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Licensing Topical Report
Generic Evaluations of GE Boiling Water Reactor Power Uprate (Volume 1)

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LICENSING TOPICAL REPORT

GENERIC EVALUATIONS OF GENERAL ELECTRIC BOILING WATER REACTOR POWER UPRATE (VOLUME I)

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DISCLAIMER OF RESPONSIBILITY

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ABBREVIATIONS

ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
BOC	Beginning of Cycle
BOP	Balance of Plant (beyond NSSS)
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owners' Group
CCFL	Counter Current Flow Limiting
CFR	Code of Federal Regulations
ECCS	Emergency Core Cooling System
ELLLA	Extended Load Line Limit Analysis
EOC	End of Cycle
EOP	Emergency Operating Procedures
EPG	Emergency Procedure Guidelines
GE	General Electric Company
GESTAR	GE Standard Application for Reactor Fuel
gpm	Gallons Per Minute
HCTL	Heat Capacity Temperature Limit
HPCI	High Pressure Coolant Injection
HPCS	High Pressure Core Spray
IGSCC	Intergranular Stress Corrosion Cracking
I&E	Inspection and Enforcement
LOCA	Loss-of-Coolant Accident
LPCI	Low Pressure Coolant Injection (function of RHR)
L2	Reactor Water Level Two
L3	Reactor Water Level Three
LTR	Licensing Topical Report
LTR1	Reference 1-1
LTR2	This document
MAPLHGR	Maximum Average Planar Linear Heat Generation Rate

ABBREVIATIONS
(Continued)

MCFI	Minimum Core Flooding Interval
MCPR	Minimum Critical Power Ratio
MEOD	Maximum Extended Operating Domain
MSIV	Main Steam Isolation Valve
NEDC	GE Nuclear Energy Division Customer Report
NEDM	GE Nuclear Energy Division Memorandum Report
NEDO	GE Nuclear Energy Division Open Distribution Report
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OLCMPR	Operating Limit Minimum Critical Power Ratio
PCT	Peak Cladding Temperature
PLHGR	Peak Linear Heat Generation Rate
psi	pounds per square inch
psia	pounds per square inch absolute
psid	pounds per square inch difference
psig	pounds per square inch gauge
RCIC	Reactor Core Isolation Cooling
REDY	Reactor Dynamics Code
REM	Roentgen Equivalent Man
RHR	Residual Heat Removal
RPV	Reactor Pressure Vessel
SAFE	Long-Term Thermal Hydraulic Transient and Accident Model
SAFER/GESTR	Long-Term Inventory Model for BWR Loss-of-Coolant Analysis
SIL	GE Services Information Letter
SLMCPR	Safety Limit Minimum Critical Power Ratio
SRV	Safety/Relief Valve
TMI	Three Mile Island

ABSTRACT

This document presents specific evaluations of areas of licensing review that are generically applicable to the uprating of all or groups of BWR plants. It is intended to be used in conjunction with a plant-specific licensing evaluation to be submitted by the utility. This set of reports provides information sufficient for assessment of the acceptability of the plant to be licensed at the uprated power level. Four main groups of information are presented: (1) licensing evaluations of NRC and industry generic communications applicable to power uprate, and generic assessment of licensing topics; (2) analytical evaluations that can be approved generically; (3) bounding evaluations of components and equipment; and (4) a discussion of the impact of power uprate on safety margin. Approval of these generic areas by the NRC will provide significant benefits to the plant-specific submittal review process.

1.0 INTRODUCTION

Most GE BWR plants have been designed with an as-built equipment and system capability to produce a steam flow rate which is at least 5% above the originally licensed operating conditions. Generic guidelines for justifying operation at 5% increased steam flow have been documented in Reference 1-1.

This document presents specific evaluations of areas of licensing review that are generically applicable to the uprating of all or groups of BWR plants. This document focuses on BWR/4, 5 & 6 plants. It is intended to be used in conjunction with a plant-specific licensing evaluation (to be submitted by the utility) to provide all information necessary for assessment of the acceptability of the plant to be licensed at the uprated power level. Information that is not included in this report must be provided separately by the utility.

The primary focus of this document is on areas of review that have been treated generically in the past. Four main groups of information are presented: (1) licensing evaluations of NRC and industry generic communications applicable to power uprate, and generic assessment of licensing topics; (2) analytical evaluations that can be approved generically; (3) bounding evaluations of components and equipment; and (4) a discussion of the impact of power uprate on safety margins.

2.0 LICENSING EVALUATIONS

2.1 GENERIC COMMUNICATIONS

Implementation of power uprate requires a review and update of the plant licensing basis as described in the Updated Safety Analysis Report (USAR) and other docketed plant documentation. An exhaustive review, therefore, encompasses documents issued by the USNRC, INPO and vendors (generic communications) and licensing commitments made by individual applicants to ensure compliance with the plant's original licensing basis and for compliance with applicable new regulatory requirements and operating experience in the nuclear industry.

Generic communications were identified as applicable to GE BWRs prior to being reviewed for power uprate compliance. NRC communications issued as applicable to PWRs Only, Research Reactors Only, Not Related to Power Plants, For Comment Only or Withdrawn were excluded, as were Dropped generic issues. The generic communications reviewed and evaluated for compliance consist of (1) NRC-issued Regulations, TMI items, Generic Safety Issues and Action Items, Regulatory Guides, Generic Letters, I&E Bulletins, I&E Notices, I&E Circulars; and (2) industry-issued INPO Significant Operating Experience Reports, GE Services Information Letters, and GE Rapid Information Communication Services Information Letters.

2.1.1 Typical Plant Parameter Changes

The following plant parameter changes associated with power uprate were considered salient to the compliance review.

- o Increased plant thermal power and electrical output.
- o Increased primary and secondary system heat loads.
- o Increased plant feedwater and steam flows or core flow.
- o Increased process pressure(s), pressure drop(s), and temperatures.

- o Increased flow-induced primary and secondary system vibration or wear and corrosion.
- o Potentially decreased fuel thermal margin.
- o Increased equilibrium core radiological source terms.
- o Increased DBA hydrogen and oxygen generation.
- o Increased RPV operating fluence.
- o Increased equilibrium normal coolant radiological source terms.
- o Increased process stream equilibrium corrosion products, impurities (soluble and insoluble), activation or fission products (solid and gaseous) and entrained noncondensible gases.

2.1.2 Compliance Evaluation

Each applicable document was evaluated, in its context, to determine if its requirements, reporting requests, or operating information could affect power uprate compliance [yes or no]. The impact assignment considered (1) the screening criteria identified in Paragraph 2.1.2.1 in conjunction with the above parameter changes, (2) the generic criteria, methodology, and scope in Reference 1-1, and/or (3) the analytical and hardware evaluations in Sections 3 and 4 of this report.

2.1.2.1 Screening Criteria

Generic communications were judged to have an impact on power uprate when, for the typical power uprate parameter changes identified in Subsection 2.1.1, one or more of the following conditions apply:

- o Equipment (component) performance or qualification requirements are potentially modified.
- o System or equipment performance capabilities are significantly challenged or reduced.
- o System processing capability is potentially reduced.
- o Plant or fuel core operating limits are potentially modified.
- o Plant accident or transient analyses are potentially modified.
- o Plant procedural changes are potentially required.

If none of the above conditions applied, the generic communication was judged to have no impact on power uprate.

2.1.2.2 Results

Generic communications which could impact power uprate have been considered. Each was carefully reviewed and judged as either (1) not affected by power uprate, (2) affected and dispositioned generically, or (3) affected and dispositioned on a plant-specific basis.

2.1.2.3 Individual Plant Evaluations

Plant-specific submittals will address all exceptions to the generic communication dispositions justified in Section 2.1.

Applicability of 2.1: BWR/3, 4, 5 & 6

2.2 SETPOINT METHODOLOGY

The setpoint methodology developed by GE performs setpoint calculations to comply with Regulatory Guide 1.105. The methods are not dependent on rated power level, pressure or other potential plant parameter changes due to power uprate. That is, the methods, and therefore the application of the methods, are not impacted by power uprate. It is expected that the analytic limits for some setpoints (e.g., high pressure scram) will change due to uprate, but the revised nominal and allowable setpoint values will be determined for the new plant conditions using methods acceptable to the NRC.

Applicability of 2.2: All BWRs utilizing GE setpoint methodology.

2.3 EMERGENCY OPERATING PROCEDURES

Emergency Operating Procedures (EOPs) include variables and limit curves which define conditions where operator actions are required. Some of these variables and limit curves depend upon the plant value of rated reactor power. The operator actions in the EOPs will not

change as a result of increasing rated reactor power; only the conditions at which some of the actions are specified will change. Changing some of the variables and limit curves will require modifying the values in the EOPs and updating utility support documentation. EOP curves and limits may also be included in the safety parameter display system. It must also be updated accordingly. This section discusses which EOP values will be affected by increasing rated reactor power.

Plant EOPs are based on generic Emergency Procedure Guidelines (EPGs) developed by the BWR Owners' Group (Reference 2-1). The EPGs contain various curves and limits that must be determined on a plant-unique basis when EOPs are implemented from the EPGs.

Appendix A to the EPGs (Reference 2-2) provides the technical basis for each curve and limit that must be determined for plant-specific EOPs where the basis is not obvious from its use and definition in the EPGs. Appendix A also provides a list of plant-specific data which is required to calculate each variable or curve, including if it is a function of reactor power.

Appendix C to the EPGs (Reference 2-3) provides an acceptable method by which the variables or curves may be determined on a plant-specific basis. Appendix C to the EPGs is organized in "Worksheets", with some worksheets calculating more than one variable or curve. The equations provided in Appendix C were examined to determine how each variable or curve is affected by the plant value of rated reactor power.

In addition to a specific dependence on rated reactor power, some variables or curves depend upon the result of other worksheets (this is also defined in Reference 2-3). Therefore, where rated reactor power affects a top level worksheet, the other worksheets that depend upon the output of that top level worksheet may also be affected even if rated reactor power is not a direct input to the dependent worksheet.

Finally, the scope of recalculation required is dependent upon the magnitude of the plant changes associated with the power uprate. The recalculations required can be considered by grouping them in the following categories:

I. Change rated reactor power only

II. Change lowest safety/relief valve (SRV) lift pressure setpoint in addition to rated reactor power.

III. Change containment operating temperatures in addition to rated reactor power.

IV. Change fuel type in addition to rated reactor power, but the new fuel has the same peak linear heat generation rate (PLHGR) and the same fuel rod dimensions.

V. Change fuel type in addition to rated reactor power, and the new fuel has a different PLHGR and/or fuel rod dimensions.

These categories encompass all the expected changes associated with power uprate that affect the EOP variables and curves. For example, if the power uprate causes both the lowest SRV lift pressure setpoint to change and has a new fuel type loaded, then both categories II and IV (or V) need to be examined. However, when a plant-specific uprate program is defined, the affected plant values will be verified against the plant data required for EOP calculations to ensure that no other values are affected.

Applicability of 2.3: All BWRs

3.0 ANALYTICAL EVALUATIONS

The following sections provide generic evaluations of aspects of power uprate that involve analytical investigation. In some areas, cases are presented which bound specific plant sizes and/or BWR product lines. In others, a generic review of the impact of power uprate is provided to show that performance of all applicable plants remains within acceptance criteria or current licensing practice. In this way, generic review will significantly help the review of the individual lead projects and subsequent applicants.

3.1 LOSS OF FEEDWATER FLOW TRANSIENT

This section documents the generic basis and results for evaluation of the Loss of Feedwater Flow transient event. This case is the original design basis for the performance of the Reactor Core Isolation Cooling (RCIC) System on all BWR/4, 5 & 6 plants. The RCIC System is designed to maintain adequate water level in the reactor during a Loss of Normal Feedwater transient even with single failure of the other high pressure water supply system [High Pressure Coolant Injection (HPCI) on BWR/4s, and High Pressure Core Spray (HPCS) on BWR/5 & 6 plants]. In principle, the other high pressure system must also meet the same performance requirements with the assumed single failure of the RCIC System, but that is always less limiting, since RCIC is the smaller of the two systems on all plants.

Two criteria are applied to this event:

(1) Limiting Criterion: The RCIC System (the smaller of the two high pressure coolant supply systems) shall maintain sufficient water level inside the core shroud to assure that the top of the active fuel remains covered throughout the event.

(2) Operational Criterion: The RCIC System shall maintain wide range sensed reactor water level high enough that the very low level instrument trip setpoint (Level 1) for low pressure emergency core cooling system initiation and main steam isolation valve (MSIV) closure (if applicable) is not activated. This operational aspect is evaluated for the Level 1 setpoint for each plant but is not generically documented here.

The following sections provide the bounding results for BWR/4, 5 & 6 product line reactors. Table 3-1 lists the primary parameters affecting the analysis of the BWR/4, 5 & 6 plants. Table 3-2 shows the previous results for various product line plants and the results calculated for uprated plants. The uprated results also include changes in some of the parameters to be compatible with current or planned changes to plant technical specifications.

3.1.1 Evaluation of Plants

Analyses of representative plants were performed with the input parameters listed in Table 3-1. The excess design capability decreases very slightly with power uprate, but the analysis results easily meet the limiting criteria. Analysis at 102% of uprated power has shown a reduction in the lowest water level (slightly more severe water level transient), but a large amount of water (more than 5 ft) remains above the top of the active fuel as shown in Table 3-2. The sensed water level outside the shroud has also been checked and shows that abundant operational flexibility still exists with uprate.

Figures 3-1a through 3-1d show the short-term results for the event. The feedwater shutoff reduces core inlet subcooling and reduces power level by the time low level scram is reached.

Figures 3-2a and 3-2b show the long-term results of the case calculated for the uprated unit. After feedwater flow is terminated, level decreases rapidly, reaching scram (at Level 3) and then dropping to Level 2, which initiates RCIC. The longer term calculation conservatively assumes that power remains at rated until the scram, so Level 3 is reached slightly sooner. RCIC flow begins at ~80 seconds, and the minimum level inside the core shroud occurs at ~1300 seconds. Thereafter, level recovers slowly until the operator manually takes over control or until the level reaches the high level trip (L8).

3.1.2 Conclusion

All BWR/4, 5 and 6 plants will maintain adequate water level for loss of feedwater flow transients for uprated power operation.

<u>Applicability of 3.1</u>: BWR/4, 5 & 6

3.2 STABILITY

On-going activities by the BWR Owners' Group (BWROG) and the NRC are addressing ways to minimize the occurrence and potential effects of power oscillations that have occasionally been observed for certain BWR operating conditions. The NRC has documented its concerns in NRC Bulletin No. 88-07 and Supplement 1 to that bulletin (Reference 3-1).

While a more permanent resolution is being developed, procedures have been incorporated which restrict plant operation in the high power, low core flow region of the BWR power/flow operating map. Figure 3-3 shows the power/flow map regions that have been defined in accordance with this bulletin. Specific operator actions have been established to provide clear instructions (depending upon plant type) for the possibility that a reactor inadvertently (or under controlled conditions) enters any of the defined regions.

It is not the purpose of this generic evaluation to change the currently established regions of operational restrictions or the actions defined for the operator. Instead, it is intended to clarify the continued application of the requirements of the bulletin during uprated power operation. Up to this point in time, all the power references associated with the stability constraints have been defined in terms of the current rated power of the plants. The focus of this evaluation is to redefine the pertinent restrictions in terms of uprated power.

A rescaled version of the power/flow map is provided in Figure 3-4, showing its adjusted form for uprated operation. It also shows the restricted regions that correspond to the NRC Bulletin constraints.

A key aspect of the map which does not change is the amount and scaling of core flow. Therefore, 40%, 45%, and 100% core flow will continue to have the same meaning as before power uprate.

Changes to the flow control rod lines have been introduced to maintain the same interim operating restrictions. First, if the plant had previously been licensed to the maximum extended operating domain (MEOD) defined in Appendix C of Reference 1-1, the upper rod line is reduced proportional to the uprate. This changes the core flow at which 100% power is

achieved from 75% to 81%. Plants that have licensed smaller ranges of operation will utilize the same 100% power/core-flow intercept point as before uprate.

The former "80%" and "100%" flow control rod lines have been redefined in terms of the new, uprated power level. These changes maintain the same intent as the current interim requirements when the definitions are reduced proportional to the amount of the uprate: 100%/104.3%. Therefore, on the new map, the regions are described by the "77%" and the "96%" flow control/rod lines.

The power/flow map is important to plant uprating in many ways. The interim agreements and constraints continue to be honored for uprated operation. Final resolution can continue to proceed as directed by the joint effort of the BWROG and the NRC.

Each GE fuel type introduced for use in the reactor is evaluated to ensure that stability behavior is acceptable using methods and criteria already approved by the USNRC. This practice will be continued for GE reload fuel supplied for uprated cores. Equivalent alternate practices will be followed for fuel supplied by others. Therefore, current constraints on fuel designs that are intended to maintain stability margin will be continued for uprated operation.

Applicability of 3.2: BWR/4, 5 & 6

3.3 CORE SPRAY DISTRIBUTION

The applicability of core spray distribution analysis assumptions at power uprate conditions discussed herein are based on GE LOCA/ECCS methods. The core spray distribution assumptions, modeling and application methodology in SAFER/GESTR-LOCA analyses for BWR/4, 5 & 6 plants were found to be adequate at uprated power conditions.

3.3.1 Short-Term Impact

The LOCA modeling of core spray distribution is conservative. The flow rate to the high power bundles and average core is determined by counter current flow limiting (CCFL) characteristics (e.g., upward steam flow, upper tieplate flow area, etc.). The model allows CCFL breakdown in the peripheral core region only after the upper plenum water level rises above the elevation of the core spray sparger. This conservative model constraint accommodates the increased

power conditions that will be present in the peripheral portions of the uprated core. When peripheral CCFL breakdown occurs, there is rapid drainage of water from the upper plenum to the lower plenum through the peripheral bundles, supporting refilling of the core from the bottom. The result is that there is very little overall credit for core spray distribution during the short-term response to a postulated LOCA, and the effects on core spray distribution are conservatively treated by the GE SAFER/GESTR-LOCA procedures, with or without the differences in decay heat distribution for power uprate.

3.3.2 Long-Term Cooling Impact

Credit is taken for core spray distribution while decay heat is still high and the water level in the core cannot be maintained above the top of the active fuel. For these conditions, at least one core spray loop will be operating. Test data supporting spray distribution are conservatively based on the short-term portion of the transient when power levels and steam generation from the core or depressurization are much higher than during the long-term portion of the transient. Any steam being generated in the long-term is easily quenched and the fuel is adequately cooled by the large core spray flow rate even with power uprate.

3.3.3 Conclusion

There is no significant impact on the analysis assumptions related to core spray distribution as a result of power uprate. Short-term effects of power uprate on core spray distribution are conservatively treated by SAFER/CESTR-LOCA analysis procedures. The effects of power uprate on core spray distribution during long-term cooling are negligible.

Applicability of 3.3: BWR/4, 5 & 6

3.4 SAFETY LIMIT MINIMUM CRITICAL POWER RATIO (SLMCPR)

GE has reviewed the generic safety limits for GE fuel and found them to be acceptable for most BWR/2, 3, 4 & 5 product line plants for the increase in power density associated with power uprate.

The SLMCPR will be confirmed for each plant by comparing the uprated core average bundle power to the applicable SLMCPR basis in the USNRC approved GE licensing methodology. In most cases, the current generic safety limits for GE fuel will be valid. However, if the uprated core average bundle power exceeds the previously documented basis, a new safety limit will be established using the same NRC approved procedures, thereby assuring that the same licensing safety margins are maintained.

Applicability of 3.4: BWR/2-5

3.5 CONTAINMENT ATMOSPHERE COMBUSTIBILITY

10CFR50.44 requires combustibility control equipment with the capability of handling an amount of hydrogen equivalent to that generated from either (1) a metal-water reaction involving 75% of the fuel cladding surrounding the active fuel region (excluding the cladding surrounding the plenum volume for Mark III type containments) or (2) for Mark I and II type containments an inerted atmosphere, and (a) in compliance with 10CFR50.46, five times the amount of hydrogen calculated for 10CFR50.46 compliance or the amount that would result from reaction of all the metal in the outside surfaces of the cladding cylinders surrounding the fuel (excluding the cladding surrounding the plenum volume) to a depth of 0.00023 inch, whichever amount is greater, or (b) if not evaluated against 10CFR50.46, the amount of hydrogen resulting from the reaction of 5% of the mass of metal in the cladding cylinders surrounding the fuel (excluding the cladding surrounding the plenum volume).

In achieving power uprate, the reactor fuel assemblies for uprated power have no significant difference in the amount of cladding material. Containment structural integrity also continues to be demonstrated. Therefore, power uprate will not affect the design basis source term that various individual plants applied to the design of installed combustible gas control equipment (hydrogen recombiners, containment hydrogen igniters, containment purge/repressurization systems, etc.).

Applicability of 3.5: All BWRs

3.6 MATERIALS AND COOLANT CHEMISTRY

3.6.1 Intergranular Stress Corrosion Cracking and Erosion/Corrosion

The nuclear power industry and NRC have developed strict guidelines to minimize the primary concern of intergranular stress corrosion cracking (IGSCC) and the secondary concern of erosion/corrosion in BWRs. Programs of periodic monitoring, inspection and replacement of affected plant components and mitigation controls of materials and materials environment (coolant chemistry) in primary fluid systems and secondary power conversion systems are well established for operating plants. The purpose of this generic evaluation is to justify that the evolved constraints are sufficient for operation at the uprated power level.

IGSCC occurs in the primary systems of BWRs in susceptible components (e.g., piping, safe ends, reactor internals) when critical combinations of susceptible materials condition, environment and tensile stress are present. It is important to note that if any of these three factors are not sufficiently present, IGSCC does not occur. Once a crack is nucleated, both chemical and mechanical conditions can promote propagation that could eventually lead to localized penetrations (leaks) in susceptible materials such as sensitized austenitic stainless steels. Inherent material properties (ductility and toughness), excess design stress capability, water chemistry process controls and operating practices that enhance the BWR materials environment by inhibiting crack growth or preventing new crack formation ensure that the probability of structural failure of reactor coolant pressure boundary components remains extremely low.

Although a minor concern in BWRs, erosion/corrosion or flow-assisted corrosion can cause wall thinning in carbon steel in single-phase water or two-phase (water-steam) secondary power conversion systems. Possibily affected large diameter carbon steel piping systems include feedwater and extraction steam. Smaller diameter carbon steel piping systems such as moisture-separator reheater drain lines are also subject to possible erosion/corrosion. Piping geometries and flow restrictions that induce high turbulence and/or cavitation, steam moisture content, water temperature, fluid velocity, carbon steel alloy composition, and water chemistry (pH, oxygen content) are parameters that, in complex combinations, contribute to flow-assisted loss of wall thickness. Wall thinning can lead to eventual structural failure. While such an event could contribute to a primary system transient, potential contribution to public risk is low for these systems in BWRs due to the high coolant oxygen content.

Small primary system stress increases due to the very small primary system pressure increase associated with power uprate could increase the tendency toward IGSCC initiation and propagation in susceptible components. If, however, the stress generated in susceptible components remains below the threshold level or if the coolant chemistry environments are unchanged or improved, the probability of crack nucleation will not be increased. Operating experience has shown that significant reactor water chemistry excursions usually coincide with transient power level changes, not steady-state operation at a particular (even though possibly higher) pressure or power level. Any increased probability for causing a primary system leak due to power uprate is small. Since the plant is adequately protected by process controls, operating practices and leak detection capability (all of which remain unchanged), the impact on public risk is not increased.

In the secondary power conversion systems, the increased (5%) feedwater flow associated with power uprate increases the tendency for erosion/corrosion or cavitation induced wall thinning. Operating experience suggests that transient flow and water chemistry conditions may also play an important role in inducing erosion/corrosion damage. System configuration, materials and coolant chemistry conditions could, if favorable, act against the tendency. While the probability of secondary system structural failures may increase, the increased propensity is also believed to be small.

Power uprate does not significantly increase the potential for IGSCC due to the small change in parameters which cause IGSCC. Maintenance, inspection and procedures already in place are not affected by operation with power uprate and will continue to monitor plant performance for IGSCC and erosion/corrosion effects.

3.6.2 Coolant Chemistry

The increased feedwater flow (5%) and potentially higher source terms associated with power uprate may result in increased radiation levels in primary system carbon steel piping due to crud buildup. This potential increase will be minimized by continued control of impurities in the primary and secondary systems and maintaining proper oxygen level in accordance with industry practices and recommended procedures. The Technical Specification limits on water chemistry are not affected by power uprate.

Applicability of 3.6: All BWRs

Table 3-1

PRIMARY PARAMETERS USED FOR BWR/4, 5 & 6
EVALUATIONS OF LOSS OF FEEDWATER FLOW TRANSIENT

- Power Level
- Decay Heat
- RCIC Design Capacity
- RCIC and (HPCI or HPCS) Low Water Level Initiation Setpoint
- Low Level Scram Setpoint
- RCIC Startup Delay
- Low Level Setpoint for Closure of MSIVs

Table 3-2

RESULTS FOR BWR/4, 5 & 6 LOSS OF FEEDWATER FLOW TRANSIENT

[Difference between Minimum Water Level Inside the Core Shroud and Top of the Active Fuel]

Plant Analyzed	Analysis for Previous Rated Power (104.3%) Level above TAF (ft)	Analysis for Uprated Power (102% of Uprated) Level above TAF (ft)
BWR/4	≥ 6.8	≥ 5.4
BWR/5	≥ 9.8	≥ 7.8
BWR/6	≥ 7.8	≥ 7.0

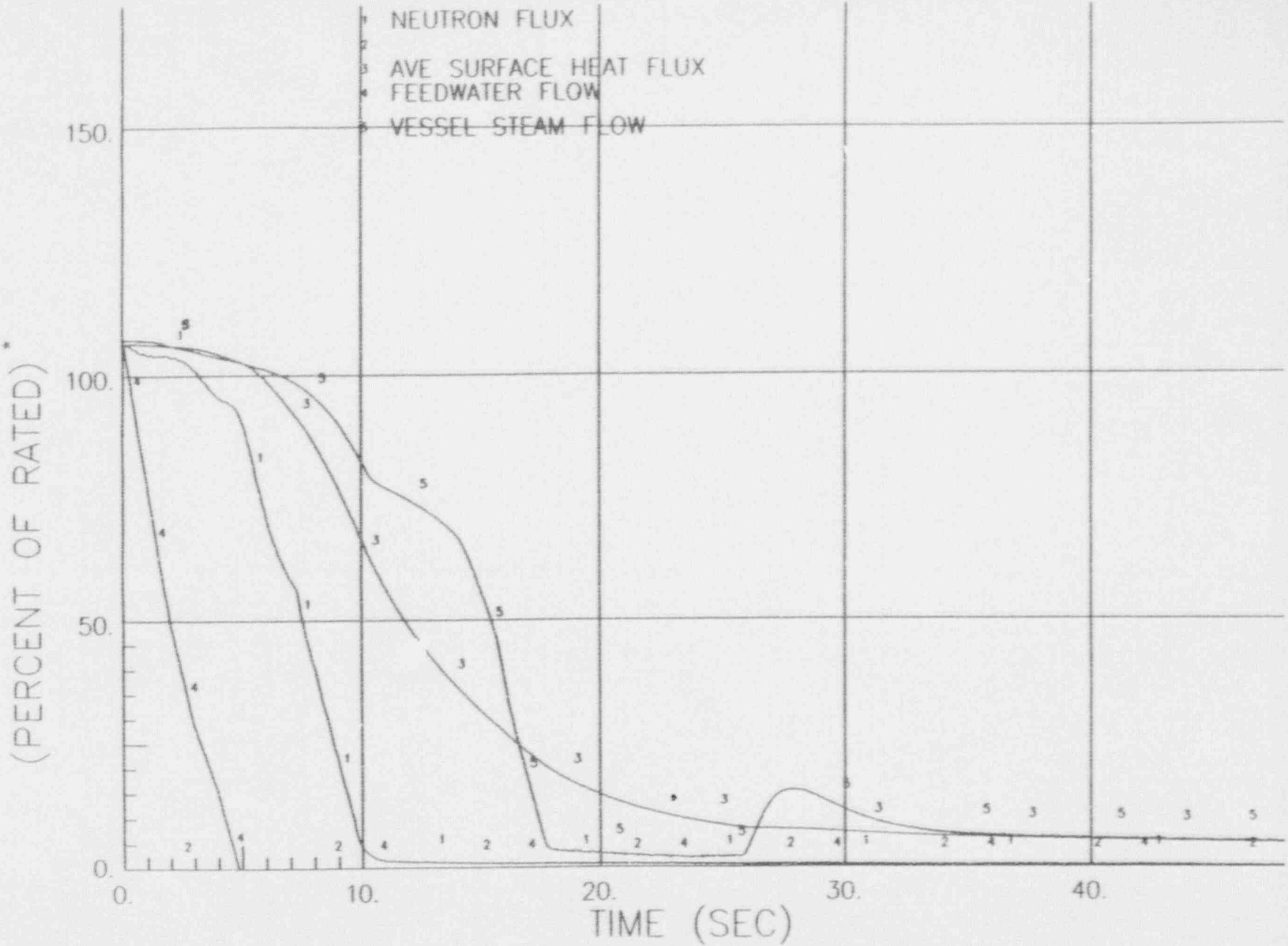


Figure 3-1a. BWR Loss of Feedwater Flow Transient, Short-Term Plot (102% of Uprated Power) (*Percent of Current Rated)

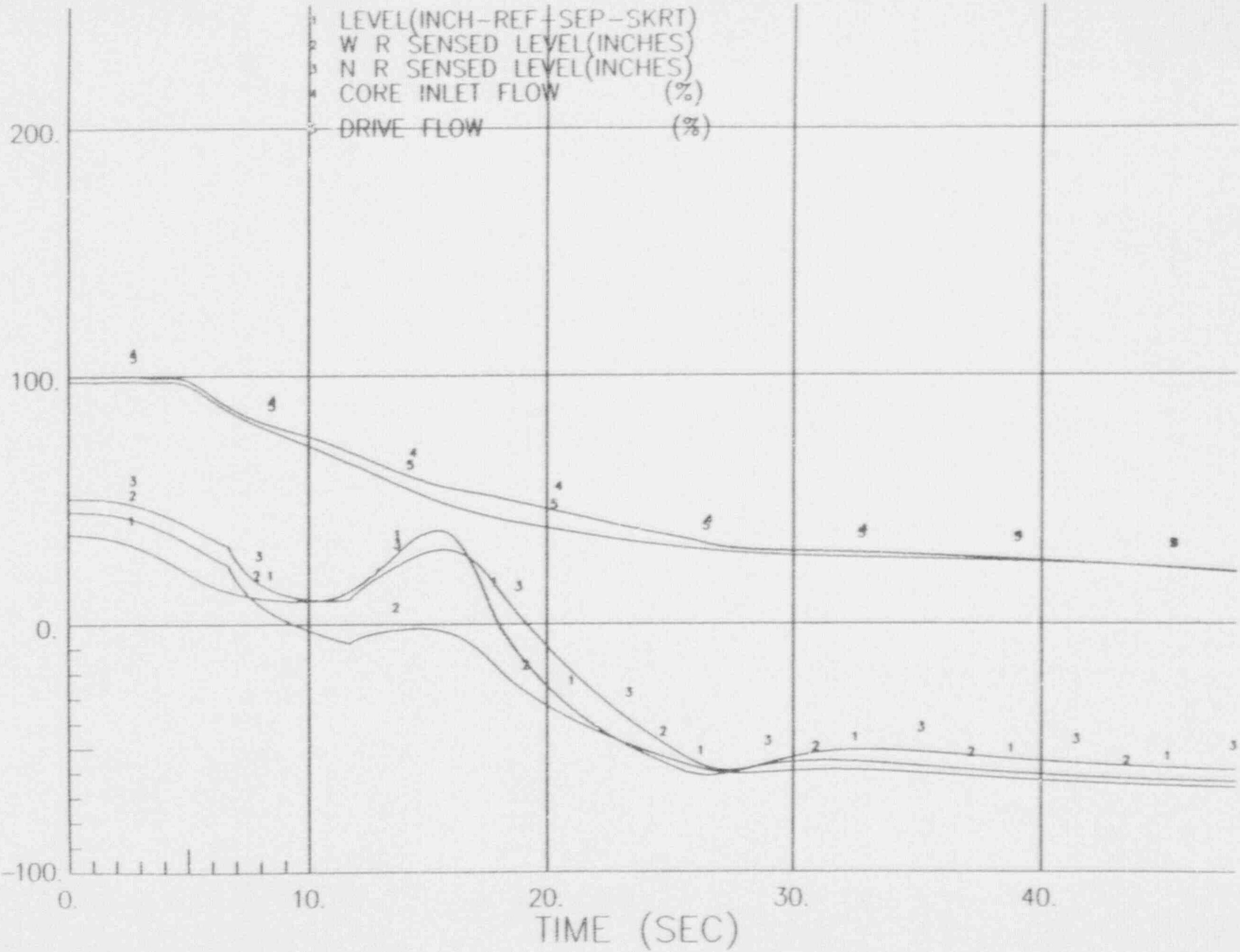


Figure 3-1b. BWR Loss of Feedwater Flow Transient, Short-Term Plot (102% of Uprated Power)

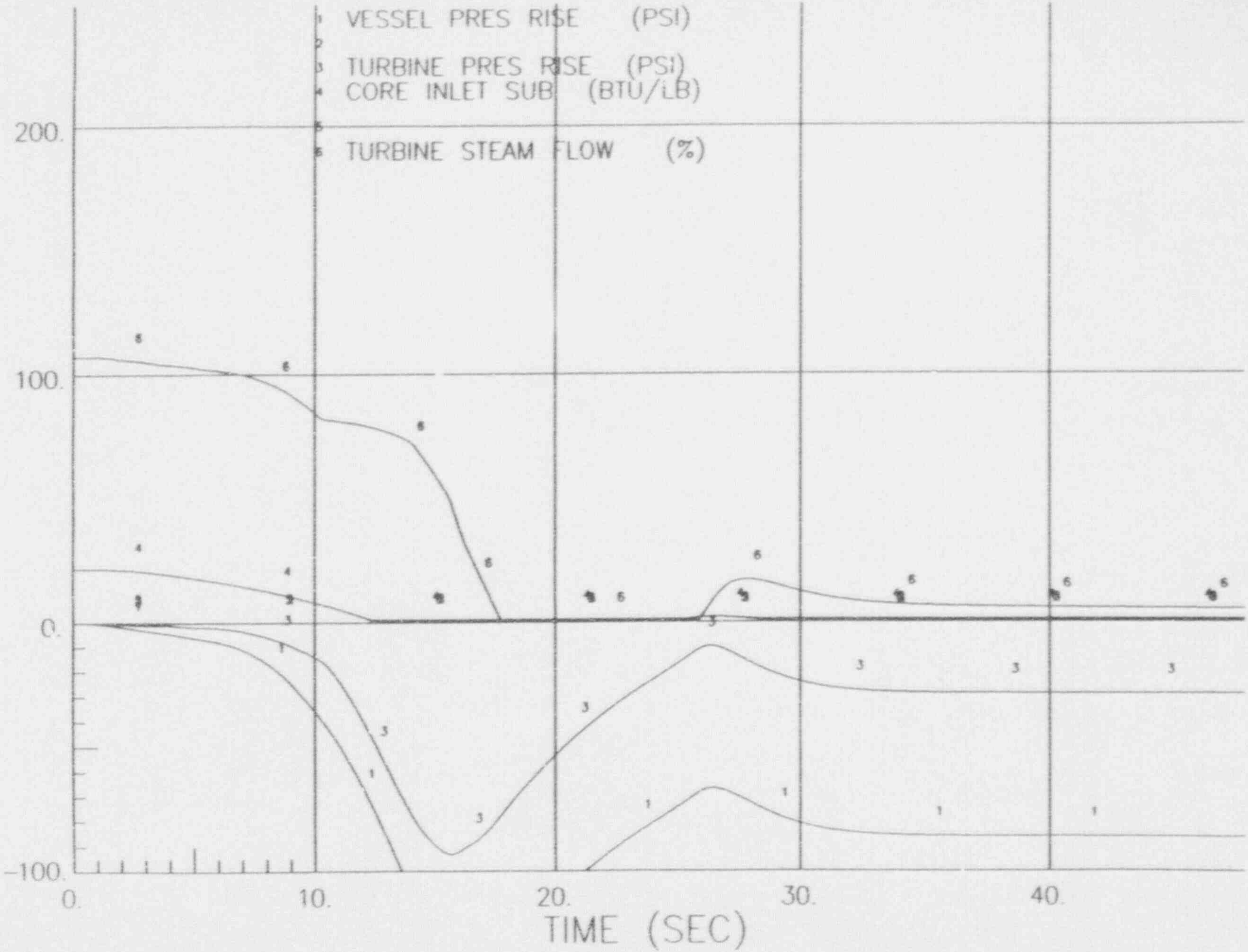


Figure 3-1c. BWR Loss of Feedwater Flow Transient, Short-Term Plot (102% of Uprated Power)

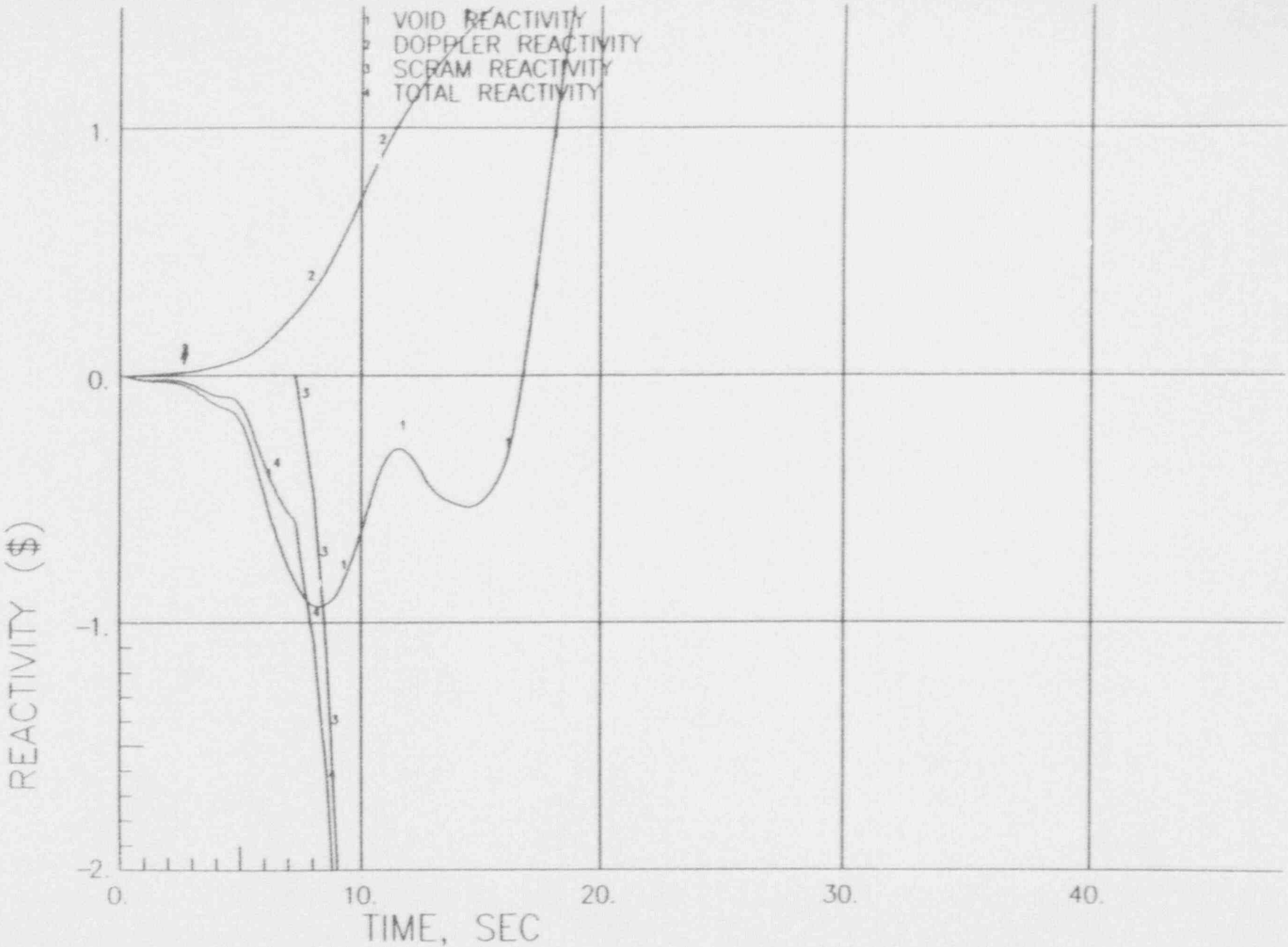


Figure 3-1d. BWR Loss of Feedwater Flow Transient, Short-Term Plot (102% of Uprated Power)

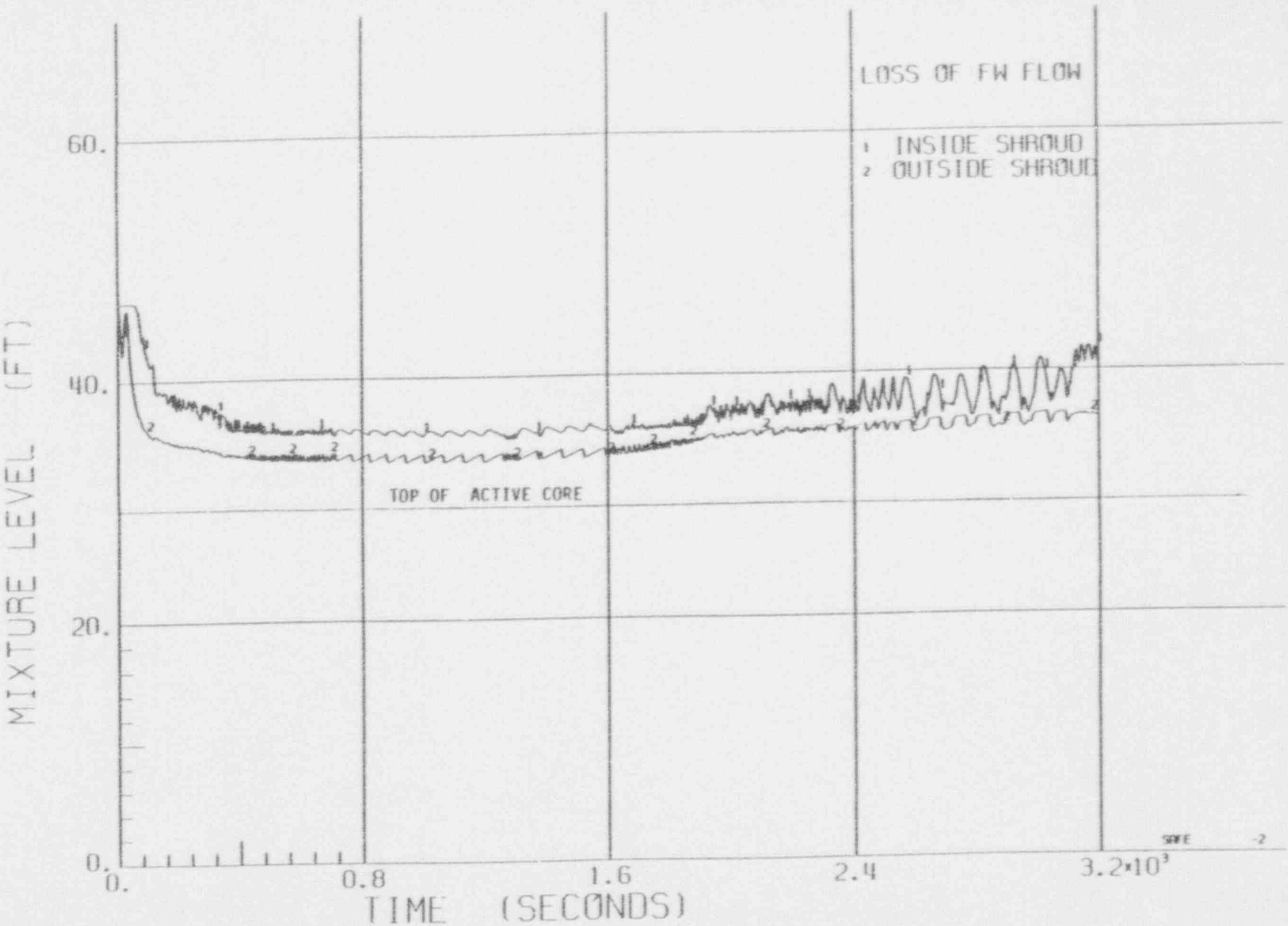


Figure 3-2a. BWR Loss of Feedwater Flow Transient, Long-Term Plot Mixture Water Level (Inside and Outside the Core Shroud) (102% of Uprated Power)

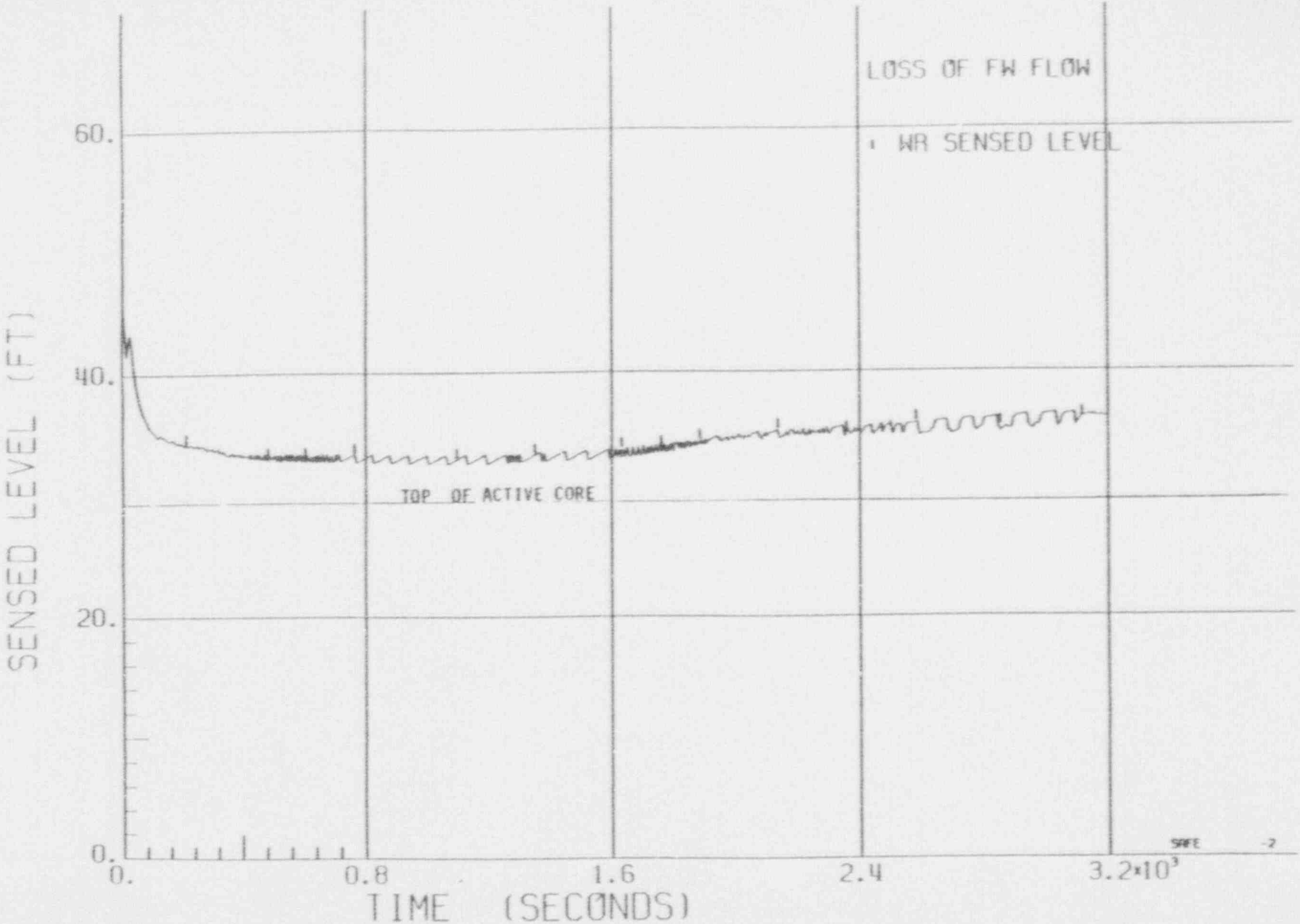
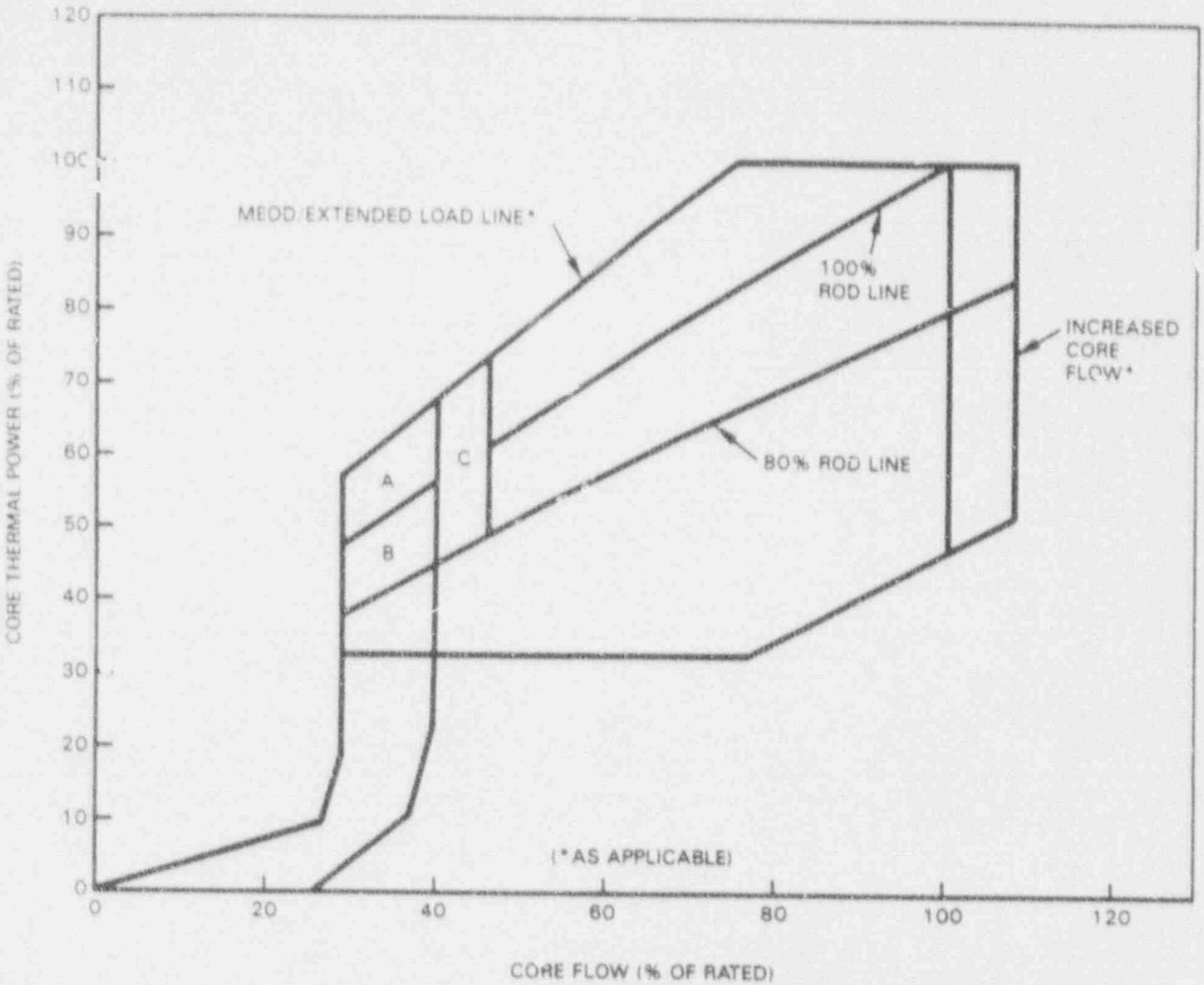


Figure 3-2b. BWR Loss of Feedwater Flow Transient, Long-Term Plot Sensed Water Level (Outside the Core Shroud) (102% of Uprated Power)

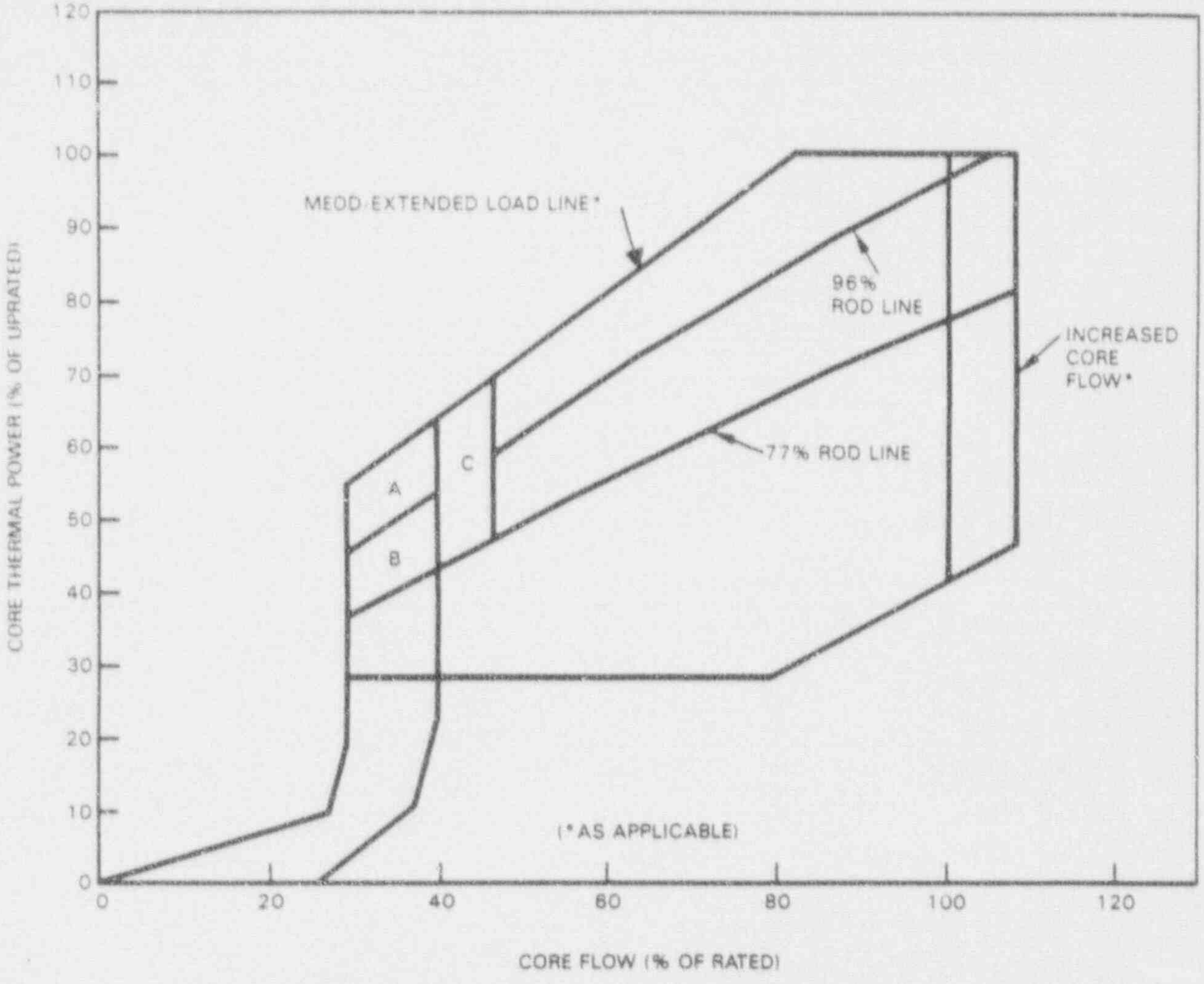


REGION A REACTOR POWER GREATER THAN THE 100% ROD LINE
CORE FLOW LESS THAN 40% OF RATED CORE FLOW

REGION B REACTOR POWER BETWEEN THE 80% AND 100% ROD LINES
CORE FLOW LESS THAN 40% OF RATED CORE FLOW

REGION C REACTOR POWER GREATER THAN THE 80% ROD LINE
CORE FLOW BETWEEN 40% AND 45% OF RATED CORE FLOW

Figure 3-3. Pre-Uprate Power/Flow Map Showing Stability Constraints



REGION A REACTOR POWER GREATER THAN THE 96% ROD LINE
CORE FLOW LESS THAN 40% OF RATED CORE FLOW

REGION B REACTOR POWER BETWEEN THE 77% AND 96% ROD LINES
CORE FLOW LESS THAN 40% OF RATED CORE FLOW

REGION C REACTOR POWER GREATER THAN THE 77% ROD LINE
CORE FLOW BETWEEN 40% AND 45% OF RATED CORE FLOW

Figure 3-4. Uprate Power/Flow Map Showing Adjusted Stability Constraints

4.0 HARDWARE CAPABILITY EVALUATIONS

For each system discussed in this section, the impact due to power uprate is evaluated assuming the following primary operating condition changes:

- o Increased power level of ≤4.3% (i.e., heat flux, stored heat, fission products, neutron fluence) corresponding to the production of 105% of the vessel steam flow at the currently licensed thermal power.
- o Increased reactor pressure (≤ 40 psi).
- o Increased reactor temperature (≤ 5°F).
- o Increased steam and feedwater flow rates of ≤ 5%.

4.1 LOW PRESSURE EMERGENCY CORE COOLING SYSTEM (ECCS)

The hardware for the low pressure portions of residual heat removal (RHR) and low pressure core spray (LPCS) are not affected by power uprate. The upper limit of the low pressure ECCS [LPCS and RHR low pressure coolant injection (LPCI)] injection setpoints will not be changed for power uprate; therefore, the low pressure portions of these systems will not experience any higher pressures. The licensing and design flow rates of the low pressure ECCS will not be increased. In addition, the RHR System shutdown cooling mode flow rates and operating pressures will not be increased. Evaluations will be performed to ascertain if sufficient pump NPSH still exists during accident conditions at the slightly increased peak suppression pool temperature associated with power uprate. These evaluations will be performed in support of the plant-specific power uprate licensing submittal.

Applicability of 4.1: All BWRs

4.2 HIGH PRESSURE COOLANT INJECTION (HPCI) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEMS

The HPCI and RCIC Systems are designed to provide reactor vessel water makeup inventory during small (and intermediate) break loss-of-coolant accidents (HPCI only with other ECCS as

backup), and transients involving loss of feedwater flow (RCIC with HPCI as a backup). Makeup water to the vessel is required to maintain sufficient reactor water inventory, since steam generation continues after shutdown due to reactor core fission product decay heat. A reactor steam turbine-driven pumping system is used by each of these systems to provide makeup water from the condensate storage tank to the reactor vessel. An alternate source of makeup is available from the suppression pool.

For the small break loss-of-coolant accident (LOCA), the reactor water inventory is reduced due to flow through the postulated break. The reactor may be at elevated pressure for some time and the HPCI will provide makeup flow to the reactor vessel during the high pressure portion of this event. Frequently, the most limiting failure for a small break LOCA is failure of the HPCI, so it is backed up by the other ECCS systems.

For an isolation transient, the excess steam production will be released from the reactor vessel through the steam relief valves. For many plants, isolation results in a loss of steam supply to turbine-driven feedwater pumps and a loss of feedwater flow to the reactor. If normal feedwater flow is lost, the HPCI or RCIC will provide the makeup water. The safety/relief valves (SRVs) will open and close (as needed to control pressure) and the HPCI or RCIC will eventually restore the reactor water level to normal. A high water level trip (i.e., Level 8) prevents overfilling the vessel if operator actions do not stop the system(s) sooner.

For a postulated Loss of Feedwater Flow transient without immediate isolation, there will be continued steam flow through the main steam turbine bypass system to the main turbine condenser, with the reactor pressure being maintained at approximately 970 psig. The RCIC or HPCI System will provide sufficient water to prevent the reactor core from being uncovered and eventually the reactor water level will be restored up to or beyond the initial operating water level (i.e., to the Level 8 shutoff if not stopped earlier by the operator). Section 3.1 discusses the flow capacity capability of the current RCIC System to maintain adequate water level for uprated power conditions.

The HPCI and RCIC Systems were designed to provide their rated flows over a vessel pressure range of 150 psig up to a maximum pressure based on the lowest SRV spring safety setpoint. As discussed in Section 4.6, the SRV opening setpoints are generically assumed to be increased ≤ 40 psi for power uprate to maintain adequate simmer margin. Increasing the SRV setpoint pressure has a potential impact on the maximum operating pressure of the high pressure water

injection systems (for events with isolation). This section provides an assessment of the capability of the HPCI and RCIC Systems to meet their design requirements at the new uprated power condition, including the increase in SRV setpoints of up to 40 psi.

The design pressures (and temperatures) of the HPCI and RCIC turbines are 1250 psig at 575°F. The design pressures of the HPCI and RCIC pumps are 1500 psig. Uprated operating conditions are still bounded by these design conditions and System stresses and loads remain acceptable.

4.2.1 Assessment of HPCI Performance

In order to estimate the HPCI pump and turbine operational requirements at the uprated power conditions, the present pump total dynamic head was increased by approximately 3% due to the postulated SRV setpoint increase of approximately 40 psi (93 ft head). The required HPCI design water flow rate remains unchanged. The vendor test curves for the HPCI pumps were used to determine the new speed and horsepower requirements for the HPCI steam turbine. Turbine performance curves developed by the Terry Turbine Company were used to estimate the required increase in steam flow rate and control valve inlet steam pressure at the new conditions. The turbine shaft speed is only increased about 1% at the new operating condition. The overspeed trip setpoints may be reset at 1% higher speed to maintain the difference between normal and trip speeds (Section 4.2.3), but must not exceed a maximum trip speed of 5100 rpm.

The new operating points for both the pump and the turbine are within their allowable operating envelopes. Thus, the HPCI System was found to have the capability to deliver its design injection flow rate at the higher reactor pressures, and the turbine has the capacity to develop the horsepower and speed required by the pump to meet the new pressure requirements.

4.2.2 Assessment of RCIC Performance

The RCIC System operational requirements were reviewed in the same manner as the HPCI System. The RCIC System was also found to have the capability to deliver its design flow at the higher reactor pressures (i.e., increased by ≤ 40 psi), and the turbine has the capacity to develop the horsepower and speed required by the pump to meet the new pressure

requirements. Like HPCI, the RCIC overspeed trip must be considered as discussed in Section 4.2.3 but must not exceed a maximum trip speed of 5740 rpm.

4.2.3 Assessment of Turbine Overspeeding

The HPCI and RCIC Systems are capable of operating at the increased steam supply pressures. The startup transient for the HPCI and RCIC Systems at the potentially higher inlet pressure may result in increased turbine overspeeding, increasing the probability of the System to trip. For the HPCI System, the following (or equivalent) modifications will be implemented (if not already installed): (1) a hydraulic bypass line will be installed around the hydraulic actuator, and (2) a minor change in the procedure for calibrating the turbine's electronic control system will be implemented to control the HPCI turbine overspeed response. These HPCI modifications are for systems equipped with turbine assemblies manufactured by the Terry Turbine Company and have already been distributed to the appropriate BWRs.

The RCIC System startup transient response is also dependent on reactor pressure. Increasing the reactor pressure for power uprate increases the probability of a RCIC turbine overspeed trip on startup. The increase in steam pressure will increase the acceleration rate of the turbine and pump rotors, thus making the governor valve closure rate even more critical (it is initially open). Therefore, if not previously implemented, the RCIC bypass start modification (or equivalent modification) will be installed to assure system availability at the power uprate conditions. The bypass start modification (which installs a small steam bypass line around the steam admission valve) effectively limits the initial peak speed during startup, thus reducing the chance of a turbine overspeed trip.

4.2.4 Conclusion

The HPCI and RCIC Systems are capable of injecting their design flow rates at the higher reactor pressure (i.e., increased ≤ 40 psi) associated with power uprate. However, to avoid the possibility of turbine overspeed trips, a HPCI hydraulic control modification and RCIC bypass start modification will both be installed as part of the power uprate program. Any other methods of mitigating HPCI or RCIC System startup transient response will be justified on a plant-specific basis. The slightly higher pressure and resulting loads on the HPCI/RCIC Systems are acceptable, since the uprated operating conditions remain below the design pressures and temperatures for the Systems.

Applicability of 4.2: BWR/4 (HPCI)
BWR/4, 5 & 6 (RCIC)

4.3 HIGH PRESSURE CORE SPRAY (HPCS)

4.3.1 HPCS Design and Function

The HPCS System on BWR/5 & 6 plants consists of a single, motor-driven centrifugal pump located outside the primary containment, a peripheral ring spray sparger in the reactor vessel located above the core, and associated System piping, valves, controls and instrumentation. The System is designed to operate from normal off site auxiliary power or from a standby diesel generato. supply (if off site power is not available).

The HPCS System is designed to pump water into the reactor vessel over a wide range of System operating pressures. The primary purpose of the HPCS System is to maintain reactor vessel coolant inventory after a postulated small break LOCA. This type of accident does not immediately depressurize the reactor vessel. The HPCS System maintains reactor water level and helps depressurize the reactor vessel. For large break LOCAs, the HPCS System acts primarily after depressurization has occurred to cool the reactor core. In addition, the HPCS System serves as a backup to the RCIC System to provide makeup water in the event of a Loss of Feedwater Flow transient. For this transient, both the RCIC and the HPCS Systems are designed to individually provide sufficient makeup water to prevent the reactor core from being uncovered.

For the Loss of Feedwater Flow transient coupled with reactor isolation, the SRVs will open, then cycle, and the HPCS (and/or RCIC) System will provide sufficient makeup water to keep the fuel covered and eventually return the reactor water level to normal or to the high water level trip (i.e., Level 8 shutoff).

The HPCS System on all applicable reactors was designed to provide makeup water over the entire operating pressure range. Typical HPCS design specification requirements for LOCA application are defined as minimum flow rates at specific vessel pressure differences (vessel minus containment) covering the full range of operation. In addition, there is a specified maximum runout flow into the reactor vessel (i.e., at low pressure in the vessel). The physical equipment is designed to be compatible with the reactor vessel design pressure of 1250 psig, which bounds its potential range of operation.

4.3.2 Post-LOCA Performance

For the large break LOCA analyses, the vessel operating pressure increase associated with power uprate would have little influence on the effectiveness of HPCS, since its role is primarily after depressurization has occurred. For the small break LOCA analyses, the effect of higher vessel pressure on HPCS flow has a second order effect on calculated peak cladding temperature (PCT). The calculated PCT is not affected significantly by the slightly smaller HPCS flow due to the increase in operating pressure. The calculated PCT will be verified on a plant/fuel bundle specific basis and will be documented in the plant-specific report. For many plants, the calculated PCT has large excess design capability to required LOCA criteria, so relaxations of HPCS initiation and/or System performance requirements are being submitted in addition to the power uprate. The HPCS System has ample excess design capability to support power uprate from the viewpoint of LOCA mitigation.

4.3.3 Transient Performance

For the Loss of Feedwater Flow transient with one System assumed failed and one System operating to restore reactor water level, the difference between calculated minimum water level and core uncovery will be greater for the case of the HPCS operating alone than it is for the RCIC operating alone. This is because the HPCS flow rate at a vessel pressure of approximately 1000 psi is at least three times larger than the RCIC flow rate for all applicable plants. When MSIV closure is coupled with the Loss of Feedwater Flow transient, the reactor pressure is increased to the SRV range, reducing HPCS flow capability. It is a GE design basis to ensure that effective HPCS capacity remains greater than RCIC in the pressure range maintained by the lowest SRVs as they cycle open/closed during an isolation event. The most limiting case for the Loss of Feedwater Flow transient is the non-isolation RCIC-alone case, since some inventory is transmitted out of the vessel (to the main condenser). In Section 3.1, the results for the Loss of Feedwater Flow transient event are provided using RCIC alone. A calculated Loss of Feedwater Flow transient for the most limiting case yielded less severe results for the HPCS-alone case (with isolation) as compared to the worst RCIC-alone case (without isolation).

4.3.4 Conclusion

It is concluded that the HPCS design flow rate vs. pressure requirements are adequate to prevent the core from uncovering for the Loss of Feedwater Flow transient for uprated operation. The peak cladding temperature will be confirmed for the LOCA cases on a fuel design specific basis for each plant-specific power uprate licensing submittal. HPCS performance has excess design capability and poses no restriction to power uprate. The physical equipment is designed for conditions that bound the uprated conditions, and therefore it is acceptable for power uprate.

Applicability of 4.3: BWR/5 & 6

4.4 CONTROL ROD DRIVES AND SCRAM PERFORMANCE

The Control Rod Drive (CRD) System controls gross changes in core reactivity by positioning neutron absorbing control rods within the reactor. It is also required to scram the reactor in emergency situations by rapidly inserting withdrawn rods into the core. The CRD System was evaluated at the uprated steam flow and dome pressure conditions.

The increase in dome pressure due to uprate produces a corresponding increase in the bottom head pressure. For pre-BWR/6 plants, the initial control rod insertion is slowed down due to the increased pressure. However, by the time the rod has inserted 20% of the scram stroke, the higher reactor pressure will speed up control rod insertion. Therefore, the small increase in the reactor pressure has no detrimental effect on scram time.

For CRD insertion and withdrawal, the required minimum differential pressure between the Hydraulic Control Unit (HCU) and the vessel bottom head is about 250 psi. The generic CRD pumps were evaluated against this requirement and were found to have sufficient capacity. The flows required for CRD cooling and driving are guaranteed by the automatic opening of the System flow control valve, thus compensating for the small increase in reactor pressure. The flow control valve opening will be slightly modified, as needed, to continue to operate within the optimum range.

If needed to assure the scram time performance indicated in the Technical Specifications, the minimum pressure required in the HCU will be increased by raising the unit pressure. The CRD System pumps can pressurize the HCU accumulators to the new value.

Based on the above, the CRD System will continue to carry out all its functions within current performance requirements at uprated power.

Applicability of 4.4: Pre-BWR/6 plants

4.5 RECIRCULATION SYSTEM

The Recirculation System consists primarily of the recirculation pump, the valves including the recirculation flow control valves (BWR/5 & 6) or the variable frequency motor-generator sets (BWR/4), the jet pumps and the associated piping. This generic evaluation shows that the current design can accommodate the change in operating conditions associated with power uprate with no reduction in safety margin. This generic evaluation is applicable to plants with recirculation System design pressures at least 200 psi higher than the normal operating pressure after the power uprate.

An increase of approximately 4.3% in power will be accompanied by an increase of ≤ 40 psi and 5°F (or less) at the reactor coolant pressure boundary. These increases are small when compared with the original operating conditions of about 1000 psig pressure (in the reactor dome) and a saturated coolant temperature of about 540°F. Thus, it can be seen that the power uprate will only slightly change the normal operating conditions of the Recirculation System. This evaluation covers (1) the effects of the changes to the capabilities of recirculation components to meet reactor coolant pressure boundary requirements, and (2) the effects of the changes to the safety function of the recirculation components.

A review of plant-specific operating data will be performed to confirm that the existing Recirculation System will accommodate the expected small increase in flow resistance due to the increase in core average void fraction at the uprated power condition when operating at maximum core flow. Results will be documented in the plant-specific licensing report. Evaluation of Recirculation System vibration will also be documented in the plant-specific report. This area is predominantly influenced by the maximum core flow (unchanged), rather than power level.

4.5.1 Reactor Coolant Pressure Boundary Requirements

The Recirculation System is safety-related because it is part of the reactor coolant pressure boundary. The design pressures and temperatures of the pumps and valves continue to adequately bound System operation, even at power uprate conditions. The service pressure of the pumps and valves after power uprate will be increased by ≤ 40 psi over the current normal operating pressure. For plants which have recirculation pumps and valves w... a design pressure level at least 240 psi over the current normal operating pressure, there is sufficient excess design capability to easily accommodate power uprate. The contribution from 5°F tem... ...ture increase has only a secondary effect, and is insignificant in the stress analysis calculation, resulting in no loss in available capabilities. Individual analyses of pressure boundary components will be performed to demonstrate that original ASME code requirements are met.

4.5.2 Impact on Safety Function

For some BWR plants which have low pressure coolant injected through the recirculation discharge piping, one or both discharge valves in the recirculation System must close after a LOCA to assure that injected coolant is not lost through a possible broken recirculation pipe. This safety function discharge valve closure is not affected by the ≤ 40 psi System pressure increase, because the reactor has already started to depressurize and the pressure permissive, which permits the recirculation discharge valves to close, is not changed.

<u>Applicability of 4.5.1 and 4.5.2</u>:

- BWR/4 (plants with recirculation system design pressure ≥ 200 psi higher than normal operating pressure after uprate)
- BWR/5 & 6

4.5.3 Recirculation Pumps

Recirculation pumps were originally designed and manufactured to design pressures of 1500-1650 psig. Based on a review of limiting overpressure transients, it has been determined that the existing recirculation pump design pressure and temperature for each plant evaluated will provide adequate excess design capability to accommodate uprated power conditions.

The slightly increased drive flow associated with power uprate would be achieved by slightly increased pump speed (still within design values) for variable speed pumps which are used in BWR/4 plants, or by slightly increased opening of the flow control valve (still within design values) in BWR/5 & 6 plants with constant speed pumps.

The mechanical seals will be capable of handling the slightly higher pressure, since each of the two seal stages has the capability to withstand full reactor pressure. The slightly increased operating pressure will not have any detrimental effect on the other pump or motor bearings, heat exchangers or running clearances.

The slightly increased recirculation pump flow associated with power uprate could result in an incrementally more severe service duty on the recirculation pump shafts, which are susceptible to cracking. Pump shaft cracking is managed by pump vibration monitoring and inservice inspections with or without power uprate.

<u>Applicability of 4.5.3</u>: BWR/4, 5, & 6

4.6 SAFETY/RELIEF VALVES

The performance of BWR safety/relief valves (SRVs) was evaluated under uprated power conditions for the impact of higher steam flow (5%), a temperature increase of approximately 5°F and an operating pressure increase of ≤40 psi.

The increase in steam flow will not affect BWR SRVs. The normal position of SRVs is closed and the opening transient will not be significantly different for transients which are initiated from higher steam flow conditions. The existing SRVs were designed to have sufficient capacity to accommodate the transients which occur from uprated power. Each specific plant uprate report will confirm this capability.

An increase in steam temperature of 5°F will result in approximately 1°F increase in the age-susceptible components of the electro-pneumatic actuator. This is insignificant to the design life of the components.

The SRVs have a design pressure of either 1250 or 1375 psig (plant dependent). The structural integrity of the valves due to the service pressure increase of ≤40 psi (≤1055 psig at the location of the SRVs) is not impacted. Similarly, the small increase in saturated temperature conditions does not impact the integrity of the SRVs.

To ensure adequate simmer margin (the difference between the valve spring opening setpoint pressure and normal operating pressure), the valve spring setpoint pressure is increased proportionally to the operating pressure.

Procedures currently used for recertification of the SRVs shall require revision to provide testing at the higher normal operating pressure. SRVs are currently tested off site at a valve test facility, such that the revision will need to be performed on subvendor documents.

Pressure switches, which are used in some plants to open SRVs in pressure transients, will require resetting. The pressure switch setpoints will continue to be chosen high enough to limit the number of SRV actuations under minor transients, yet low enough to provide the relief action for which credit is taken in transient analyses.

Applicability of 4.6: All BWRs

4.7 MAIN STEAM ISOLATION VALVES

The main steam isolation valves (MSIVs) are part of the reactor coolant pressure boundary and perform important safety functions. The MSIVs must be able to close within 3 to 5 (or 10) seconds (plant dependent) at all design and operating conditions upon receipt of a closure signal. They are designed to not exceed leakage limits set forth in the plant Technical Specifications (Tech Specs).

The changes in operating conditions associated with power uprate are small when compared with the original normal operating conditions of about 1000 - 1030 psig pressure in the reactor dome and a coolant temperature of about 540 - 545°F. The MSIVs are designed to accommodate such small operating condition changes. This evaluation covers (1) the effects of the changes to the structural capability of the MSIV to meet pressure boundary requirements, and (2) the effects of the changes to the safety functions of the MSIV.

4.7.1 Reactor Coolant Pressure Boundary Requirements

Based on review of original and power uprate plant overpressurization analyses, it has been determined that the existing design pressure and design temperature of each plant is adequate, as the operating pressure and temperature changes due to power uprate are very small compared to the original operating conditions. Thus, the impact on the pressure integrity of the MSIV is negligible. The impact of temperature increase is only a secondary effect, and is insignificant in stress analysis calculations. The combined pressure and temperature changes to the specified duty cycles are so small that the impact to the thermal cycle fatigue is also negligible (the existing fatigue usage for the MSIV is very low, typically less than 0.1). The MSIVs will continue to be monitored for compliance with all ASME code requirements for power uprate operating conditions.

4.7.2 Impact on Safety Function

The increase in steam flow rate will increase the assistance to close the MSIVs. Thus, the MSIV closure time should be slightly faster for power uprate conditions. However, closure times of the MSIVs are still constrained by the limits set forth in the plant Tech Specs.

Structurally, the valves are designed to withstand the closure impact from 200% of the original rated steam flow. This upper flow value is based on the size of the steam flow limiting venturi which is not being changed. The equivalent statement based on uprate conditions is 190.5% of uprated steam flow. Higher operating pressure at power uprate conditions tends to result in higher initial seating pressure across the MSIV after isolation, thus reducing leakage. Therefore, valve leakage will be slightly lower for power uprate conditions. Furthermore, MSIVs are under close scrutiny for leakage and closure time from various surveillance requirements in the plant Tech Specs, and their safety performance is routinely monitored.

4.7.3 Other Considerations

Class 1E components, such as MSIV limit switches and solenoid valves, are located close to the valve bonnet and, therefore, could be affected by the slightly higher operating temperature due to power uprate. However, the design conditions for these Class 1E components are expected to bound the power uprate operating conditions, and any potential impact on the environmental

qualification of these components will not be significant. This will be confirmed on a plant-specific basis to assure that potential accident conditions are bounded.

<u>Applicability of 4.7</u>: BWR/4, 5, & 6

5.0 IMPACT ON SAFETY MARGIN

This section provides a licensing discussion of the impact of power uprate on plant safety margins. The safety margins prescribed by the Code of Federal Regulations (CFR) will be maintained by meeting the appropriate regulatory criteria. Similarly, margin specified by application of the American Society of Mechanical Engineers (ASME) design rules will be maintained, as will other margin-assuring criteria used to judge the acceptability of the plant. Already available excess design capability will easily accommodate the modest increases in system operating conditions at uprated power. Environmental margins will be maintained by not increasing any of the present limits for releases such as ultimate heat sink maximum temperature or plant vent radiological limits.

The impact of power uprate on safety margin in the following categories will be discussed: (1) Fuel Thermal Limits, (2) Design Basis Accidents, (3) Transient Evaluations, and (4) Environmental Consequences.

5.1 FUEL THERMAL LIMITS

No change is required in the basic fuel design to achieve the uprated power levels or to maintain the margins as discussed above. No increase in the allowable peak bundle power is requested for power uprate. A slightly flatter radial power distribution may be utilized to supply the additional total power and still maintain limiting fuel bundles within their constraints (Figure 5-1). The fuel operating limits, such as maximum average planar linear heat generation rate (MAPLHGR) and operating limit minimum critical power ratio (OLMCPR), will still be met at the uprated power level. The plant-specific submittal will confirm the acceptability of these operating limits as set for uprated operation. Reload analysis will continue to meet the criteria accepted by the NRC. New fuel designs will meet acceptance criteria approved by the NRC (e.g., GE fuel will meet the criteria accepted by the NRC as specified in Reference 5-1).

5.2 DESIGN BASIS ACCIDENTS

For BWR licensing evaluations, capability is demonstrated for coping with the full spectrum of hypothetical pipe break sizes in the largest recirculation, steam, and feedwater lines, a postulated break in one of the ECCS lines, and even down to breaks in the size of instrument lines. This

break spectrum concept analytically investigates the full spectrum of large and small, high and low energy line breaks and the success of the plant systems in dealing with them while accommodating a single active equipment failure in addition to a postulated LOCA. Several of the most significant licensing assessments are made using these LOCA ground rules. These assessments are:

- o <u>Challenges to Fuel (ECCS performance analyses)</u>: The predominant licensing criterion is fuel peak clad temperature (PCT), but other 10CFR50.46 criteria are evaluated, as necessary.

- o <u>Challenges to the Containment</u>: Key parameters defining challenges to the containment are (1) maximum containment pressure and (2) maximum suppression pool temperature.

- o <u>Radiological Releases</u>: The key parameter is the quantity of activity released to the environment as compared to regulatory criteria.

5.2.1 Challenges to Fuel

The ECCS performance evaluation is conducted through application of the 10CFR50 Appendix K evaluation models and then showing conformance to the acceptance criteria of 10CFR50.46. The results of ECCS-LOCA analysis at uprated power conditions will be provided in the plant-specific power uprate licensing report. A spectrum of pipe breaks is investigated from the largest recirculation line down to instrument lines. As shown schematically in Figure 5-2, the licensing safety margin is not impacted by power uprate. The increased PCT consequences for power uprate are insignificant compared to the large amount by which the results are below the regulatory criteria.

5.2.2 Challenges to the Containment

The plant Safety Analysis Report (SAR) provides the results of analyses of the plant containment response to various large and small LOCAs. The impact of power uprate on the plant-specific peak values for containment pressure and temperature will be provided in the plant-specific power uprate licensing report, as will the impact of power uprate on the conditions which affect the containment dynamic loads. Where the [illegible]t conditions with power uprate are within the range of conditions used to define the dyna[illegible]s, current safety criteria are met and no further structural analysis will be done.

5.2.3 Design Basis Accident Radiological Consequences

Radiological consequences are evaluated in the plant SARs for each of the design basis accidents (DBAs). The magnitude of the potential consequences is dependent upon (1) the quantity of fission products released to the environment, (2) the atmospheric dispersion factors and (3) dose exposure pathways. For the case of power uprate, the atmospheric dispersion factors and the dose exposure pathways do not change. Therefore, the only factor which will influence the magnitude of the consequences is the quantity of activity released to the environment. This quantity is a product of the activity released from the core and the transport mechanisms between the core and the effluent release point.

The only transport mechanism influenced by power uprate is the quantity of coolant mass discharged to the environment for a steamline break or instrument line break accident. For the steamline break accident, the largest coolant mass release typically occurs at hot standby conditions (unaffected by power uprate). For power uprate operation, there is more steam generated; therefore, there is a larger quantity of steam present at any time. For a steamline break occurring from high power, the larger steam volume results in a smaller level swell during the depressurization caused by a steamline break. Consequently, less liquid mass is entrained out the break. Thus, power uprate will not result in an increase in the mass loss for the steamline break accident. For the instrument line break accident, the increased mass loss will be, at most, proportional to the ratio of the reactor pressure increase to the existing reactor pressure (less than 40/1000, about 4%), which is less than the power increase.

The quantity of activity in the primary coolant and in the offgas to be used in the evaluation of these postulated events is unaffected by power uprate, since these values are based on technical specification limits, which remain unchanged for power uprate. It is therefore concluded that for these two accidents, the increase in radiological consequences will be, at most, proportional to the increase in power.

For the remaining DBAs, the only parameter of importance is the activity released from the fuel. Since the mechanism of fuel failure is not influenced by power uprates on the order of 4.3%, the only parameter of importance is the actual inventory of fission products in the fuel rod. Therefore, the increase in the quantity of fission products can be assumed to be proportional to the increase in power.

Based on the quantity of activity in the reactor coolant and the inventory in the reactor core, the radiological releases for the DBAs are either unchanged or increase, at most, by the amount of the uprate. This includes the DBA LOCA, as defined by Regulatory Guide 1.3, which has historically been postulated to produce the most severe radiological consequences. Since most of the radiological assessments presented in plant SARs for the original power level were actually made for 104.3% of the original power level, the assessment for these events at uprate will usually be only a 2% increase in the calculated dose. In the worst assumed case, the dose increase will not exceed the percent of power increase. Using currently approved GE methodology, the DBA LOCA analyses for all BWR/4 - 6 plants show significant margins below the threshold of fuel cladding failure.

A review of the postulated DBA radiological consequences in those plants' SARs which are currently considering power uprate has shown, in general, that, given an increase in the existing consequences proportional to power uprate, the onsite and offsite exposures are below regulatory guidelines for all DBAs. Thus, the postulated radiological consequences are within the regulatory guidelines and all radiological safety margins are maintained.

5.3 TRANSIENT EVALUATIONS

The effects of plant transients are evaluated by investigating a number of disturbances of process variables and malfunctions or failures of equipment according to a scheme of postulating initiating events. These events are primarily evaluated against the Safety Limit Minimum Critical Power Ratio (SLMCPR). The SLMCPR is a limit that is established using standard NRC-approved procedures (see Section 3.4 and Figure 5-3).

The most limiting transient is slightly more severe when initiated from the uprated power level and results in a slightly larger change in MCPR than that initiated from the original power level (≤0.02 larger change in MCPR for the most limited plant). Table 5-1 gives typical power uprate results for one of the most limiting transients. The operating limit MCPR is increased appropriately to insure that the SLMCPR is not infringed upon when the transient is initiated from the uprated power level. Licensing safety margin is not impacted. This area of compliance will be documented in each plant-specific uprate submittal and confirmed for each cycle of operation in the reload analysis. The licensing safety margin is preserved for uprated operation.

5.4 ENVIRONMENTAL CONSEQUENCES

The environmental effects of power uprate will be controlled at the same levels as for the original analyses. That is, none of the present limits for plant environmental releases, such as ultimate heat sink temperature or plant vent radiological units, will be increased for uprated operation. Figure 5-4 depicts how plant operation would be managed for a plant already on (or near) heat sink limits for a portion of the year such that the existing limits would not be violated with uprate. In this example, the plant would take advantage of uprate when the weather was such that the waste heat could be discharged without exceeding limits, and the plant power would be reduced as necessary to not violate the limit when, for example, the river volume flow was reduced during a dry season. This management scheme is appropriate at both the original or uprated power level should unusual environmental conditions develop which need to be accommodated by the plant. A comparable management scheme is intended for all such environmental limits with which the plant is presently required to comply. The current safety margins are thereby maintained.

5.5 TECHNICAL SPECIFICATION CHANGES

The Technical Specifications (Tech Specs) insure that the plant and system performance parameters are maintained at the values assumed in the Safety Analysis. That is, the Tech Specs (setpoints, trip settings, etc.) are selected such that the actual equipment is maintained equal to or conservative with respect to the assumptions used in the Safety Analyses. Proper account is taken of inaccuracies introduced by instrument drift, instrument accuracy, and calibration accuracy. For example, to assure conservatism in a high water level driven safety analysis, the high water level trip is set lower in the Tech Specs than in the safety analysis to assure that the actual plant behavior will be less severe than that represented by the Safety Analysis. Similarly, the Tech Specs address equipment availability and put limits on equipment out-of-service such that the actual plant can be expected to have at least the complement of equipment available to deal with plant transients as that assumed in the Safety Analysis. Since the Safety Analysis shows that the results are acceptable within regulatory limits, then the plant can be expected to behave even more mildly to hypothetical transients than portrayed by the Safety Analysis and thereby assure the public health and safety. Tech Spec changes consistent with the uprated power level will be made in accordance with methodology already approved for the plant and will continue to provide the same level of protection as the Tech Specs currently issued by the NRC.

5.6 CONCLUSION

The predominant plant licensing challenges have been reviewed to demonstrate that power uprate can be accommodated without exceeding any presently existing regulatory limits applicable to the plant. Therefore, there is no reduction in licensing safety margin due to power uprate.

Applicability of 5.0: All BWRs

Table 5-1

TYPICAL TRANSIENT ANALYSIS RESULTS
FOR A LIMITING EVENT (8x8 FUEL)

Parameter	Original Rated Power	Uprated Power
Transient	Generator Load Rejection with failure of turbine bypass	Generator Load Rejection with failure of turbine bypass
Transient Delta MCPR	0.20	≤ 0.22
Operating Limit MCPR*		
- ODYN Option B	1.28	≤ 1.30
- ODYN Option A	1.32	≤ 1.34

* SLMCPR = 1.07

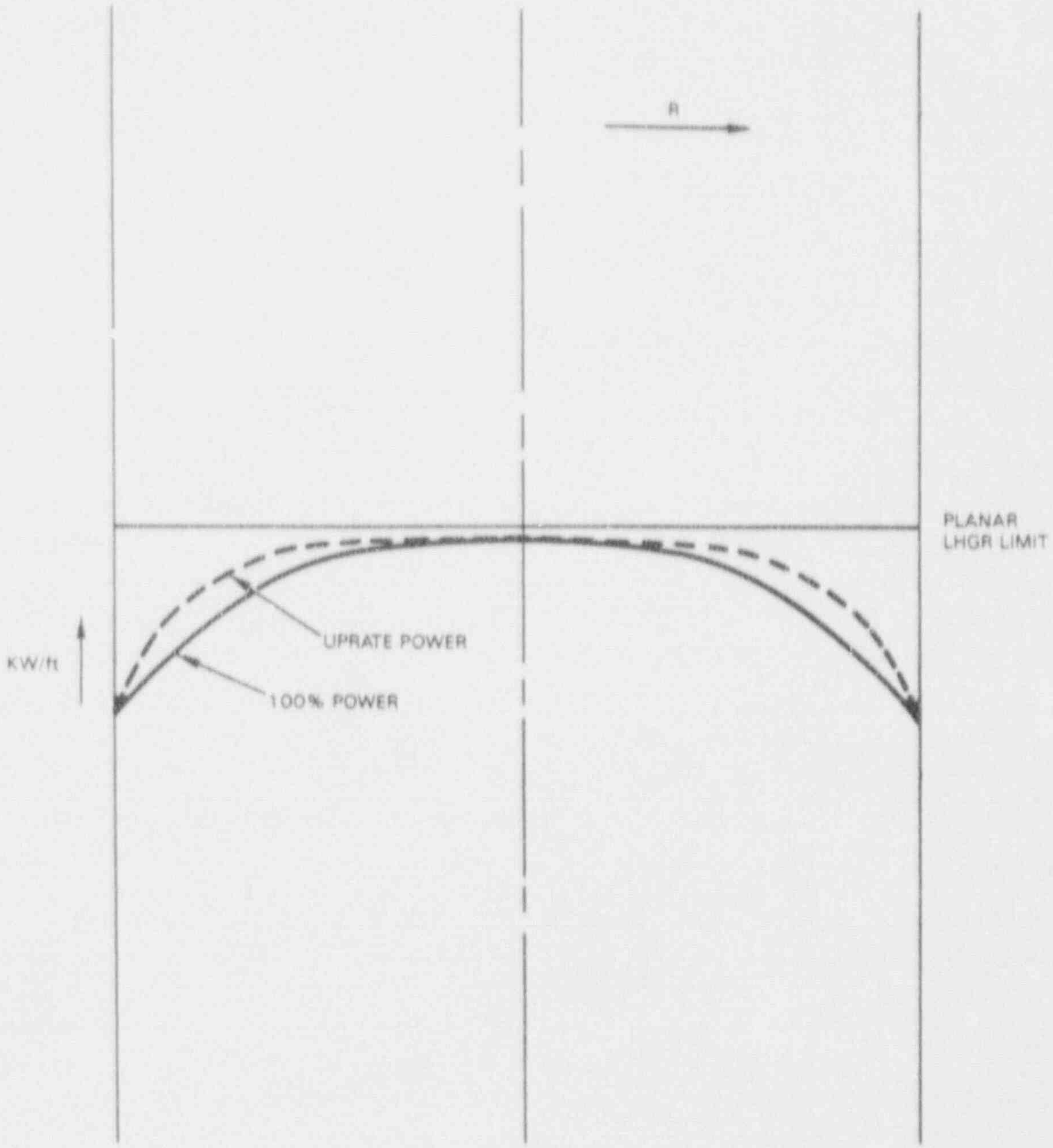


Figure 5-1. Radial Power Profile

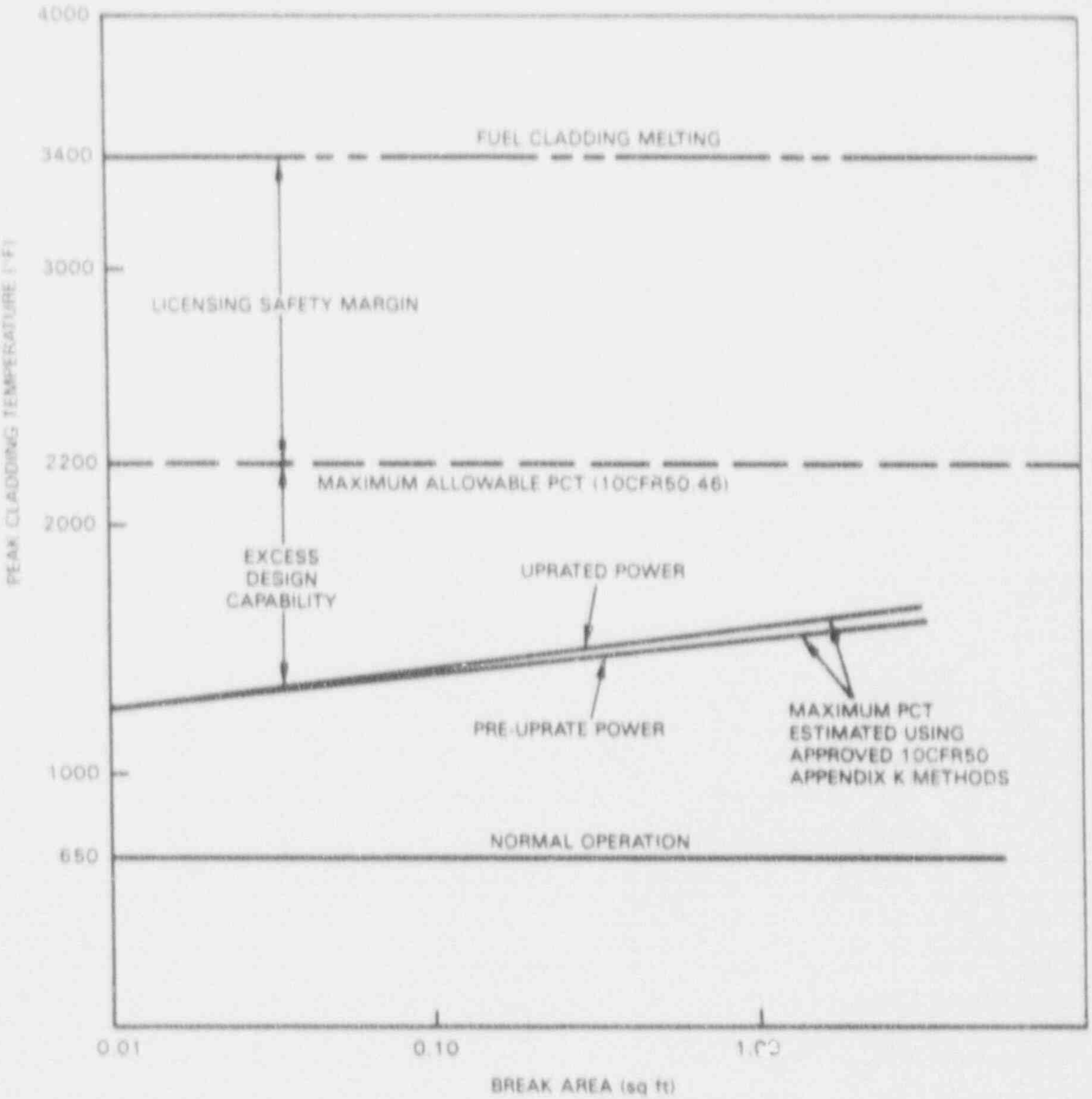


Figure 5-2. Schematic Diagram of LOCA PCT Margin

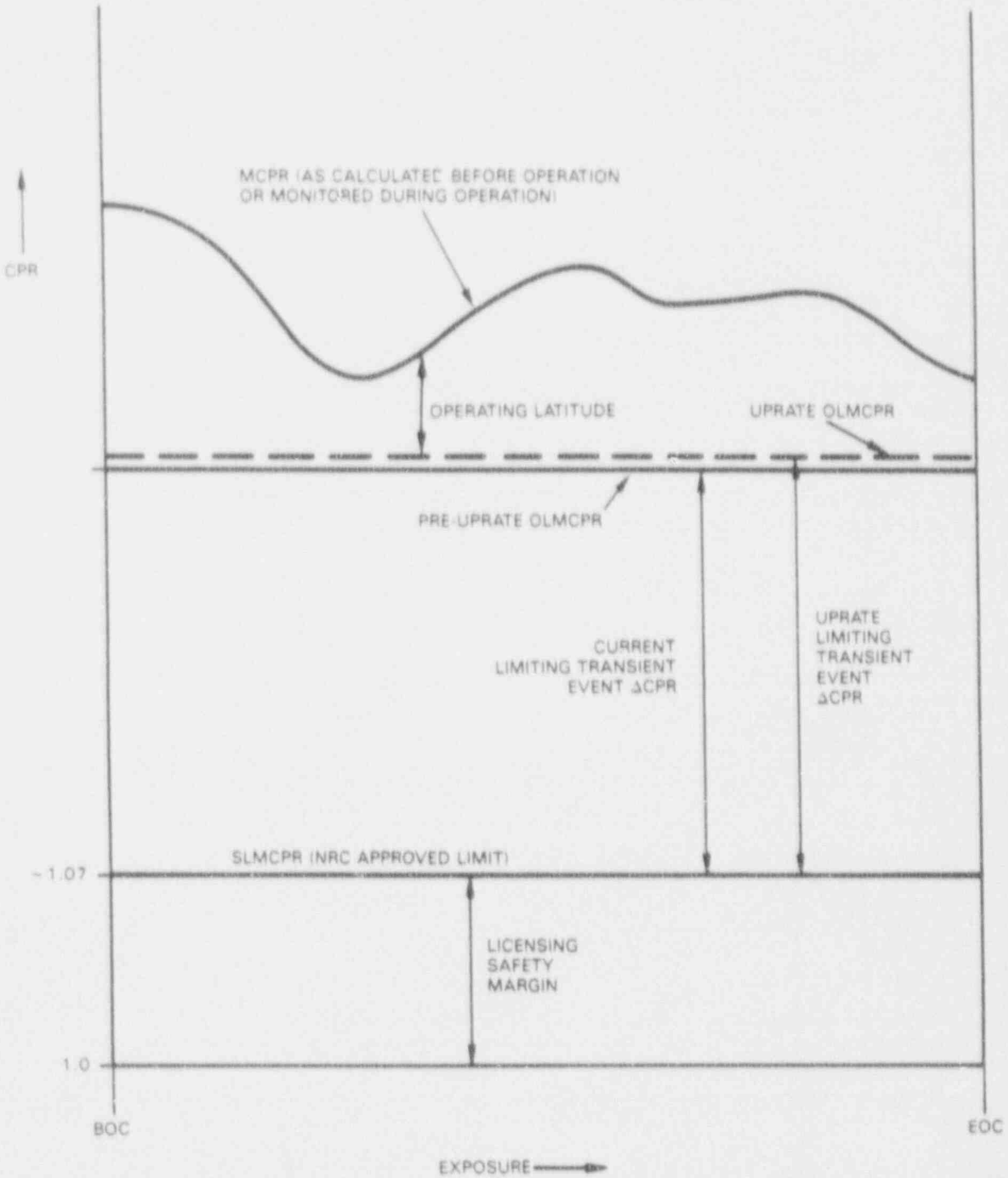


Figure 5-3. Schematic Diagram of Transient Margin

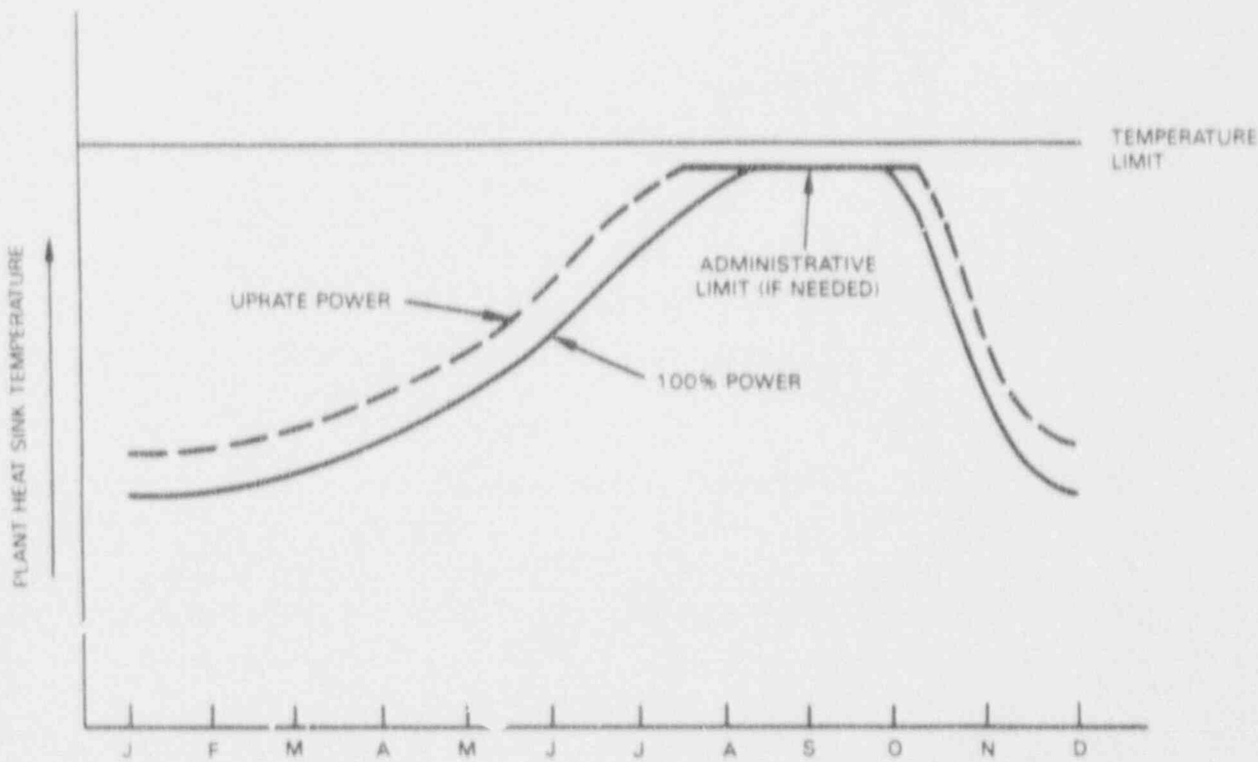


Figure 5-4. Schematic Diagram of Plant Heat Sink

6.0 REFERENCES

1-1 NEDO-31897, "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate", February 1992.

2-1 NEDO-31331, "BWR Owners' Group Emergency Procedure Guidelines Revision 4," March 1987.

2-2 NEDO-31331 Supplement A, "Appendix A to BWROG EPG Revision 4," March 1987.

2-3 OEI Document 8390-4C, "Appendix C to BWROG EPG Revision 4," transmitted by letter from S. T. Rogers to Emergency Procedure Committee Members, December 7, 1987.

3-1 NRC Bulletin No. 88-07 (and Supplement 1), "Power Oscillations in Boiling Water Reactors", December 30, 1988.

5-1 NEDO-31908, "Licensing Criteria for Fuel Designs," January 1991.

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