented herein demonstrates GPUN's capability and competence to perform reload design analyses for TMI-1 cores.

8.0 REFERENCES

1. TMI-1 Final Safety Analysis Report (Update Version), March 1994.

2. NRC Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," October 4, 1988.

3. NRC Letter From A. C. Thadani (NRC) to J. D. McCarthy (BWOG Core Performance Committee), "Acceptance for Referencing of Licensing Topical Report BAW-10179P, 'Safety Criteria and Methodology for Acceptable Cycle Reload Analyses' (TAC No. M80189)," March 16, 1993.

4. BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses," August 1993.

5. GPUN TR-091, Rev. 0, "Steady State Reactor Physics Methodology for TMI-1," January 23, 1995.

6. GPUN TR-078, Rev. 0, "TMI-1 Transient Analyses Using the RETRAN Computer Code," January 6, 1995.

7. GPUN TR-087, Rev. 0, "TMI-1 Core Thermal-Hydraulic Methodology Using the VIPRE-01 Computer Code," March 10, 1995.

8. U.S. Nuclear Regulatory Commission, "Code of Federal Regulations - Title 10; Energy."

9. Title 10 Code of Federal Regulations (10CFR), Part 100, Reactor Site Criteria.

10. Title 10 Code of Federal Regulations (10CFR), Part 50, Licensing of Production and Utilization Facilities:

    a. Part 50.36, Technical Specifications.
    b. Part 50.46, Acceptance Criteria for ECCSs for LWRs.
    c. Appendix A, General Criterion 10, Reactor Design.
    d. Appendix A, General Criterion 12, Suppression of Reactor Power Oscillations.
    e. Appendix A, General Criterion 26, Reactivity Control System Redundancy and Capability.
    f. Appendix A, General Criterion 28, Reactivity Limits.

11. M. Edenius and B. H. Forssen, "CASMO-3, A Fuel Assembly Burnup Program V4.7," Studsvik/NFA-89/03, Rev. 3, June 1993, Studsvik Energitenik.

12. J. A. Umbarger, "TABLES-3, Library Preparation Code for SIMULATE-3," Studsvik/SOA-92/03, Rev. 0, April 1992.

13. K. S. Smith, J. A. Umbarger and A. S. DiGiovine, "SIMULATE-3, Advanced Three-Dimensional Two-Group Reactor Analysis Code V4.02," Studsvik/SOA-92/01, Rev. 0, April 1992.

14. GPUN TDR-930, Rev. 0, "Qualification of BWC CHF Correlation on VIPRE-01 Code," August 3, 1988.

15. BAW-10143P-A, "BWC Correlation of Critical Heat Flux," B&W, April 1985.

16. BAW-2250, Rev. 1, "TMI-1 Cycle 11 Reload Report," September 1995.

17. BAW-10147PA-R1, "Fuel Rod Bowing in B&W Fuel Design," B&W, May 1983.

18. BAW-1829, "Thermal-Hydraulic Cross Flow Application," B&W, April 1984.

19. TMI-1 Facility Operating License No. DPR-50, Appendix A, "Technical Specifications."

20. NRC R.G. 1.49, "Power Levels of Nuclear Plants," December 1973

21. BAW-10110, "CHATA: Core Hydraulics and Thermal Analysis," January 1976.

22. BAW-1397, "Crystal River Unit 3 Fuel Densification Report," August 1973.

23. B&W Document No. 86-117640-00, "Detector Lifetime Extension Final Report for TMI-1," September 1988.

24. BAW-10122A, Rev. 1 "Normal Operating Controls," May 1984.

25. ANL-7225, "Studies in TREAT of Zircaloy-2 Clad UO_2-Core Simulated Fuel Elements," January-June 1966.

26. ORNL-CCC-371, "ORIGEN2; Isotopic Generation and Depletion Code - Matrix Exponential Method," December 1985.

APPENDIX

Setpoint Methodology Demonstration Analyses

A.1 Introduction

Demonstration analyses were performed to generate key setpoints for TMI-1 Cycle 10, applying the methodology presented in this topical report and References A.1 through A.3, and compare them to those in References A.4 and A.5. Figures A.1 through A.9 and Table A.1 show these comparisons. As can be seen in these figures and Table A-1, all the GPUN setpoints and limits agree very well with those from the vendor's analyses.

A.2 Discussion of Setpoint Results

[ ]

A.3 References

A.1 GPUN TR-091, Rev. 0, "Steady State Reactor Physics Methodology for TMI-1," January 23, 1995.

A.2 GPUN TR-078, Rev. 0, "TMI-1 Transient Analyses Using the RETRAN Computer Code," January 6, 1995.

A.3 GPUN TR-087, Rev. 0, "TMI-1 Core Thermal-Hydraulic Methodology Using the VIPRE-01 Computer Code," March 10, 1995.

A.4 BAW-2187, "TMI-1 Cycle 10 Reload Report," May 1993.

A.5 GPUN TR-099, Rev. 2, "TMI-1 Cycle 10 Core Operating Limits Report," June 1995.

A.6 TMI-1 Facility Operating License No. DPR-50, Appendix A, "Technical Specifications."