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Entergy Nuclear Operations, Inc. – FitzPatrick
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(Non-Proprietary Version)

"J. A. FitzPatrick Nuclear Power Plant APRM/RBM/Technical Specifications/ Maximum Extended Operating Domain (ARTS/MEOD)"
NEDC-33087, Revision 1, September 2005

GE Nuclear Energy

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Revision 1
Class I
DRF 0000-0004-9268
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J. A. Fitzpatrick Nuclear Power Plant

APRM/RBM/Technical Specifications / Maximum Extended Operating Domain (ARTS/MEOD)

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J. A. Fitzpatrick Nuclear Power Plant APRM/RBM/Technical Specifications/ Maximum Extended Operating Domain (ARTS/MEOD)

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ACRONYMS

Term	Definition
ADS	Automatic Depressurization System
AL	Analytical Limit
AOO	Anticipated Operational Occurrence
AP	Annulus Pressurization
APRM	Average Power Range Monitor
ARI	Alternate Rod Insertion
ARTS	APRM/RBM/Technical Specifications
ATWS	Anticipated Transient Without Scram
AV	Allowable Value
BOC	Beginning-of-Cycle
BT	Boiling Transition
Btu	British Thermal Unit
BWR	Boiling Water Reactor
CH	Chugging
CLTP	Current Licensed Thermal Power
CO	Condensation Oscillation
ΔCPR	Change in Critical Power Ratio
CRGT	Control Rod Guide Tube
DBA	Design Basis Accident
°F	Degree Fahrenheit
DIVOM	Delta CPR over Initial MCPR Versus the Oscillation Magnitude
DLO	Dual Loop Operating
DTPF	Design Total Peaking Factor
ECCS	Emergency Core Cooling System
ELLLA	Extended Load Line Limit Analysis
ENOI	Entergy Nuclear Operations, Inc.
EOC	End-of-Cycle
F	Core Flow (% of RCF)

Term	Definition
FCL	Flow Control Line
FCTR	Flow Control Trip Reference
FFWTR	Final Feedwater Temperature Reduction
FIV	Flow-Induced Vibration
FLE	Fuel Loading Error
FRFI	Fast Recirculation Flow Increase
FRP	Fraction of Rated Power
FSTF	Full-Scale Test Facility
ft	Foot / feet
FW	Feedwater
FWCF	Feedwater Controller Failure
FWTR	Feedwater Temperature Reduction
GE	General Electric
gpm	Gallon per minute
HCOM	Hot Channel Oscillation Magnitude
HEM	Homogeneous Equilibrium Model
hr	Hour
IBA	Intermediate Break Accident
ICF	Increased Core Flow
ICGT	Incore Guide Tube
in	Inch
IRLS	Idle Recirculation Loop Start-up
JAF	J. A. Fitzpatrick Nuclear Power Plant
JPSL	Jet Pump Sensing Line
lb or lbs	Pound or pounds
lbf	Pounds-force
lbm	Pounds-mass
LDR	Load Definition Report
LFWH	Loss of Feedwater Heating
LHGR	Linear Heat Generation Rate

Term	Definition
LHGRFAC	LHGR Multiplier
LOCA	Loss-Of-Coolant Accident
LPCI	Low Pressure Coolant Injection
LPRM	Local Power Range Monitor
LRNBP	Generator Load Rejection with No Bypass
MAPFAC	MAPLHGR multiplier
MAPLHGR	Maximum Average Planar Linear Heat Generation Rate
MCHFR	Minimum Critical Heat Flux Ratio
MCPR	Minimum Critical Power Ratio
MCPR(F)	Flow-dependent Minimum Critical Power Ratio
MCPR(P)	Power-dependent Minimum Critical Power Ratio
MEOD	Maximum Extended Operating Domain
MELLLA	Maximum Extended Load Line Limit Analysis
MFLPD	Maximum Fraction of Limiting Power Density
M/G	Motor-Generator
Mlb	Million Pounds
MOC	Middle-of-Cycle
MOP	Mechanical Over-Power
MPS	Minimum Pump Speed
MSIV	Main Steam Line Isolation Valve
MSIVC	Main Steam Line Isolation Valve Closure
MTPF	Maximum Total Peaking Factor
MWt	Megawatts thermal
N/C	Not Calculated
NFWT	Normal Feedwater Temperature
NPSH	Net Positive Suction Head
N/R	Not Reported
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OLMCPR	Operating Limit Minimum Critical Power Ratio

Term	Definition
OLTP	Original Licensed Thermal Power
OOS	Out-of-Service
P	Core Power (% of CLTP)
ΔPCT	Change in Peak Cladding Temperature
PCT	Peak Cladding Temperature
PLU	Power Load Unbalance
PRFO	Pressure Regulator Failure Open
PS	Pool Swell
psia	Pounds Per Square Inch - Absolute
psid	Pounds Per Square Inch - Differential
psig	Pounds Per Square Inch - Gauge
PUAR	Plant Unique Analysis Report
PULD	Plant Unique Load Definition
RBM	Rod Block Monitor
RCF	Rated Core Flow
RCPB	Reactor Coolant Pressure Boundary
RHR	Residual Heat Removal
RIPD	Reactor Internal Pressure Difference
RMS	Root Mean Square
RPM	Revolutions per minute
RPT	Recirculation Pump Trip
RPV	Reactor Pressure Vessel
RR	Reactor Recirculation
RSLB	Recirculation Suction Line Break
RTP	Rated Thermal Power
RWCU	Reactor Water Cleanup
RWE	Rod Withdrawal Error
SBA	Small Break Accident
sec	Second
SER	Safety Evaluation Report

Term	Definition
SLCS	Standby Liquid Control System
SLMCPR	Safety Limit Minimum Critical Power Ratio
SLO	Single Loop Operation
SRV	Safety-Relief Valve
SRVDL	Safety-Relief Valve Discharge Line
ΔT	Change in Temperature
TBV	Turbine Bypass Valve
TCV	Turbine Control Valve
TLO	Two Loop Operation
TOP	Thermal Over-Power
TTNBP	Turbine Trip with No Bypass
UFSAR	Updated Final Safety Analysis Report
VPF	Vane Passing Frequency
W_c	Core Flow (% of RCF)
W_d	Recirculation Drive Flow

1.0 INTRODUCTION

Many factors restrict the flexibility of a Boiling Water Reactor (BWR) during power ascension from the low-power / low-core flow condition to the high-power / high-core flow condition. Once rated power is achieved, periodic adjustments must also be made to compensate for reactivity changes due to xenon effects and fuel burnup. Some of the factors currently existing at J. A. Fitzpatrick Nuclear Power Plant (JAF) that restrict plant flexibility in quickly achieving rated power are:

1. The currently licensed allowable operating power/flow map; and
2. The Average Power Range Monitor (APRM) flow-biased flux scram and flow-biased rod block setdown requirements.

The Maximum Extended Operating Domain (MEOD) is defined as the combination of the power/flow operating map expansion with Maximum Extended Load Line Limit Analyses (MELLLA) and increased core flow (ICF). MELLLA corresponds to plant operation above the current licensed JAF Extended Load Line Limit Analysis (ELLLA) boundary and ICF corresponds to operation above the current licensed rated core flow (RCF).

The current APRM and Rod Block Monitor (RBM) flow-biased rod block trips restrict the power ascension capability of BWRs. These operating restrictions are further compounded by the existing setdown requirements for these trips. The operating restrictions resulting from the existing APRM and RBM systems can be significantly relaxed or eliminated by the implementation of a series of APRM/RBM/Technical Specifications (ARTS) improvements. These improvements increase plant operating efficiency by updating the thermal limits administration. For the JAF application, the ARTS program will not include the modification of the RBM system from a flow-dependent to a power-dependent system. Therefore, the existing flow-dependent RBM system setpoint is relaxed so that the potential for RBM interference when operating in the MELLLA region can be avoided or minimized. The operating flexibility associated with the ARTS activities complement those of the MELLLA mode of operation. The improvements associated with ARTS, along with the objectives attained by each improvement, are as follows:

1. A power-dependent Minimum Critical Power Ratio (MCPR) thermal limit similar to that used by BWR6 plants is implemented as an update to reactor thermal limits administration.
2. The APRM trip setdown and design total peaking factor is replaced by more direct power-dependent and flow-dependent thermal limits to reduce the need for manual setpoint adjustments and to allow more direct thermal limits administration. This improves human/machine interface, updates thermal limits administration, increases reliability, and provides more direct protection of plant safety.
3. Justification is provided for raising the current flow-biased RBM trip setpoints to a power level outside the new MEOD operating domain.

4. The Rod Withdrawal Error (RWE) evaluation was performed assuming no credit for the rod block signal from the flow-biased RBM setpoint to ensure applicability of the off-rated thermal limits and the flexibility to relax the RBM setpoints.

This report presents the results of the safety analyses and system response evaluations performed for operation of JAF in the region above the rated rod line for a representative core of GE12 and GE14 fuel-types (Cycle 16 core design). The current operating envelope is modified to include the extended operating region bounded by the upper boundary line which passes through the 100% current licensed thermal power (CLTP) / 80% of RCF point, the rated power line, and the rated load line, as shown in Figure 1-1. Plant operational boundaries as shown in Figure 1-1 that are beyond the original licensed allowable operating power/flow map are referred to as the MEOD region. Operation in the MEOD region is intended to enhance the plant operational flexibility and increase plant capacity factor.

1.1 Background

The power/flow operating map (Figure 1-1) includes the operating domain changes for MEOD consistent with approved operating domain improvements for other BWRs. This performance improvement program expands the operating domain to the MEOD upper boundary line, corresponding to approximately the 116% rod line (see Section 1.2.1), to 100% CLTP at approximately 80% of RCF. This operating domain is defined by the following boundaries:

- The MEOD boundary line, extended up to the existing maximum CLTP of 2536 MWt. The MEOD boundary is defined as the line that passes through the 100% CLTP / 80% of RCF state point.
- The CLTP of 2536 MWt.
- The ICF region to 105% of RCF above the cavitation avoidance region.

The MEOD boundary line defines an increase in the extent of the current operating domain above the ELLLA Boundary line currently licensed for JAF.

1.2 ARTS/MEOD Bases

1.2.1 Analytical Bases

A modified power/flow curve has been derived to provide relief from the operating restrictions inherently imposed during ascension to power by the existing power/flow curve. [[

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A 2% allowance in the analytical limit (AL) to account for uncertainties in the core flow to drive flow mapping process, is acceptable under the stability methodology. The ARTS/MEOD application is determined on a plant-specific basis via a safety and impact evaluation for meeting thermal and reactivity margins for BWR plants. When compared to the existing power/flow operating domain, operation in the MEOD region results in plant operation along a higher constant flow control line, which at off-rated operation allows for higher core power at a given core flow. This increases the fluid subcooling in the reactor vessel downcomer region and alters the power distribution in the core that can potentially affect steady-state operating thermal limit and transient/accident analyses results. The effect of this operating mode has been evaluated to support compliance with the Technical Specification fuel thermal margins during plant operation. This report presents the results of the safety analyses and system response evaluations performed for operation of JAF in the region above the ELLLA and up to the MEOD boundary line. The scope of the analyses performed covers the initial application for JAF operation with ARTS/MEOD. For subsequent reload cycles, Entergy will include the ARTS/MEOD operating condition in the plant-specific reload licensing basis.

The safety analyses and system evaluations performed to justify operation in the MEOD region consist of a non-fuel dependent portion and a fuel dependent portion that is fuel cycle dependent. In general, the limiting anticipated operational occurrences (AOOs) MCPR calculation and the reactor vessel overpressure protection analysis are fuel dependent. These analyses, as discussed in this report, are based on the assumption of a representative core with GE14 fuel. Subsequent cycle-specific analyses will be performed by Entergy in conjunction with the reload licensing activities. The non-fuel dependent evaluations such as containment response are based on the current hardware design and plant geometry, and as such they are applicable to JAF. The limiting AOOs, as identified in Reference 1, were reviewed for the MEOD region based on a review of existing thermal analysis limits at plants similar to JAF and use of generic power-dependent and generic flow-dependent MCPR and Maximum Average Planar Linear Heat Generation (MAPLHGR) limits/setpoints. For the fuel-dependent evaluations of reactor pressurization events, these reviews indicate that there is a small difference in the operating limit minimum critical power ratio (OLMCPR) for operation in the MEOD region and the CLTP condition (100% of CLTP / 100% of RCF). The actual operating limit is calculated on a cycle specific basis to bound the entire operating domain. The analyses results also indicate that performance in the MEOD region is within allowable design limits for overpressure protection, loss-of-coolant accident (LOCA), containment dynamic loads, flow-induced vibration and reactor internals structural integrity, and meets the Anticipated Transient Without Scram (ATWS) licensing criteria.

The analyses which justify operation in the MEOD region under the stated conditions are discussed in this report and its supporting references. These analyses include fuel performance event evaluations, mechanical evaluations of the reactor internals, structural vibration assessment, LOCA evaluations, and containment loads evaluations. Nuclear Regulatory Commission (NRC)-approved or industry-accepted computer codes and calculational techniques are used in the ARTS/MEOD analyses. A list of the Nuclear Steam Supply System (NSSS) computer codes used in the evaluations is provided in Table 1-1.

1.2.2 Flow-Biased APRM Scram and Rod Block Design Bases

The purpose of this section is to discuss the setpoint changes for these systems for ARTS/MEOD operation and to provide inputs to the JAF Technical Specifications mark-up process. JAF employs long-term stability Option 1-D that credits the flow biased APRM scram line in the low flow region of the power/flow map for MCPR Safety Limit protection. Further discussion of the APRM High Flux (Flow Bias) scram as it applies to the JAF Option I-D stability solution is included in Section 7.0. Outside of the region of possible instability, the APRM flow biased flux scram line is conservatively not credited in any JAF licensing analyses.

For the current licensed power/flow map, the flow-biased APRM scram line allowable value (AV) is defined as: $0.58*W_d + 66\%$, clamp at 117%, where W_d is the recirculation drive flow in percent of rated, and where 100% drive flow is that required to achieve 100% core power and flow. The flow-biased APRM rod block line AV is currently set at: $0.58*W_d + 54\%$, with no clamp. The RBM flow-biased AV is currently set at: $0.66*W_d + 42\%$, clamp at 110%.

At the current ELLLA conditions, a single APRM High Flux (Flow Bias) scram equation is adequate for both the stability and non-stability related portions of the power/flow map, and the APRM Upscale (Flow Bias) rod block line limit is currently set 12% power below scram, with no maximum. For ELLLA operation, the margin between the APRM flow-biased rod block line and the ELLLA operating boundary line is significantly reduced, in comparison to the operational margin originally available with respect to the 100% rod line.

With the proposed power/flow map expansion to include the MEOD region, the upper boundary of the licensed operating domain is extended. To accommodate this expanded operating domain, and to restore the pre-existing margin between the MEOD boundary line and the flow-biased APRM rod block line and to ensure compliance with the JAF long-term thermal-hydraulic stability solution (see Section 7.0 for further discussion), the following Allowable Values (AV) are redefined for two loop operation (TLO) and for single loop operation (SLO):

APRM flow-biased Scram AVs for TLO are:

$0.38 * W_d + 61.0\%$	for $0 < W_d \leq 24.7\%$
$1.15 * W_d + 42.0\%$	for $24.7 < W_d \leq 47.0\%$
$0.63 * W_d + 73.7\%$	for $47.0 < W_d \leq 68.7\%$
clamp at 117% of CLTP	for $W_d > 68.7\%$

APRM flow-biased Rod Block AVs for TLO are:

$0.38 * W_d + 49.0\%$	for $0 < W_d \leq 24.7\%$
$1.15 * W_d + 30.0\%$	for $24.7 < W_d \leq 47.0\%$
$0.63 * W_d + 61.7\%$	for $47.0 < W_d \leq 78.3\%$
clamp at 111.0% of CLTP	for $W_d > 78.3\%$

APRM flow-biased Scram AVs for SLO are:

$0.38 * W_d + 57.9\%$	for $0 < W_d \leq 32.7\%$
$1.15 * W_d + 32.8\%$	for $32.7 < W_d \leq 50.1\%$
$0.58 * W_d + 61.3\%$	for $50.1 < W_d \leq 95.9\%$
clamp at 117% of CLTP	for $W_d > 95.9\%$

APRM flow-biased Rod Block AVs for SLO are:

$0.38 * W_d + 45.9\%$	for $0 < W_d \leq 32.7\%$
$1.15 * W_d + 20.8\%$	for $32.7 < W_d \leq 50.1\%$
$0.58 * W_d + 49.3\%$	for $50.1 < W_d \leq 106.3\%$
clamp at 111.0% of CLTP	for $W_d > 106.3\%$

In the low flow stability region[1], the scram AVs are based on the scram ALs given in terms of core flow (see Section 7) using the JAF core flow to drive flow relationship, and the AL to AV margin is based on instrument error. The mapping of core flow to drive flow in this region showed that the single scram line with respect to core flow can be approximated by two straight lines with different slopes and intercepts, in the drive flow domain. In the high flow region the APRM trip system utilizes the non-stability based APRM flow-biased scram AV equation. The scram AV is clamped at 117% power (to match current operation) for drive flows greater than 68.7%. The APRM rod block AVs were calculated to have the same margin to scram AV as in the current JAF setpoint calculations, because the rod block is a precurser to scram and there is no reason to change this margin for ARTS/MEOD. A plot of the APRM flow-biased scram and rod block AV lines is shown in Figure 7-1.

The current JAF APRM flow-biased setpoints are implemented by an analog Flow Control Trip Reference (FCTR) card installed in each of the APRM channels. These current JAF FCTR cards can only accommodate a single flow-biased scram equation. The multiple APRM flow biased equations stated above will be implemented for JAF by use of digital FCTR cards.

The flow-biased RBM system being retained at JAF can obstruct operation in the low flow range at high power of the MEOD region. Therefore, the RBM setpoint was relaxed to allow full MEOD operation. Furthermore, the RBM is not credited in the RWE for the reload analysis. Any thermal limit penalties resulting from the consequences of the unblocked RWE are considered in the reload core design such that no operating restrictions are anticipated. The recommended AV for the RBM flow-biased Normal Rod Block line is: $0.66*W_d + 57.3\%$, clamp at 125%. Because no credit is taken for RBM in RWE mitigation, the RBM Clamp and Normal Rod Block can be set at the highest values that can be calibrated in the hardware, so that normal operation is not constrained. There is no change in hardware for the RBM.

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[1] The breakpoint between the stability and high flow regions is chosen conservatively to be close, but with adequate margin to the Buffer Zone boundary.

The above setpoint equations are based on the current GE ARTS/MELLLA methodology, and the current JAF AVs and margins.

1.3 APRM IMPROVEMENTS

The functions of the APRM system are to:

1. Generate trip signals to automatically scram the reactor during core-wide neutron flux transients before the actual core-wide neutron flux level exceeds the safety analysis design bases. This prevents exceeding design bases and licensing criteria from single operator errors or equipment malfunctions.
2. Block control rod withdrawal whenever operation occurs in excess of set limits in the operating map and before core power approaches the scram level.
3. Provide an indication of the core average power level of the reactor in the power range.

The JAF APRM system calculates an average of the in-core Local Power Range Monitor (LPRM) chamber signals. The LPRMs are averaged such that the APRM signal is proportional to the core average neutron flux and can be calibrated as a means of measuring core thermal power. The APRM signals are compared to a recirculation drive flow-referenced scram trip and a recirculation drive flow-referenced control rod withdrawal block trip.

The JAF Technical Specifications require that the plant operate such that the core Maximum Fraction of Limiting Power Density (MFLPD), which is equivalent to the ratio of Maximum Total Peaking Factor (MTPF) to the Design Total Peaking Factor (DTPF), does not exceed the Fraction of Rated Power (FRP). This requirement limits the maximum local power at lower core power and flows to a fraction of that allowed at rated power and flow. If the MTPF exceeds the DTPF, the flow-referenced APRM trips must be lowered (setdown)[2] to limit the maximum power that the plant can achieve. The basis for this "APRM trip setdown" requirement originated under the original BWR design Hench-Levy Minimum Critical Heat Flux Ratio (MCHFR) thermal limit criterion and provides conservative restrictions with respect to current fuel thermal limits.

JAF currently operates under the GE Thermal Analysis Basis critical power correlation, which replaced the minimum critical heat flux basis. The GE Thermal Analysis Basis emphasis on bundle critical power rather than local critical heat flux allows for a more direct determination of fuel thermal limits.

The JAF ARTS/MEOD program utilizes the results of the AOO analyses to define initial condition operating thermal limits, which conservatively assure that all licensing criteria are satisfied without DTPF and setdown of the flow-referenced APRM scram and rod block trips.

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[2] Alternately accomplished by APRM gain increases.

The objective of the APRM improvements is to justify removal of the APRM trip setdown and DTPF requirement. Two licensing areas that can be affected by the elimination of the APRM trip setdown and DTPF requirement are: (1) fuel thermal-mechanical integrity and (2) LOCA analysis.

The following criteria assure satisfaction of the applicable licensing requirements. They were applied to demonstrate the acceptability of elimination of the APRM trip setdown requirement:

1. The Safety Limit MCPR (SLMCPR) shall not be violated as a result of any AOO.
2. All fuel thermal-mechanical design bases shall remain within the licensing limits described in the GE generic fuel licensing report GESTAR-II (Reference 2).
3. Peak cladding temperature and maximum cladding oxidation fraction following a LOCA shall remain within the limits defined in 10 CFR 50.46.

The safety analyses used to evaluate the OLMCPR such that the SLMCPR is satisfied and to ensure that the fuel thermal-mechanical design bases are satisfied as documented in Section 3.0 of this report. These analyses also establish the fuel type specific power-dependent and flow-dependent MCPR, MAPLHGR, and LHGR curves for JAF. The effect on the LOCA response due to the ARTS program implementation is discussed in Section 8.0 of this report.

Table 1-1
Computer Codes Used for ARTS/MEOD Analyses

Task	Computer Code	Version or Revision	NRC Approved	Comments
Reactor Heat Balance	ISCOR	09	Y (1)	NEDE-24011-P Rev. 0 Safety Evaluation Report (SER)
Reactor Core and Fuel Performance	PANAC ISCOR	10 09	Y Y (1)	NEDE-30130-P-A NEDE-24011-P Rev. 0 SER
Reactor Internal Pressure Differences	LAMB TRACG ISCOR	07 02 09	(2) (3) Y (1)	NEDE-20566P-A NEDE-32176P, Rev 2, Dec 1999 NEDC-32177P, Rev 2, Jan 2000 NRC TAC No M90270, Sep 1994 NEDE-24011-P Rev. 0 SER
Transient Analysis	PANAC PANAC ODYN ISCOR	11 10 10 (5) 09	Y Y Y Y (1)	See Note (13) NEDE-30130-P-A (4) NEDE-24154P-A NEDC-24154P-A, Vol 4, Sup 1 NEDE-24011-P Rev. 0 SER
Anticipated Transient Without Scram	PANAC PANAC ODYN STEMP	11 10 10 (5) 04	Y Y Y (6)	See Note (13) NEDE-30130-P-A (4) NEDC-24154P-A, Vol 4, Sup 1
Containment System Response	M3CPT LAMB	05 08	Y (2)	NUREG-0661 and NUREG-0661, Supplement 1 NEDE-20566P-A
Reactor Recirculation System	BILBO	04V	(7)	NEDE-23504, Feb. 1977
ECCS-LOCA	LAMB GESTR SAFER ISCOR TASC	08 08 04 09 03	Y Y Y Y (1) Y	NEDE-20566P-A NEDE-23785-1P-A, Vol 1, Rev 1 (8),(9),(10),(11),(12) NEDE-24011-P Rev. 0 SER NEDC-32084P-A

NA – Not Applicable

Notes:

(1) The ISCOR code is not approved by name. However, the SER supporting approval of NEDE-24011-P Rev. 0 by the May 12, 1978 letter from D. G. Eisenhut (NRC) to R. Gridley (GE) finds the models and methods acceptable, and mentions the use of a digital computer code. The referenced digital computer code is ISCOR. The use of ISCOR to

provide core thermal-hydraulic information in reactor internal pressure differences, Transient, ATWS, Stability, and LOCA applications is consistent with the approved models and methods.

(2) The LAMB code is approved for use in Emergency Core Cooling System (ECCS)-LOCA applications (NEDE-20566P-A), but no approving SER exists for the use of LAMB for the evaluation of reactor internal pressure differences or containment system response. The use of LAMB for these applications is consistent with the model description of NEDE-20566P-A.

(3) NRC has reviewed and accepted the TRACG application for the flow-induced loads on the core shroud as stated in NRC SER TAC No. M90270.

(4) The physics code PANACEA provides inputs to the transient code ODYN. The improvements to PANACEA that were documented in NEDE-30130-P-A were incorporated into ODYN by way of Amendment 11 of GESTAR II (NEDE-24011-P-A). The use of PANAC Version 10 in this application was initiated following approval of Amendment 13 of GESTAR II by letter from G.C. Lainas (NRC) to J.S. Charnley (GE), MFN 028-086, Subject: "Acceptance for Referencing of Licensing Topical Report NEDE-24011-P-A Amendment 13, Rev. 6 General Electric Standard Application for Reactor Fuel," March 26, 1998.

(5) Version 10 of ODYN is applicable to plants that use variable pump speed (i.e., motor generator (M/G) sets, induction motor drive, or internal pumps) for recirculation flow control.

(6) The STEMP code uses fundamental mass and energy conservation laws to calculate the suppression pool heatup. The use of STEMP was noted in NEDE-24222, "Assessment of BWR Mitigation of ATWS, Volume I & II (NUREG-0460 Alternate No. 3) December 1, 1979." The code has been used in ATWS applications since that time. There is no formal NRC review and approval of STEMP or the ATWS topical report.

(7) Not a safety analysis code that requires NRC approval. The code application is reviewed and approved by GENE for "Level-2" application and is part of GENE's standard design process. Also, the application of this code has been used in previous power uprate submittals.

(8) NEDE-30996P-A, "SAFER Model for Evaluation of Loss-of-Coolant Accidents for Jet Pump and Non-Jet Pump Plants," General Electric Company, October 1987.

(9) NEDC-32868P, "GE14 Compliance with Amendment 22 of NEDE-24011-P-A (GESTAR II)," December 1998.

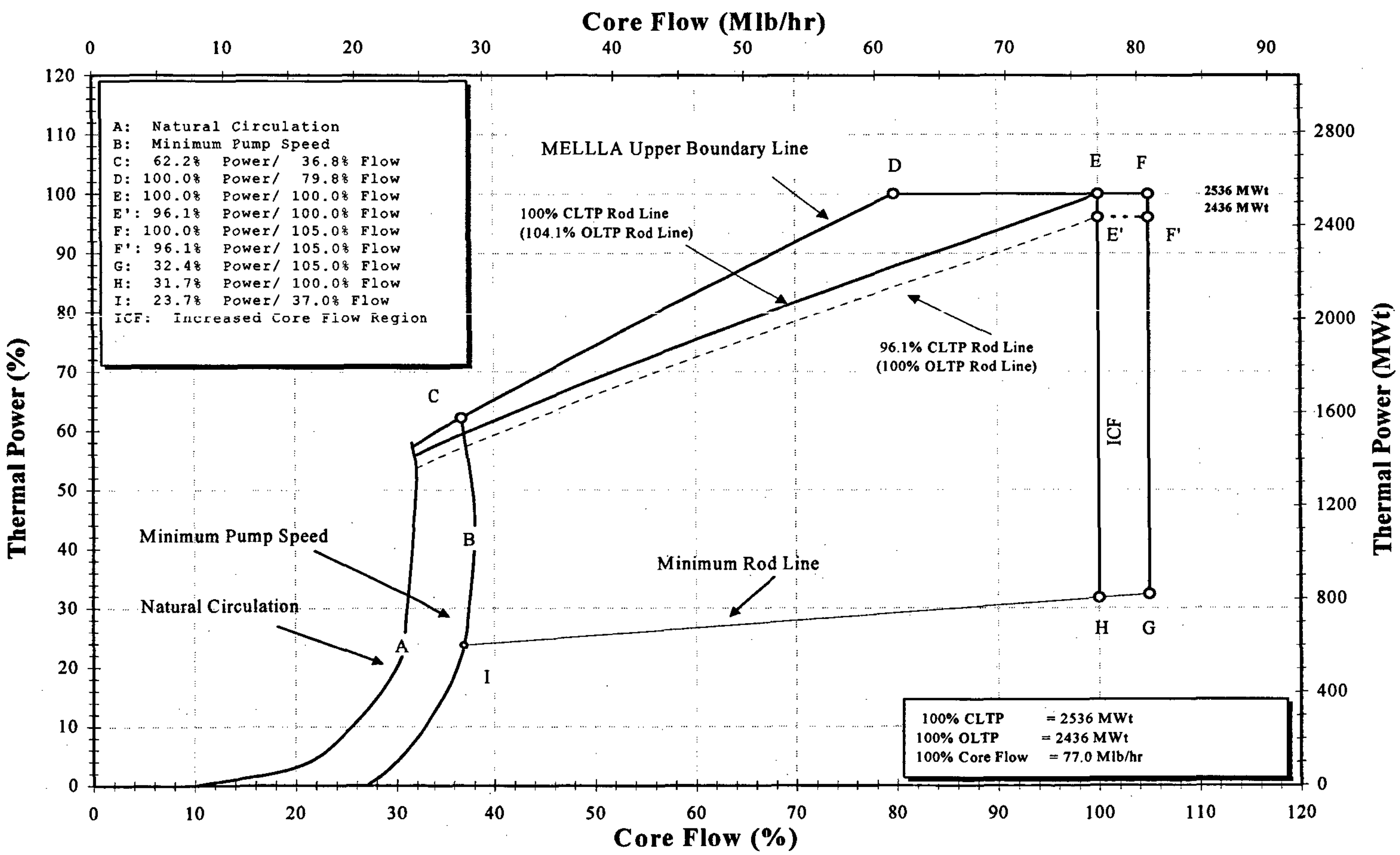
(10) NEDC-32950P, "Compilation of Improvements to GENE's SAFER ECCS-LOCA Evaluation Model," January 2000.

(11) Letter, S.A. Richards (NRC) to J.F. Klapproth (GE), "General Electric Nuclear Energy Topical Reports NEDC-32950P and NEDC-32084P Acceptability Review," May 24, 2000.

(12) NEDE-23785P-A, Vol. III, Supplement 1, Revision 1, "GESTR-LOCA and SAFER Models for Evaluation of Loss-of-Coolant Accident Volume III, Supplement 1, Additional Information for Upper Bound PCT Calculation," March 2002.

(13) The physics code PANACEA provides inputs to the transient code ODYN. The use of PANAC Version 11 in this application was initiated following approval of Amendment 26 of GESTAR II by letter from S.A. Richards (NRC) to G.A. Watford (GE), Subject: "Amendment 26 to GE Licensing Topical Report NEDE-24011-P-A, GESTAR II Implementing Improved GE Steady-State Methods", (TAC NO. MA6481), November 10, 1999.

Figure 1-1
JAF MEOD Power/Flow Map



2.0 OVERALL ANALYSIS APPROACH

This section identifies the analyses that may be affected by the proposed MEOD region. The analyses performed in the following sections assume the current plant operating parameters. For the transient and stability tasks, the JAF Cycle 16 core design is utilized (Reference 1), and these tasks will be revalidated as part of the subsequent cycle-specific reload licensing analyses. For the remainder of the ARTS/MEOD scope of work, the results are applicable to JAF, unless a plant configuration affecting this analysis is changed.

Table 2-1 identifies the safety and regulatory concerns that are potentially affected as a result of the ARTS/MEOD. Each applicable safety and regulatory concern implied in the listed items was reviewed to determine the acceptability of changing the power/flow map to include the MEOD range. In addition, the characteristics of each analyses, whether generic or plant-specific, and cycle-dependent or cycle-independent, are identified in Table 2-2.

Table 2-1
Analyses Presented In This Report

Section	Item	Result
3.0	Fuel Thermal Limits	Acceptable - Bounded by Current Results
4.0	Reactor Recirculation System	Acceptable - Bounded by Current Results
5.0	Reactor Coolant Pressure Boundary	Acceptable - Bounded by Current Results
6.0	Vessel Overpressure Protection	Acceptable - Bounded by Current Results
7.0	Thermal-Hydraulic Stability	Acceptable – New APRM flow-biased flux scram line for ARTS/MEOD
8.0	LOCA Analysis	Acceptable - Bounded by Current Results
9.0	Containment Response	Acceptable - Bounded by Current Results or Design Criteria
10.0	Reactor Internals Integrity	Acceptable - Bounded by Current Results
11.0	ATWS	Acceptable - Bounded by Current Results or Design Criteria
12.0	Steam Dryer and Separator Performance	Acceptable - Bounded by Current Results
13.0	Testing	Acceptable with the performance of the identified tests.

Table 2-2
Applicability of Analyses

Task Description	Generic or Plant-Specific	Cycle-Independent or Cycle-Dependent
Power-Dependent MCPR and Linear Heat Generation Rate (LHGR) limits (between rated power and 29% of CLTP)	Generic, with plant-specific confirmation for initial application	Cycle-independent unless change in plant configuration from licensing analysis basis
Power-Dependent MCPR and LHGR limits (between 29% and 25% of CLTP)	Plant-specific	Cycle-specific review
Flow-dependent MCPR and LHGR limits	Generic, with plant-specific confirmation for initial application	Cycle-independent unless change in plant configuration from licensing analysis basis
ECCS-LOCA	Plant-specific	Cycle-independent unless change in plant configuration from licensing analysis basis

3.0 FUEL THERMAL LIMITS

The potentially limiting AOOs and accident analyses were evaluated to support JAF operation with ARTS off-rated limits, as well as operation in the MEOD region. Analyses were performed to determine the limiting MCPR requirement and the peak vessel pressure based on JAF Cycle 16 fuel and core configuration at 100% of CLTP. The power/flow state points chosen for the review of the AOOs listed in Table 3-1 bound the current licensed operating domain and the ARTS/MEOD region. The minimum core flow at 100% of rated thermal power (RTP) used in the analysis presented in this section is 81% of RCF. Figure 1-1 shows this point as 79.8% of RCF. There is a minimal effect on the results of the fuel thermal limits analysis due to this slight difference in minimum rated core flow at 100% of RTP. In addition, to support the implementation of the ARTS program, analyses were run to determine the off-rated power-and flow-dependent MCPR, MAPLHGR, and LHGR curves, which will allow the removal of the APRM trip setdown. These evaluations are discussed in Sections 3.1 through 3.4. Section 3.5 discusses the governing MCPR, MAPLHGR, and LHGR limits, which also includes consideration of the RWE analyses (Section 3.4) and the LOCA analyses (Section 8.0).

3.1 Approach / Methodology

The core-wide AOOs included in the JAF Cycle 16 reload licensing analyses (Reference 1) and the JAF Updated Final Safety Analysis Report (UFSAR) (Reference 3) were re-examined for operation in the ARTS/MEOD region(including off-rated power and flow conditions). The following events were considered potentially limiting in the ARTS/MEOD region and were reviewed as part of the ARTS program development:

(1) Generator Load Rejection with No Bypass (LRNBP) event;

(2) Turbine Trip with No Bypass (TTNBP) event;

(3) Feedwater Controller Failure (FWCF) maximum demand event;

(4) Loss of Feedwater Heating (LFWH) event;

(5) Fuel Loading Error (FLE) event;

(6) Idle Recirculation Loop Start-up (IRLS) event; and

(7) Fast Recirculation Flow Increase (FRFI) event.

The initial ARTS/MEOD assessment of these events for all BWR type plants concluded that for plant specific applications, only the TTNBP, LRNBP, and FWCF events need to be evaluated at both rated and off-rated power and flow conditions. The LFWH evaluation at 81% flow for JAF Cycle 16, showed that there is a large margin in OLMCPR for the LFWH event compared to the LRNBP event (1.20 for the LFWH versus 1.44 for the LRNBP). Considering that the LFWH event tends to become less limiting as the power decreases (less feedwater to be affected by loss of heating), the LFWH event was not considered in the determination of the off-rated limits. The FLE is a static event that is most limiting at maximum power; thus, this event was also not considered in the determination of the off-rated limits. The other two events (IRLS and FRFI) are by design most limiting at off-rated conditions. Even when originated from their most limiting off-rated condition, the IRLS and FRFI are less limiting than the fast pressurization events (TTNBP, LRNBP, or FWCF) at rated power conditions. Thus, the IRLS and FRFI events were not considered in the determination of the off-rated limits.

The analytical methods as well as the input assumptions, such as reactor protection system setpoints and plant configurations, are consistent with the Reference 1 bases. The power/flow state conditions for the evaluation include the full range of core flow and core power in the operating map. The core flow range encompasses the minimum flow along the MEOD upper boundary line to the 105% maximum core flow. The core power range encompasses the full range from 25% to CLTP. Plant specific heat balance, core coolant hydraulics, and nuclear dynamic parameters corresponding to the off-rated conditions were developed based on JAF Cycle 16 and used in the analysis of the above transient events.

3.1.1 *Elimination of APRM Trip Setdown and DTPF Requirement*

Extensive transient analyses at a variety of power and flow conditions were performed during original development of the ARTS improvement program. This database includes evaluations representative of a variety of plant configurations and parameters such that the conclusions drawn from the studies are applicable to all BWRs. The database was utilized to develop a method of specifying plant operating limits (MCPR, MAPLHGR, and LHGR), which assures that margins to fuel safety limits are maintained for operation at rated and off-rated conditions.

The generic evaluations determined that the power-dependent severity trends must be examined in two power ranges. The first power range is between rated power and the power level (P_{Bypass}) where reactor scram on turbine stop valve closure or turbine control valve fast closure is bypassed. P_{Bypass} for JAF is 29% of CLTP. The second power range is between P_{Bypass} and 25% of CLTP. No thermal monitoring is required below 25% of CLTP, per the JAF Technical Specifications.

Generic power-dependent MCPR and MAPLHGR limits (in terms of multipliers on the plant's rated operating limits) were developed for use in the first power range. JAF specific analyses of limiting transients were performed to confirm the applicability of the generic power-dependent MCPR and MAPLHGR limits.

Between P_{Bypass} and 25% power, JAF specific evaluations were performed to establish the plant-unique limits in the low power range. These plant-specific limits include sufficient conservatism such that they will remain valid for future JAF reloads of GE fuels through the GE14 fuel design, utilizing the GEXL-PLUS correlation and the GEMINI analysis methods provided that the SLMCPR remains below or equal to 1.09.

Generic flow-dependent MCPR, MAPLHGR, and LHGR limits were also developed from the ARTS database.

3.2 INPUT ASSUMPTIONS

The limiting power/flow state condition for the operating region analysis was the rated power and maximum flow point (100%P / 105%F). Figure 1-1 shows the power/flow map used in the AOO analyses. The operating plant heat balance, core coolant hydraulics, and nuclear dynamic parameters corresponding to the rated and off-rated conditions were used for analysis and reflect the JAF Cycle 16 core configuration (Reference 1). The initial heat balance conditions for the AOO analyses at rated and off-rated conditions are presented in Tables 3-1a and 3-1b.

All AOO analyses were performed using the standard reload licensing methodology (Reference 2). The following assumptions and initial conditions were used in the AOOs analyses:

Analytical Assumptions	Bases/Justifications
Initial core flow range of 81% to 105% of RCF for thermal limits transients	Bounding power/flow state points for MEOD
Conservative end-of-cycle 16 nuclear dynamic parameters	Consistent with JAF current licensing bases
Final feedwater temperature reduction (FFWTR) of 80°F (from 424°F to 344°F)	Consistent with the definition of FFWTR of the JAF Engineering Services project
Two lowest opening setpoint Safety-Relief Valves (SRVs) declared Out-of-Service (OOS)	Consistent with JAF licensing bases for Cycle 16
Reference Dual Loop Operating (DLO) SLMCPR = 1.09 for all limits developed in this report	Consistent with JAF licensing bases for Cycle 16.
Turbine bypass assumed operable for the FWCF event analysis	Consistent with JAF current licensing bases
The LFWH, FLE, IRLS, and FRFI events are not limiting at off-rated conditions.	Consistent bases of the ARTS program

3.3 ANALYSES RESULTS

In summary, the operating limits associated with operation in the MEOD region are presented in Table 3-2. The MEOD region will also be incorporated into subsequent cycle specific reload licensing analyses. Key system responses during the analyzed AOOs are shown in Figures 3-7 through 3-10.

3.3.1 Power-Dependent MCPR Limit

As stated in the previous subsection, the generic evaluations indicate that the power-dependent severity trends are to be examined in two power ranges, above and below P_{Bypass}.

In the high power range (between rated power and P_{Bypass}), the trend for the power-dependent MCPR responses for the FWCF with the turbine bypass in service has been shown to be more severe than all other fast pressurization transient severity trends. For the FWCF, the power decrease results in greater mismatch between runout and initial feedwater flow, resulting in an increase in reactor subcooling and more severe changes in thermal limits during the event at offrated power. However, Reference 24 identified a disconnect between the performance of the turbine protection systems and the transient analysis assumptions for a generator load rejection event. In particular, in the operating domain between P_{Bypass} and the point at which the Power Load Unbalance (PLU) system is enabled, the response to a generator load rejection would be a slow closure of the turbine control valves (TCVs). The transient analysis assumes a TCV fast closure, which would initiate a reactor scram. Above the PLU enabling power level, the TCV fast closing function will occur. Therefore, between the PLU power level and P_{Bypass}, the load rejection may be more severe. For JAF, a scram is initiated on high reactor pressure. Analyses were performed with the delayed pressure scram at PLU core power level of 40% and the P_{Bypass} power level.

The corresponding results to verify the generic MCPR(P) limits analyses are summarized in Tables 3-3 and 3-4. A comparison of these plant-specific calculated values with the generic power-dependent MCPR limits verifies the applicability of the generic limits to JAF.

The K(P) above P-bypass can be applied to the TBVOOS events, by multiplying the rated OLMCPRs for this condition by the appropriate K(P).

Below P_{Bypass}, the transient characteristics change due to the bypass of the direct scram on the closure of the turbine stop valve and turbine control valve. In this low power range, the FWCF event, which takes credit for the operability of the bypass, there is no clear severity trend for the TTNBP, LRNBP, and FWCF events. This is because the direct scram on the turbine control valve and the turbine stop valve is bypassed and the scram signal is delayed until the vessel pressure reaches the high pressure scram setpoints for the TTNBP and LRNBP events, which increases the severity of these events. Therefore, the FWCF, TTNBP, and LRNBP events were examined for powers below P_{Bypass}. The extensive transient analyses database also shows a significant sensitivity to the initial core flow for transients initiated below P_{Bypass}. To account for this sensitivity to core flow, two sets of power-dependent limits are determined for power levels below P_{Bypass}, one set for high core flow and one set for low core flow.

Below P_{Bypass}, the MCPR(P) limits are actual absolute OLMCPR values, rather than multipliers on the rated power OLMCPR. These absolute MCPR limits were chosen with sufficient conservatism such that they remain applicable to future operating cycles provided the SLMCPR is less than or equal to 1.09. The JAF specific analyses results at power levels below P_{Bypass} are summarized in Tables 3-5 and 3-6. From these analytical results, bounding plant-specific MCPR(P) limits are then calculated as shown in Table 3-7 and presented in Figure 3-1 for the most limiting condition which includes FFWTR and one main steam line isolation valve inoperable (MSIV-OOS).

In order to support extended operation at JAF with TBVOOS, OLMCPR limits below P-bypass would be required and these limits would be more restrictive than those contained in this report. In addition, for extended operation with TBVOOS, there are necessary plant analyses that should be considered, which are not part of this report.

3.3.2 Power-Dependent MAPLHGR and LHGR Limits

In the absence of the APRM trip setdown requirement, a power-dependent MAPLHGR or LHGR limit, expressed in terms of a MAPLHGR multiplier, MAPFAC(P), or an LHGR multiplier, LHGRFAC(P) are substituted to assure adherence to the fuel thermal-mechanical design bases. Both incipient centerline melting of the fuel and plastic strain of the cladding are considered in determining the power-dependent MAPLHGR and LHGR limits. Generally, the limiting criterion is incipient centerline melting.

The power-dependent MAPFAC(P) and LHGRFAC(P) multipliers were generated using the same database (transient results) as used to determine the MCPR multiplier (K(P)). Similar to the development of the MCPR(P) limits, plant-specific transient analyses were performed to demonstrate the applicability of the generic MAPLHGR(P) and and LHGR(P) limits in the power range above P_{Bypass}. The results of these analyses are shown in Tables 3-8 and 3-9.

As previously discussed, a significant sensitivity to initial core flow exists below P_{Bypass}. Therefore, below P_{Bypass}, both high and low core flow sets of power-dependent MAPLHGR and LHGR multipliers are provided. Appropriate MAPFAC(P) and and LHGRFAC(P) multipliers are selected based on plant-specific transient analyses with suitable margin to assure applicability to future JAF reloads, including exposure ranges of GE fuels through GE14 design. These limits are derived to assure that the peak transient LHGR for any transient is not increased above the fuel design basis values.

The transient and initial condition selection, as well as the approach taken to confirm and develop the appropriate MAPLHGR(P) and and LHGR(P) limits for JAF, are identical to that described in the above section for MCPR(P). The results of plant-specific transient analyses below P_{Bypass} are presented in Tables 3-10 and 3-11. Based on these analyses results, bounding plant-specific MAPLHGR(P) multipliers, MAPFAC(P), are presented in Table 3-12 and shown in Figure 3-2 for the most limiting conditions which include FFWTR and one MSIV-OOS.

3.3.3 Flow-Dependent MCPR Limit

Flow-dependent MCPR limits (MCPR(F)) are necessary to assure that the SLMCPR is not violated during recirculation flow increase events. The design basis flow increase event is a slow-flow power increase event which is not terminated by scram, but which stabilizes at a new core power corresponding to the maximum possible core flow. This event was also used as the basis for the current flow-biased MCPR (K(F)). Flow runout events were analyzed along a constant xenon flow control line assuming a quasi steady-state plant heat balance. The bounding generic flow-dependent MCPR limits are shown in Figure 3-3. To verify the applicability of these generic flow-dependent MCPR limits, recirculation flow runout events were performed for an equilibrium 10 x 10 core loading at a typical mid-cycle exposure condition. In addition, an IRLS was considered generically for the application of ARTS and found to be bounded by the ARTS generic limits. For the application of ARTS, the IRLS basis is that there is an initial 50°F ΔT between the idle and operating loops. This is an appropriate assumption for thermal limits calculations and is consistent with Technical Specification requirements. The ARTS-based MCPR(F) limit is specified as an absolute value and is generic and cycle-independent provided the SLMCPR is less than or equal to 1.09.

3.3.4 Flow-Dependent MAPLHGR and LHGR Limits

Flow-dependent MAPLHGR (MAPFAC(F)) and LHGR (LHGRFAC(F)) limits were designed to assure adherence to all fuel thermal-mechanical design bases. The same transient events used to support the MCPR(F) operating limits were analyzed, and the resulting overpowers were statistically evaluated as a function of the initial and maximum core flow. From the bounding overpowers, the MAPFAC(F) and LHGRFAC(F) limits were derived such that the peak transient LHGR would not exceed fuel mechanical limits. The flow-dependent MAPLHGR and LHGR limits are generic, cycle-independent, and are specified in terms of multipliers, MAPFAC(F) and LHGRFAC(F), to be applied to the rated MAPLHGR and LHGR values, respectively.

The flow-dependent MAPLHGR and LHGR multipliers are shown in Figures 3-5 and 3-6. The MAPFAC(F) and LHGRFAC(F) limits also include consideration for peak clad temperature requirement per the LOCA analyses results (see Section 8.0).

3.3.5 Safety Limit Adjustment Procedure

The MCPR limits, provided in Figures 3-1 and 3-4 assume a dual-loop SLMCPR of 1.09. The off-rated MCPR(F) is defined by Figure 3-4. The off-rated MCPR(P) is defined by Figure 3-1. Only adjustment of the $P < P_{Bypass}$ portion of the MCPR(P) curve is required, because at $P \geq P_{Bypass}$, the K(P) applies the rated power OLMCPR adjustment to the MCPR(P). The limits should be adjusted by the following factor:

$$\left(\frac{\text{Cycle specific SLMCPR}}{1.09}\right)$$

3.3.6 Single Loop Operation Adjustment Procedure

When operating in Single Loop Operation (SLO) an adjustment is to be made to the rated power OLMCPR as well as the off-rated OLMCPR. The off-rated MCPR(F) is defined by Figure 3-4. The off-rated MCPR(P) is defined by Figure 3-1. Only adjustment of the $P < P_{Bypass}$ portion of the MCPR(P) curve is required because, at $P \geq P_{Bypass}$, the K(P) applies the rated power OLMCPR adjustment to the MCPR(P).

The equation for the adjustment is as follows when operating in SLO:

$$\text{SLO OLMCPR} = \text{OLMCPR}_{\text{dual-loop}} + \text{SLMCPR}_{\text{SLO}} - \text{SLMCPR}_{\text{dual-loop}}$$

3.4 Rod Withdrawal Error Analysis

The evaluation of the RWE event was performed without taking credit for the mitigating effect of the flow-biased RBM setpoints. The results of two analyses for JAF Cycle 16 at 60% power / 105% flow and 29% power / 105% flow are summarized in Table 3-13 and provide the OLMCPR results at off-rated conditions for the unblocked RWE event.

The RWE analysis is performed without any credit for the RBM. This allows relaxation of the flow dependent RBM setpoints in conjunction with MELLLA such that operation at high rod lines is not limited. Off-rated analyses show that the K(P) protects the off-rated RWE.

3.5 CONCLUSION

At any given power/flow state (P,F), all six limits should be determined: MCPR(P), MCPR(F), MAPFAC(P), LHGRFAC(P), MAPFAC(F), and LHGRFAC(P) (Figures 3-1, 3-4, 3-2, 3-3, 3-5, and 3-6, respectively). The most limiting MCPR and the most limiting MAPLHGR and LHGR [maximum of MCPR(P) and MCPR(F) and minimum of MAPLHGR(P) or MAPLHGR(F) and LHGR(P) or LHGR(F)] will be the governing limits. Different rated condition OLMCPRs have no effect on these limit curves. The rated OLMCPR is multiplied by the cycle-independent K(P) function to determine off-rated power-dependent MCPR requirements, MCPR(P), at power levels above P_{Bypass}. The overall MCPR limit is the greater of MCPR(P) and MCPR(F) for a given (P,F) operating condition. The rated OLMCPRs are determined by the cycle-specific fuel reload analyses.

The results of the evaluations in the above subsections should be utilized to determine the OLMCPRs for JAF's operation in the expanded power/flow map.

In order for JAF to operate with TBVOOS , the K(P) above P-bypass can be applied to the TBVOOS events by multiplying the rated OLMCPRs for this condition by the appropriate K(P). To support extended operation at JAF with TBVOOS, OLMCPR limits below P-bypass would be required and these limits would be more restrictive than those contained in this report.

The conclusions will be revalidated on a cycle-specific basis as part of the reload licensing scope.

MCPR Limits

Power Dependent MCPR(P) Operating Limits Above P-bypass :

K(P) = Calculated Operating Limit (P) / Rated OLMCPR

OLMCPR (P) = K(P) * OLMCPR (100)

Power	K(P)
100	1.000
60	1.150
45	1.280
29	1.494

Power Dependent MCPR(P) Operating Limits Below P-bypass :

Core Flow ≤ 60%		Core Flow > 60%	
Power	OLMCPR(P)	Power	OLMCPR(P)
29	2.27	29	2.38
25	2.27	25	2.43

Flow Dependent MCPR(F) Operating Limits :

MCPR(F) = Max(1.22,SL/1.07x(Af x Wc /100 + Bf))

Where Af and Bf are dependent on the Maximum Flow Line Setpoint

Wc = % of Rated Core Flow

MAX Flow	Af	Bf
102.5%	-0.571	1.655
107%	-0.586	1.697
112%	-0.602	1.747
117%	-0.632	1.809

MAPLHGR and LHGR Limits:

Both Sets of Limits, (MAPLHGR and LHGR) are calculated the same way.

TOP (MAPFAC or LHGRFAC)= Max (1;(100 + TOP limit)) / (100 + TOP calculated)

MOP (MAPFAC or LHGRFAC) = Max (1;(MOP limit)) / MOP calculated

Power Dependent MAPLHGR and LHGR Limits – Above P-bypass:

LHGR(P) = LHGRFAC(P) * LHGR(rated)

MAPLHGR(P) = MAPFAC(P) * MPALHGR (rated)

Power	MAPFAC(P)/ LHGRFAC(P)
100	1.000
29	0.629

Power Dependent MAPLHGR and LHGR Limits – Below P-Bypass:

Core Flow $\leq$ 60%		Core Flow > 60%	
Power	MAPFAC(P)/ LHGRFAC(P)	Power	MAPFAC(P)/ LHGRFAC(P)
29	0.58	29	0.55
25	0.58	25	0.55

Flow Dependent MAPLHGR and LHGR Limits:

LHGR(F) = LHGRFAC(F) * LHGRrated

MAPLHGR(F) = MAPFAC(F) * MAPLHGRrated

LHGRFAC(F) or MAPFAC(F) = Min (1.00, (Af x Wc /100 + Bf))

Wc = % of Rated Core Flow

MAX Flow	Af	Bf
102.5	0.6784	0.4861
107%	0.6758	0.4574
112%	0.6807	0.4214
117%	0.6886	0.3828

Table 3-1a
Base Conditions for ARTS/MEOD Rated Transient Analyses

	Normal	105% ICF	81% MEOD
Power (MWt / % of CLTP)	2536 / 100	2536 / 100	2536 / 100
Flow (Mlb/hr / % rated)	77.0 / 100	80.85 / 105	62.37 / 81
Steam Flow (Mlb/hr)	10.98 / 9.91*	10.98 / 9.91*	10.96 / 9.89*
FW Temperature (°F)	423.9 / 344*	423.9 / 344*	423.9 / 344*
Core Inlet Enthalpy (Btu/lb)	531.1 / 520.8*	532.1 / 522.2*	525.9 / 513.6*
Dome Pressure (psig)	1040 / 1028*	1040 / 1028*	1040 / 1028*

* Values for reduced FW temperature at 344°F.

Table 3-1b
Base Conditions for ARTS/MEOD Off-Rated Transient Analyses

	80P / 105F	60P / 105F	45P / 105F	29P / 105F	29P / 60F	25P / 105F	25P / 60F
Power (MWt)	2028.8	1521.6	1141.2	735.4	735.4	634	634
Flow (Mlb/hr)	80.85	80.85	80.85	80.85	46.2	80.85	46.2
Steam Flow (Mlb/hr)	8.500 / 7.768	6.143 / 5.688	4.461 / 4.177	2.757 / 2.617	2.727 / 2.590	2.347 / 2.237	2.318 / 2.210
FW Temperature (°F) *	399.9 / 326.8	371.3 / 306.0	344.7 / 286.5	307.4 / 259.0	306.8 / 258.6	295.6 / 250.2	294.9 / 249.7
Core Inlet Enthalpy (Btu/lb) *	531.0 / 524.1	531.0 / 526.7	532.0 / 529.2	533.9 / 532.6	526.8 / 524.7	534.6 / 533.5	528.2 / 526.5
Dome Pressure (psig) *	1017 / 1009	998 / 993	986 / 983	976 / 974	976 / 974	973 / 972	974 / 972

* Values shown are for normal / reduced FW temperatures (424°F / 344°F at rated power and flow, respectively).

Table 3-2
MEOD Transient Analyses Peak Values, Cycle 16

Initial Power/Flow (% Rated) Transient	Peak Neutron Flux (% Initial)	Peak Heat Flux (% Initial)	Thermal Over-Power (%) (TOP) GE12 / GE14	ΔCPR[b] GE12 / GE14	OLMCPR[c] GE12 / GE14	Peak Steam Line Pressure (psig)	Peak Vessel Pressure (psig)
100/100 - RWE				0.26/0.26	1.35/1.35		
100/105-EOC							
TTNBP	410	120	32/32	0.27/0.29	1.38/1.43	1240	1274
LRNBP	426	121	33/34	0.28/0.29	1.38/1.44	1239	1274
FWCF	376	123	32/32	0.27/0.29	1.38/1.44	1217	1260
FWCF [a]	375	125	34/32	0.27/0.28	1.38/1.43	1199	1235
100/81- EOC							
TTNBP	473	124	28/31	0.26/0.25	1.36/1.40	1253	1276
LRNBP	456	125	29/32	0.26/0.26	1.37/1.40	1250	1274
FWCF	473	128	29/33	0.25/0.25	1.36/1.39	1227	1256
FWCF [a]	460	129	30/34	0.25/0.25	1.36/1.39	1205	1243

Notes:

(a) Evaluated with reduced FW temperature.

(b) ΔCPR calculated, uncorrected.

(c) OLMCPR Option B.

(d) EOC = End-of-cycle

Table 3-3
ARTS Transient Analysis Results – Generic K(P) Confirmation
Above P-Bypass - Normal Feedwater Temperature

Initial Power/Flow (% Rated)	Transient	ΔCPR [(a)] GE12 / GE14	OLMCPR [(b)] GE12 / GE14 MOC/EOC	OLMCPR [(c)] GE12 / GE14 MOC/EOC	Limiting Calculated K(P) [(d)]	Generic K(P)
100/105	LRNBP	0.28/0.29	1.46/1.61	1.38/1.44		
	TTNBP	0.27/0.29	1.46/1.60	1.38/1.43		
	FWCF	0.27/0.29	1.46/1.61	1.38/1.44	1.0	1.0
80/105	LRNBP	0.27/0.28	1.46/1.60	1.38/1.43		
	TTNBP	0.27/0.28	1.46/1.60	1.38/1.43		
	FWCF	0.29/0.30	1.48/1.62	1.40/1.45	1.027	1.075
60/105	LRNBP	0.26/0.27	1.40/1.48	1.35/1.37		
	TTNBP	0.25/0.26	1.39/1.48	1.34/1.37		
	FWCF	0.31/0.32	1.45/1.54	1.40/1.43	1.026	1.15
45/105	LRNBP	0.22/0.22	1.36/1.44	1.31/1.33		
	TTNBP	0.21/0.21	1.35/1.43	1.30/1.32		
	FWCF	0.33/0.34	1.47/1.56	1.42/1.45	1.041	1.28
40/105	LRNBP/PLU	0.60/0.60	1.93/1.94	1.87/1.89	1.369	1.39[e]
29/105	LRNBP/PLU	0.72/0.73	2.11/2.12	2.07/2.08	1.512	1.52[e]
	TTNBP	0.12/0.13	1.26/1.34	1.21/1.23		
	FWCF	0.39/0.40	1.53/1.62	1.48/1.51	1.087	1.49

Notes:

(a) ΔCPR based on initial CPR which yields MCPR = 1.09, uncorrected for Options A and B.

(b) OLMCPR for Option A.

(c) OLMCPR for Option B

(d) The calculated K(P) considers the maximum OLMCPR calculated for any transient in that category divided by the operating limit for that category including exposure dependence. [[ ]]

(e) PLU limits

Table 3-4
ARTS Transient Analysis Results – Generic K(P) Confirmation Above P-Bypass - Reduced Feedwater Temperature

Power / Flow	Transient	ΔCPR [a] GE12/GE14 EOC	OLMCPR [b] GE12/GE14 EOC	OLMCPR [c] GE12/GE14 EOC	Limiting Calculated K(P) [d]	Generic K(P)
100 / 105	FWCF	0.27/0.28	1.46/1.60	1.38/1.43	1.0	1.0
80 / 105	FWCF	0.29/0.30	1.48/1.62	1.40/1.45	1.027	1.075
60 / 105	FWCF	0.32/0.33	1.46/1.55	1.41/1.44	1.033	1.15
45 / 105	FWCF	0.35/0.36	1.49/1.58	1.44/1.47	1.056	1.28
29 / 105	FWCF	0.42/0.43	1.56/1.65	1.51/1.54	1.109	1.49

Notes:

(a) ΔCPR based on initial CPR which yields MCPR = 1.09, uncorrected for Options A and B.

(b) OLMCPR for Option A.

(c) OLMCPR for Option B

(d) The calculated K(P) considers the maximum OLMCPR calculated for any transient in that category divided by the operating limit for that category including exposure dependence. [[ ]]

Table 3-5
ARTS Transient Analysis Results – MCPR(P)
Below P-Bypass - Normal Feedwater Temperature

Initial Power/ Flow (%Rated)	Transient [a]	ΔCPR [c] GE12/GE14 EOC	OLMCPR Option A [d] GE12/GE14 EOC	Calculated MCPR(P) [e]	Limiting MCPR(P)
29/105	LRNBP	0.73/0.74	2.01/2.03		
29/105	TTNBP	0.73/0.74	2.01/2.03		
29/105	FWCF	0.85/0.86	2.19/2.20	2.30/2.31	2.38
29/60	LRNBP	0.58/0.58	1.81/1.81		
29/60	TTNBP	0.58/0.58	1.81/1.81		
29/60	FWCF	0.78/0.78	2.09/2.10	2.18/2.18	2.27
25/105	LRNBP	0.77/0.78	2.07/2.09		
25/105	TTNBP	0.79/0.81	2.10/2.12		
25/105	FWCF	0.85/0.87	2.19/2.21	2.30/2.32	2.43
25/60	LRNBP	0.61/0.60	1.85/1.84		
25/60	TTNBP	0.61/0.61	1.85/1.84		
25/60	FWCF	0.77/0.77	2.07/2.07	2.17/2.17	2.27

Notes:

(a) A pressure scram occurred for the LR and TT transients. For the FWCF events, the vessel pressure did not reach the pressure scram setpoint, therefore, no scram occured.

(b) ΔCPR based on initial CPR which yields MCPR = 1.09, uncorrected for Option A.

(c) Option A OLMCPR = 1.09 * (1.0 - Δ/I (95/95)).

(d) [[ ]]

Table 3-6
ARTS Transient Analysis Results – MCPR(P)
Below P-Bypass - Reduced Feedwater Temperature

Initial Power / Flow (%Rated)	Transient [(a)]	ΔCPR [(b)]	OLMCPR Option A[(c)] GE14 EOC	Calculated MCPR(P) [(d)]	Limiting MCPR(P)
29/105	FWCF	0.89/0.90	2.24/2.26	2.35/2.38	2.38
29/60	FWCF	0.81/0.81	2.13/2.13	2.24/2.24	2.24
25/105	FWCF	0.91/0.93	2.28/2.31	2.40/2.43	2.43
25/60	FWCF	0.83/0.83	2.16/2.16	2.27/2.26	2.27

Notes:

(a) The vessel pressure did not reach the pressure scram setpoint, therefore, no scram occured.

(b) ΔCPR based on initial CPR which yields MCPR = 1.09, uncorrected for Option A.

(c) Option A OLMCPR = 1.09 * (1.0 - Δ/I (95/95)).

(d) [[ ]]

Table 3-7
Summary of K(P) and MCPR(P) Limits for Different Operational Strategies

Power (%)	K(P)	MCPR(P)[c]
100[a]	1.0	-
85[a]	1.056	-
65[a]	1.13	-
60[a]	1.15	-
45[a]	1.28	-
40[a]	1.39	-
29[a]	1.52	-
29/60[b]	-	2.27
25.0/60[b]	-	2.27
29/105[b]	-	2.38
25/105[b]	-	2.43

Notes:

(a) Above P-bypass

(b) Below P-bypass

(c) Limits for Turbine Bypass In Service with or without Reduced FW Temperature and one MSIV-OOS based on SLMCPR = 1.09.

Table 3-8
ARTS Transient Analysis Results -- Generic MAPFAC(P) and LHGRFAC(P) Confirmation Above P-Bypass – Normal Feedwater Temperature

Power / Flow	Transient	GE12/GE14 TOP EOC (a)	GE12/GE14 MOP EOC (a)	Calculated MAPFAC(P) LHGRFAC(P) TOP/MOP GE12 (b)	Calculated MAPFAC(P) LHGRFAC(P) TOP/MOP GE14 (b)	Generic MAPFAC(P) LHGRFAC(P)
100/105	LRNBP	36/38	36/38	1.0/1.0	1.0/1.0	1.0
	TTNBP	35/37	35/37	1.0/1.0	1.0/1.0	
	FWCF	36/39	36/39	1.0/1.0	1.0/1.0	
80/105	LRNBP	30/31	31/31	0.994/1.0	1.0/1.0	0.895
	TTNBP	30/30	31/30	0.996/1.0	1.0/1.0	
	FWCF	33/32	33/32	0.970/1.0	1.0/1.0	
60/105	LRNBP	25/27	27/27	1.0/1.0	1.0/1.0	0.790
	TTNBP	24/26	26/26	1.0/1.0	1.0/1.0	
	FWCF	36/33	37/34	0.948/1.0	0.994/1.0	
45/105	LRNBP	21/21	22/21	1.0/1.0	1.0/1.0	0.700
	TTNBP	19/20	21/20	1.0/1.0	1.0/1.0	
	FWCF	38/34	39/36	0.934/1.0	0.989/1.0	
40/105	LRNBP/PLU	70/68	72/71	0.722/1.0	0.693/1.0	0.687
29/105	LRNBP/PLU	85/81	86/84	0.783/1.0	0.743/1.0	0.625
	TTNBP	10/11	11/11	1.0/1.0	1.0/1.0	
	FWCF	48/44	49/44	0.871/0.876	0.918/0.974	

Notes:

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Table 3-9
ARTS Transient Analysis Results – Generic MAPFAC(P) and and LHGRFAC(P) Confirmation Above P-Bypass – Reduced Feedwater Temperature

Power / Flow	Transient	GE12/GE14 TOP EOC [(a)]	GE12/GE14 MOP EOC [(a)]	Calculated MAPFAC(P) LHGRFAC(P) TOP/MOP GE12 [(b)]	Calculated MAPFAC(P) LHGRFAC(P) TOP/MOP GE14 [(b)]	Generic MAPFAC(P) LHGRFAC(P)
100/105	FWCF	34/32	34/32	1.0/1.0	1.0/1.0	1.0
80/105	FWCF	36/33	36/33	0.949/1.0	0.995/1.0	0.895
60/105	FWCF	40/34	40/38	0.925/1.0	0.984/1.0	0.790
45/105	FWCF	45/39	45/42	0.893/0.947	0.952/1.0	0.700
29/105	FWCF	55/51	56/51	0.837/0.765	0.873/0.833	0.625

Notes:

(a) [[

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Table 3-10
ARTS Transient Analysis Results – MAPFAC(P) and LHGRFAC(P) Below P-Bypass – Normal Feedwater Temperature

Power / Flow	Transient	GE12/GE14 TOP EOC [a]	Calculated MAPFAC(P) LHGRFAC(P) GE12/GE14	Corrected MAPFAC(P) LHGRFAC(P) GE12/GE14 [b]	Limiting MAPFAC(P) LHGRFAC(P)
29/105	LRNBP	81/73	0.746/0.741	0.678/0.674	
	TTNBP	80/72	0.750/0.743	0.682/0.676	
	FWCF	115/93	0.629/0.664	0.571/0.603	0.571
29/60	LRNBP	52/41	0.890/0.908	0.809/0.826	
	TTNBP	52/41	0.890/0.908	0.809/0.825	
	FWCF	105/90	0.658/0.675	0.599/0.614	0.599
25/105	LRNBP	82/80	0.744/0.710	0.676/0.645	
	TTNBP	85/83	0.731/0.699	0.665/0.635	
	FWCF	111/99	0.639/0.644	0.581/0.586	0.581
25/60	LRNBP	56/46	0.863/0.878	0.784/0.798	
	TTNBP	57/46	0.860/0.875	0.782/0.796	
	FWCF	100/85	0.675/0.693	0.613/0.630	0.613

Notes:

[[

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Table 3-11
ARTS Transient Analysis Results – MAPFAC(P) and LHGRFAC(P) Below P-Bypass – Reduced Feedwater Temperature

Power / Flow	Transient	GE12/GE14 TOP EOC [a]	Calculated MAPFAC(P) LHGRFAC(P) GE12/GE14	Corrected MAPFAC(P) LHGRFAC(P) GE12/GE14 [b]	Limiting MAPFAC(P) LHGRFAC(P)
29/105	FWCF	121/103	0.610/0.629	0.555/0.572	0.555
29/60	FWCF	110/96	0.642/0.654	0.583/0.595	0.583
25/105	FWCF	123/112	0.605/0.604	0.550/0.549	0.549
25/60	FWCF	110/94	0.643/0.660	0.585/0.600	0.585

Notes:

[[

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Table 3-12
Summary of MAPFAC(P) and LHGRFAC(P) for Different Operational Strategies

Power (%)	MAPFAC(P) LHGRFAC(P) Above P-Bypass	MAPFAC(P) LHGRFAC(P) Below P-Bypass [c]
100[a]	1.0	---
80[a]	0.895	---
60[a]	0.790	---
45[a]	0.710	---
29[a]	0.625	---
29/60[b]	---	0.580
25.0/60[b]	---	0.580
29/105[b]	---	0.550
25/105[b]	---	0.550

Notes:

(a) Above P-bypass

(b) Below P-bypass

(c) Limits for Turbine Bypass In Service with or without Reduced FW Temperature and one MSIV-OOS.

Table 3-13
Summary of Unblocked OLMCPR Values for the RWE Event

Power/Flow	OLMCPR Values
100/100	1.35
60/105	1.45
29/105	1.56

Figure 3-1
Power-Dependent MCPR Limits, K(P) and MCPR(P)
(Includes Feedwater Temperature Reduction and One MSIV Inoperable)

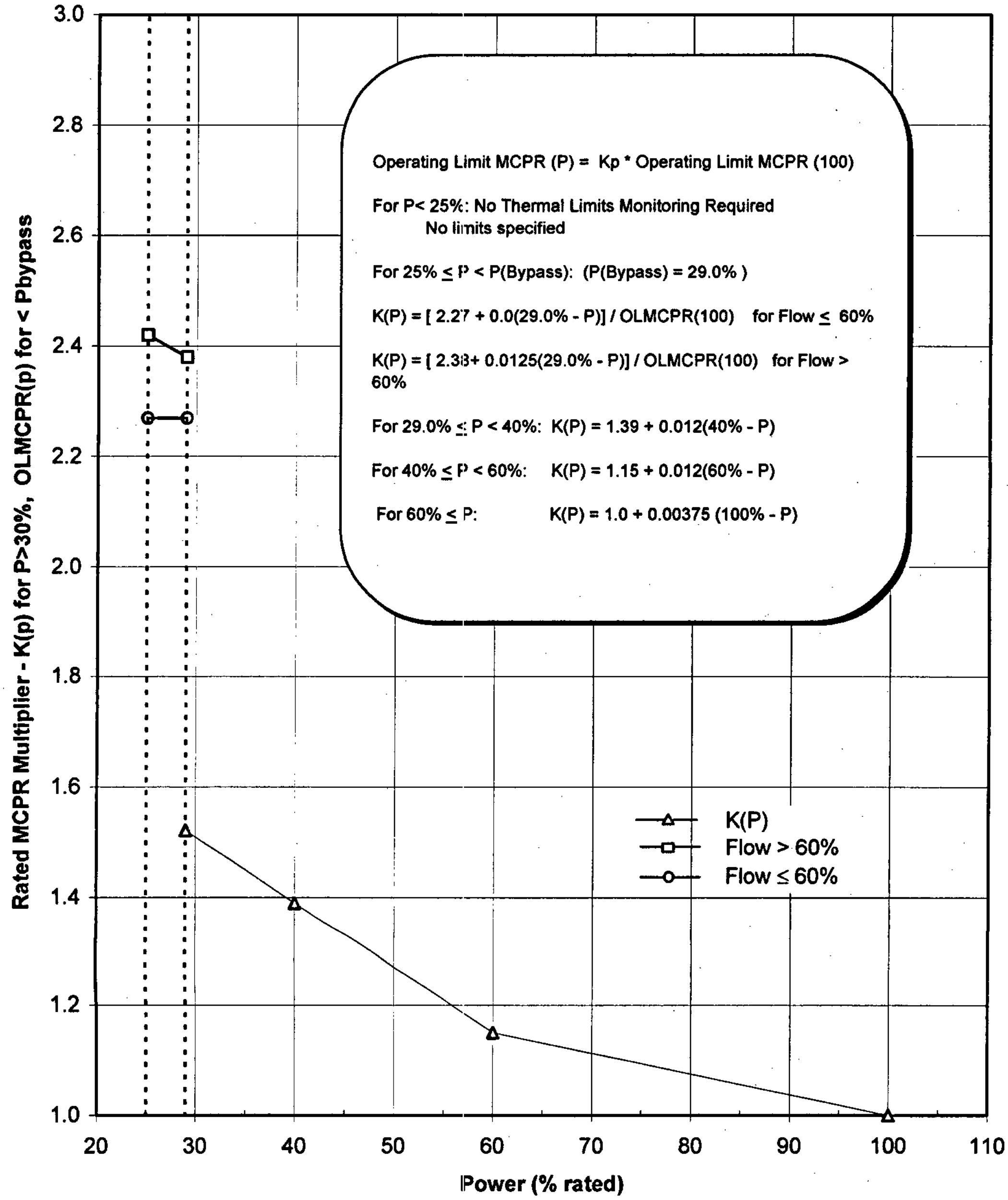


Figure 3-2
Power-Dependent MAPLHGR Multiplier, MAPFAC(P)
(Includes Feedwater Temperature Reduction and One MSIV Inoperable)

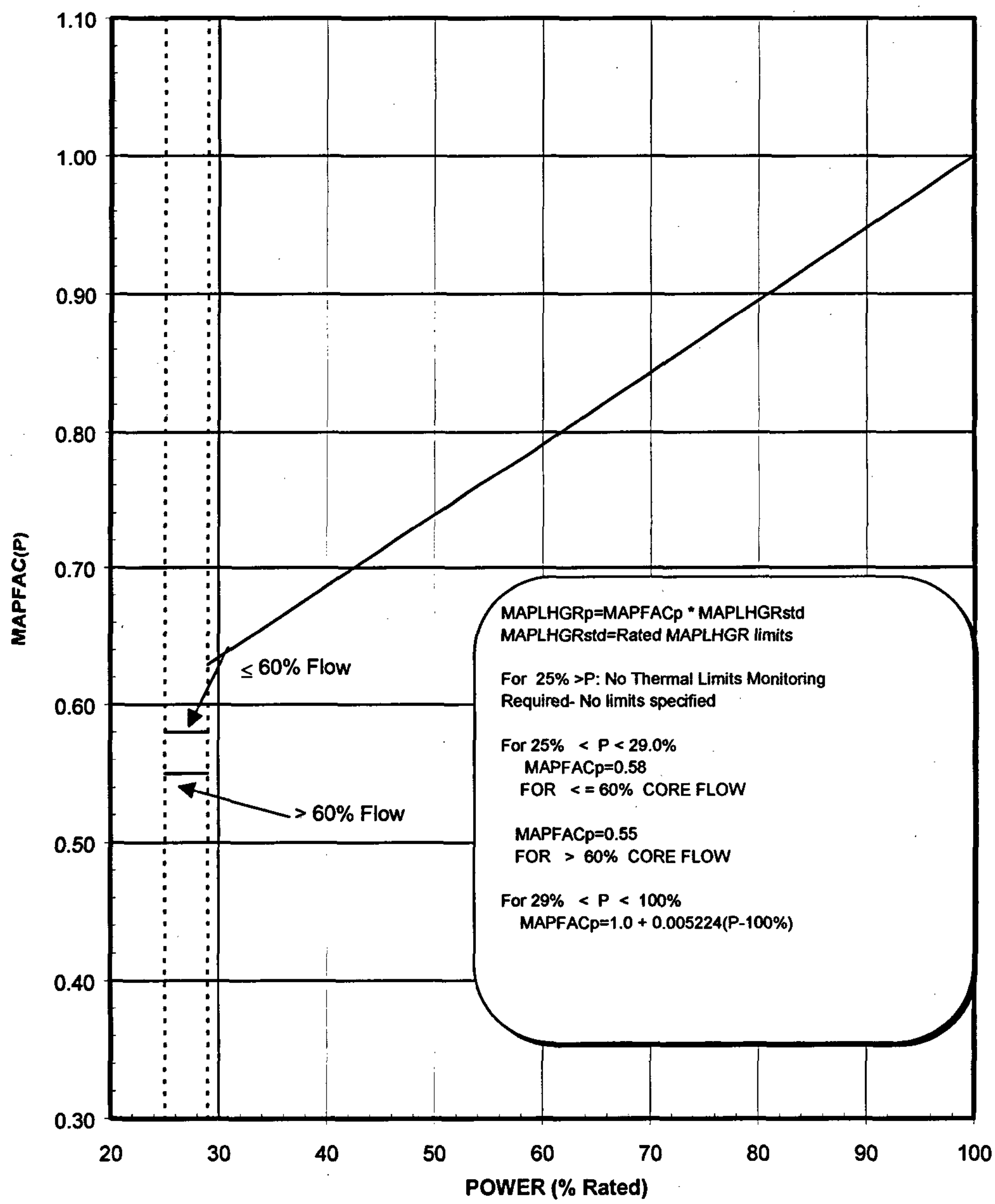


Figure 3-3
Power-Dependent LHGR Multiplier, LHGRFAC(P)
(Includes Feedwater Temperature Reduction and One MSIV Inoperable)

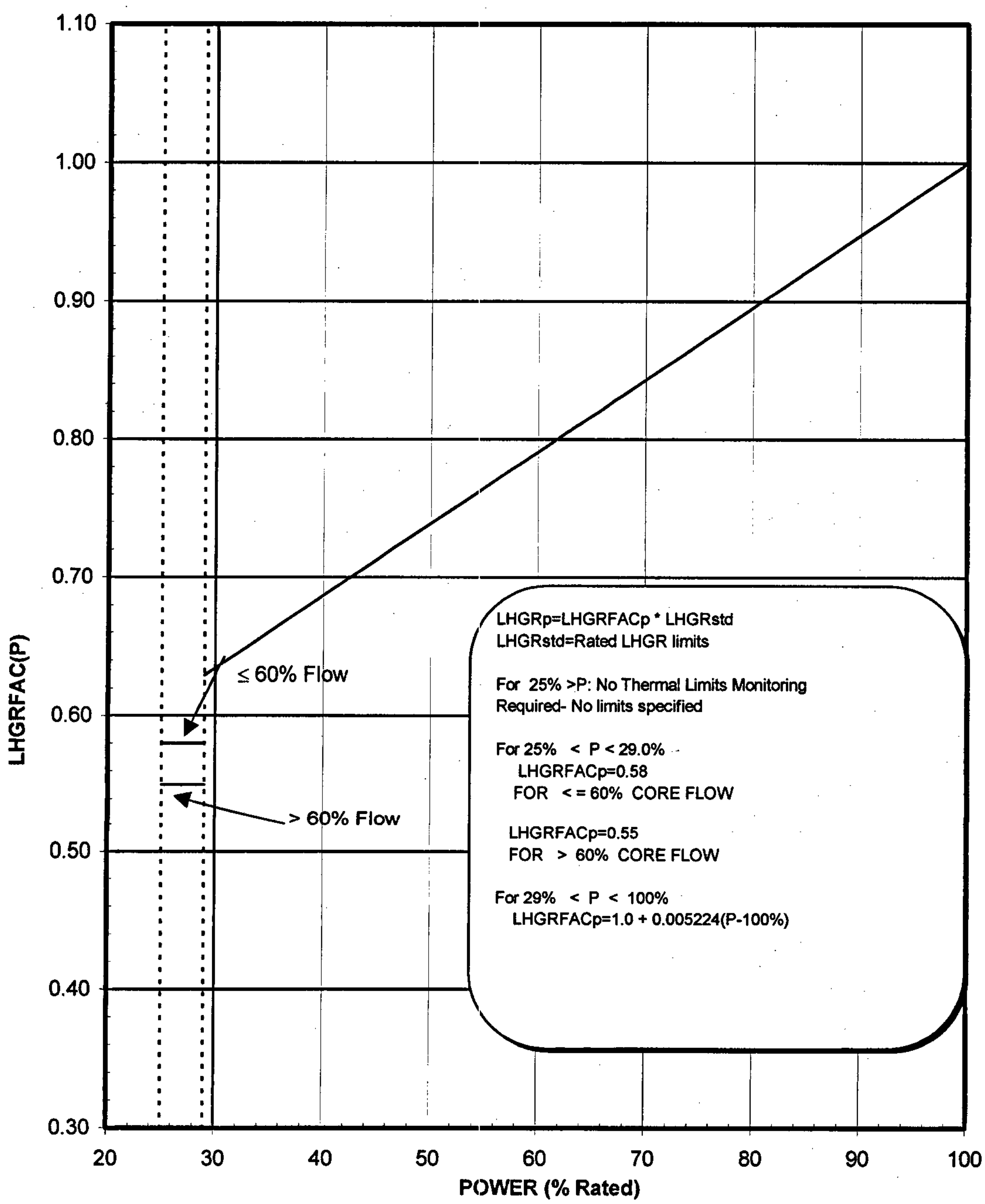


Figure 3-4
Flow-Dependent MCPR Limits, MCPR(F)

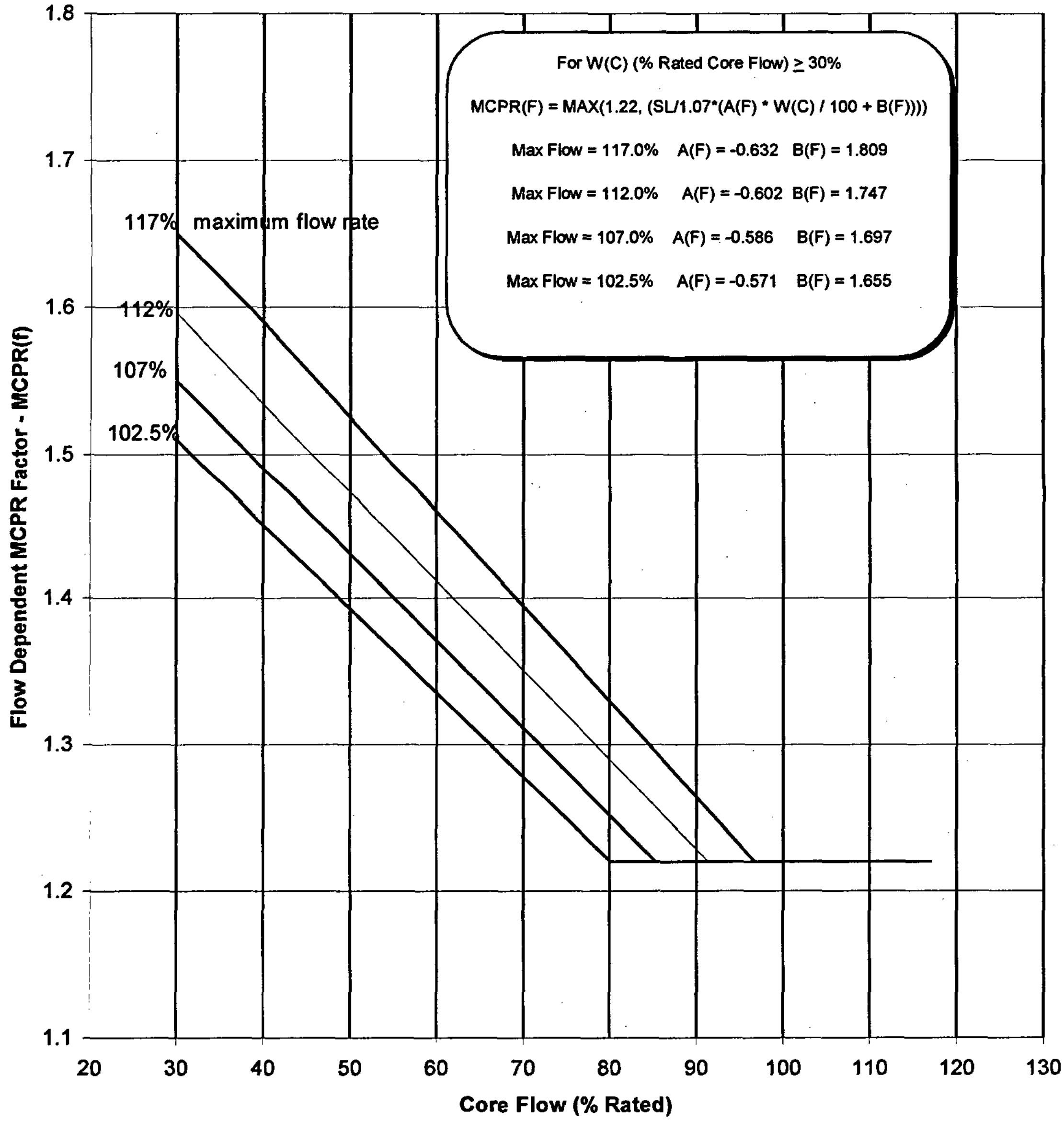


Figure 3-5
Flow-Dependent MAPLHGR Multiplier, MAPFAC(F)

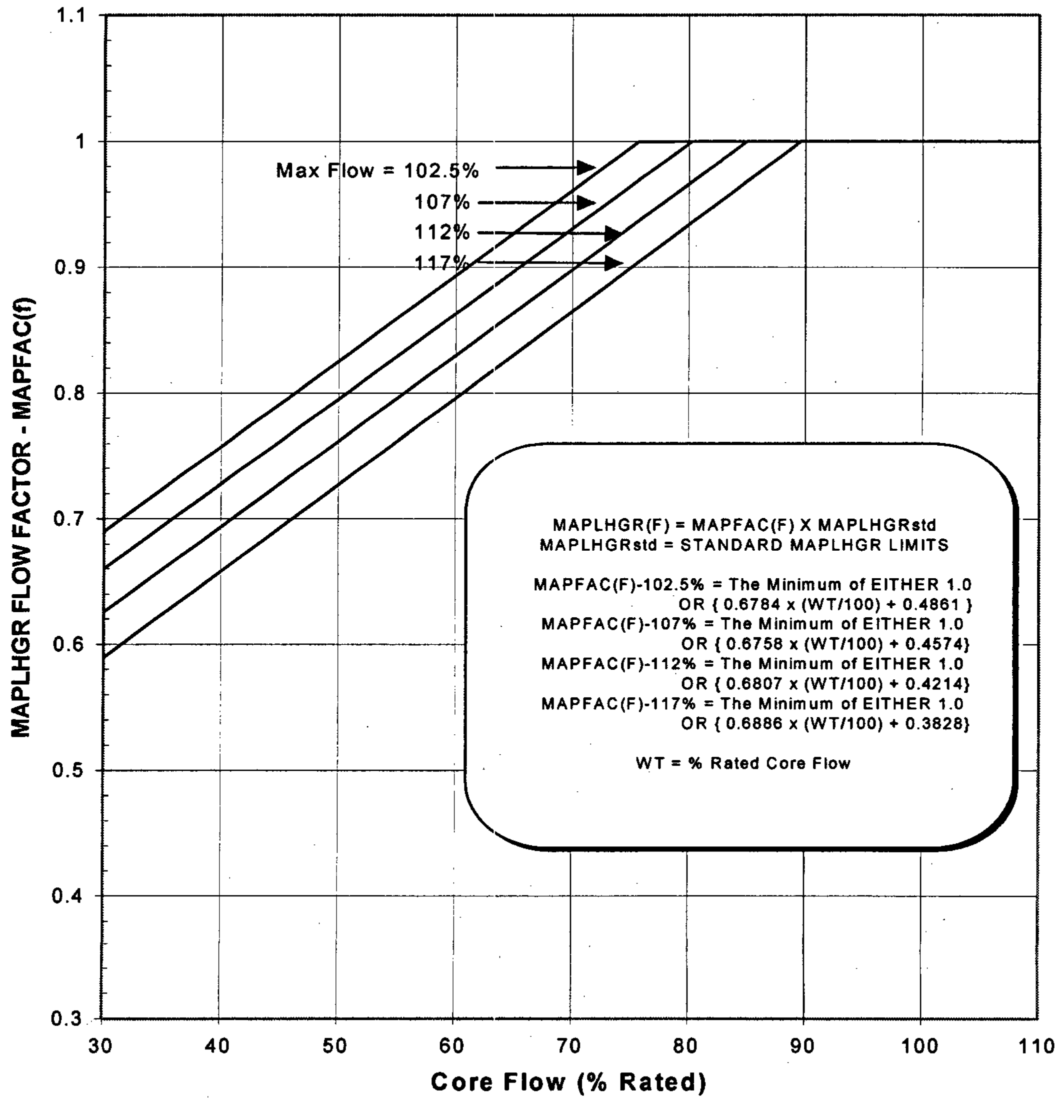


Figure 3-6
Flow-Dependent LHGR Multiplier,LHGRFAC(F)

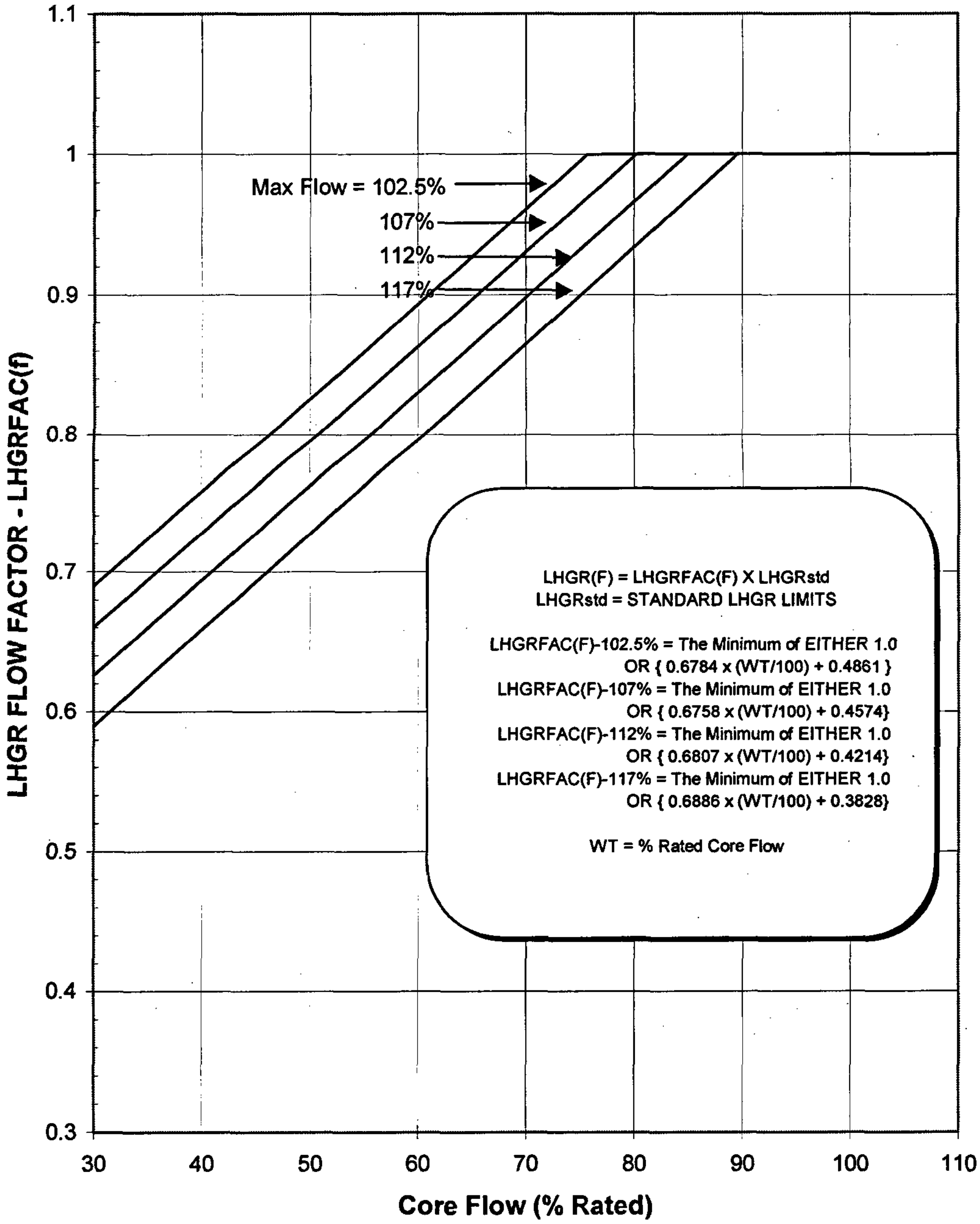


Figure 3-7
Plant Response to FW Controller Failure (BOC16 to EOC16 MELLLA)

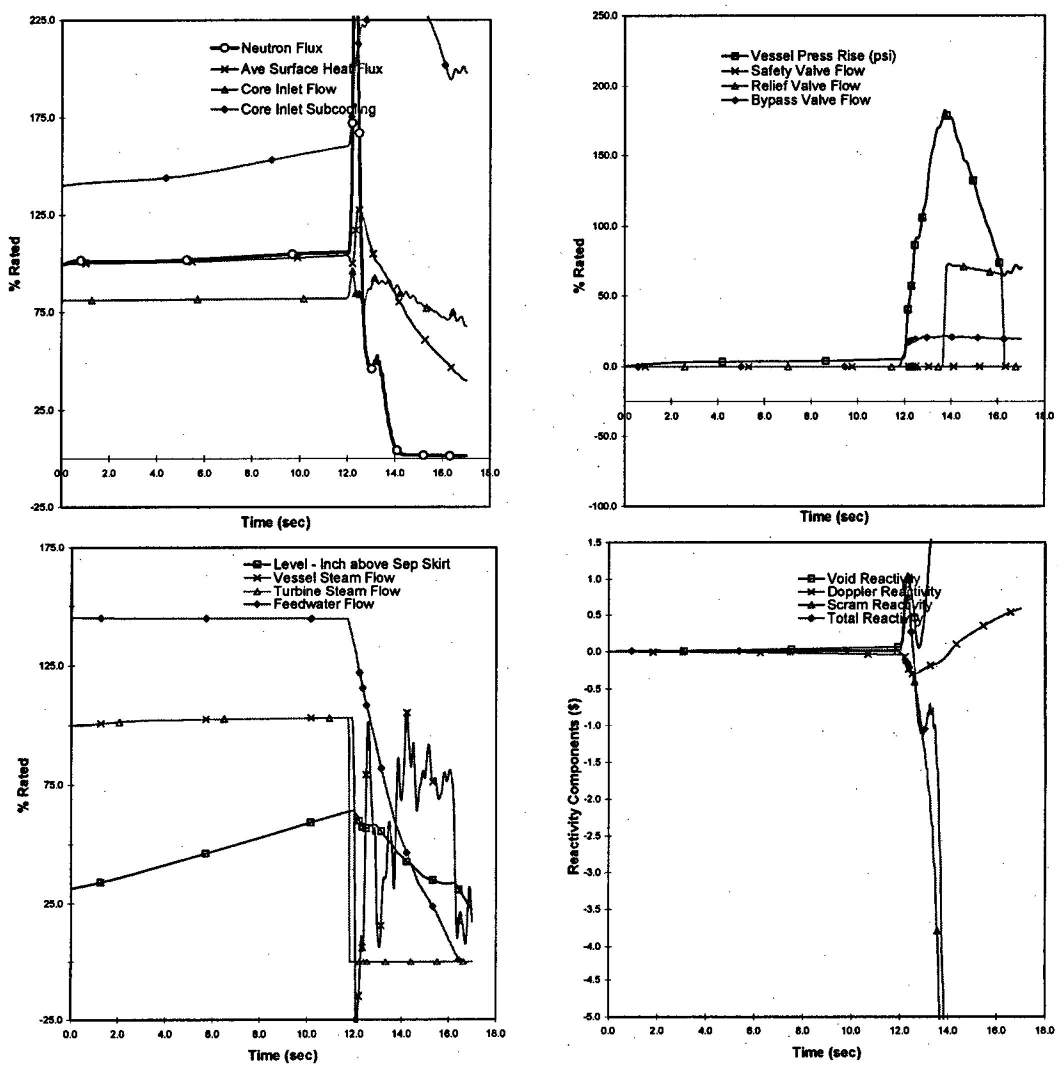


Figure 3-8
Plant Response to Load Reject w/o Bypass (BOC16 to EOC16 MELLLA)

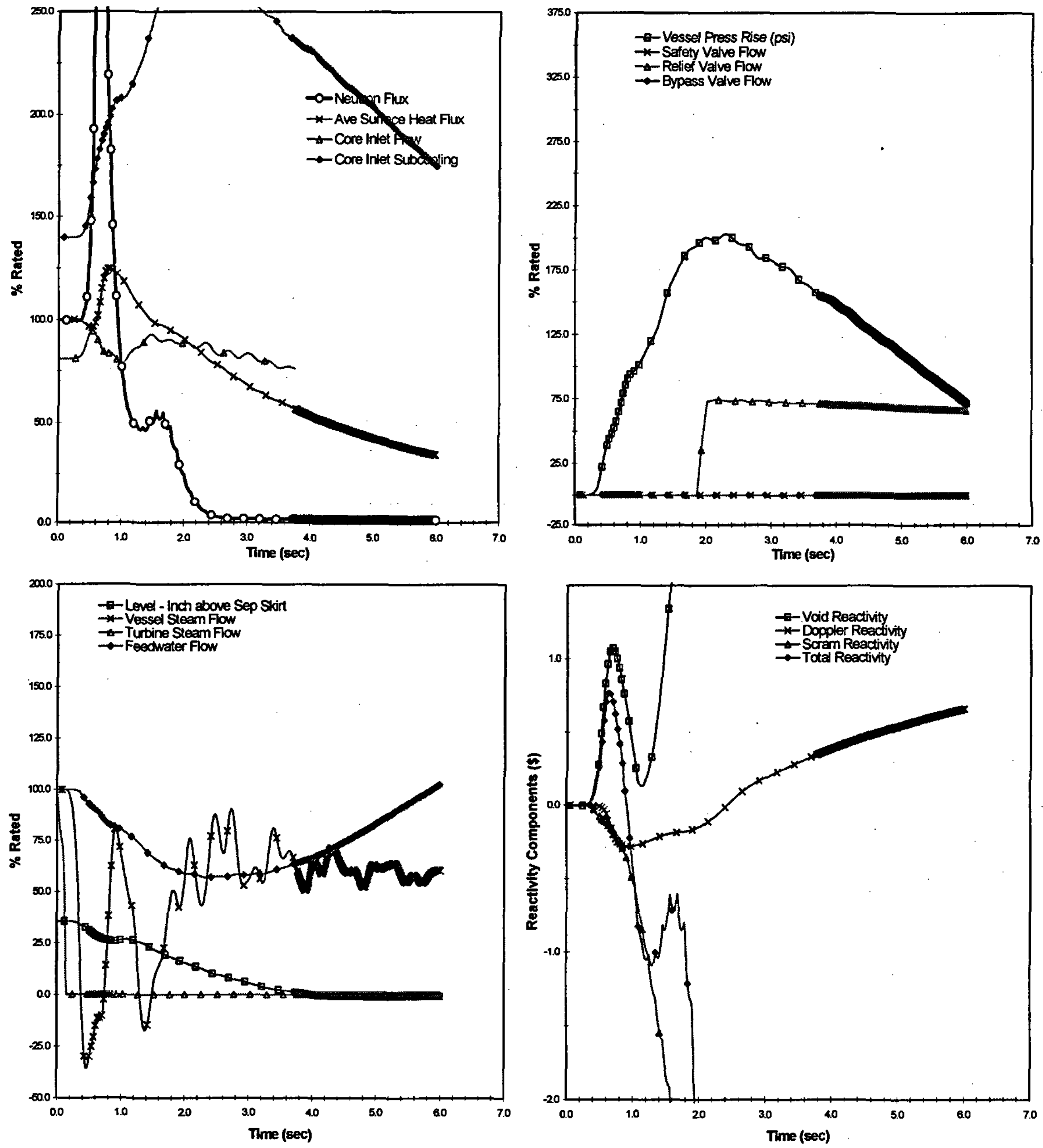


Figure 3-9
Plant Response to Turbine Trip w/o Bypass (BOC16 to EOC16 MELLLA)

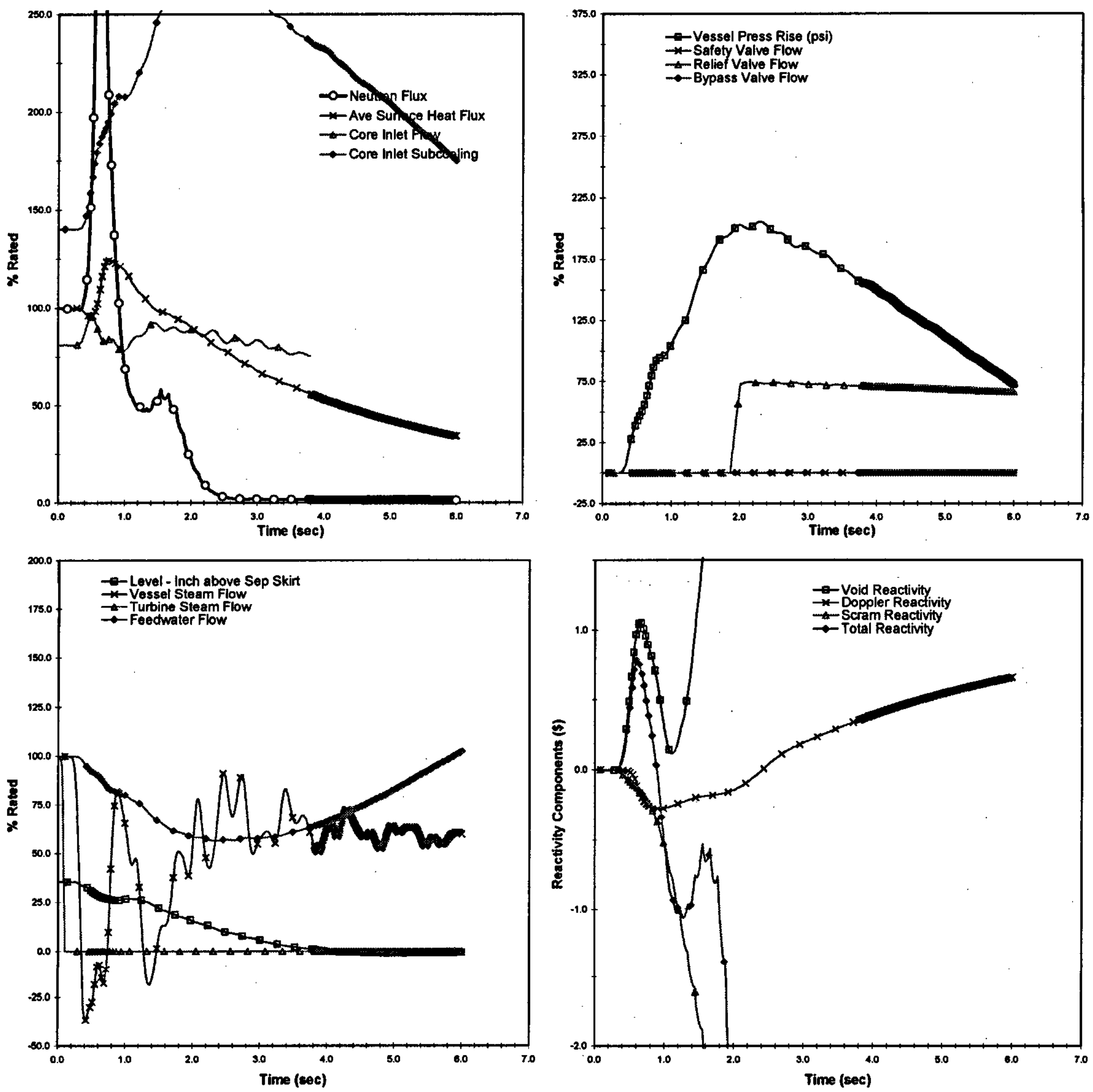
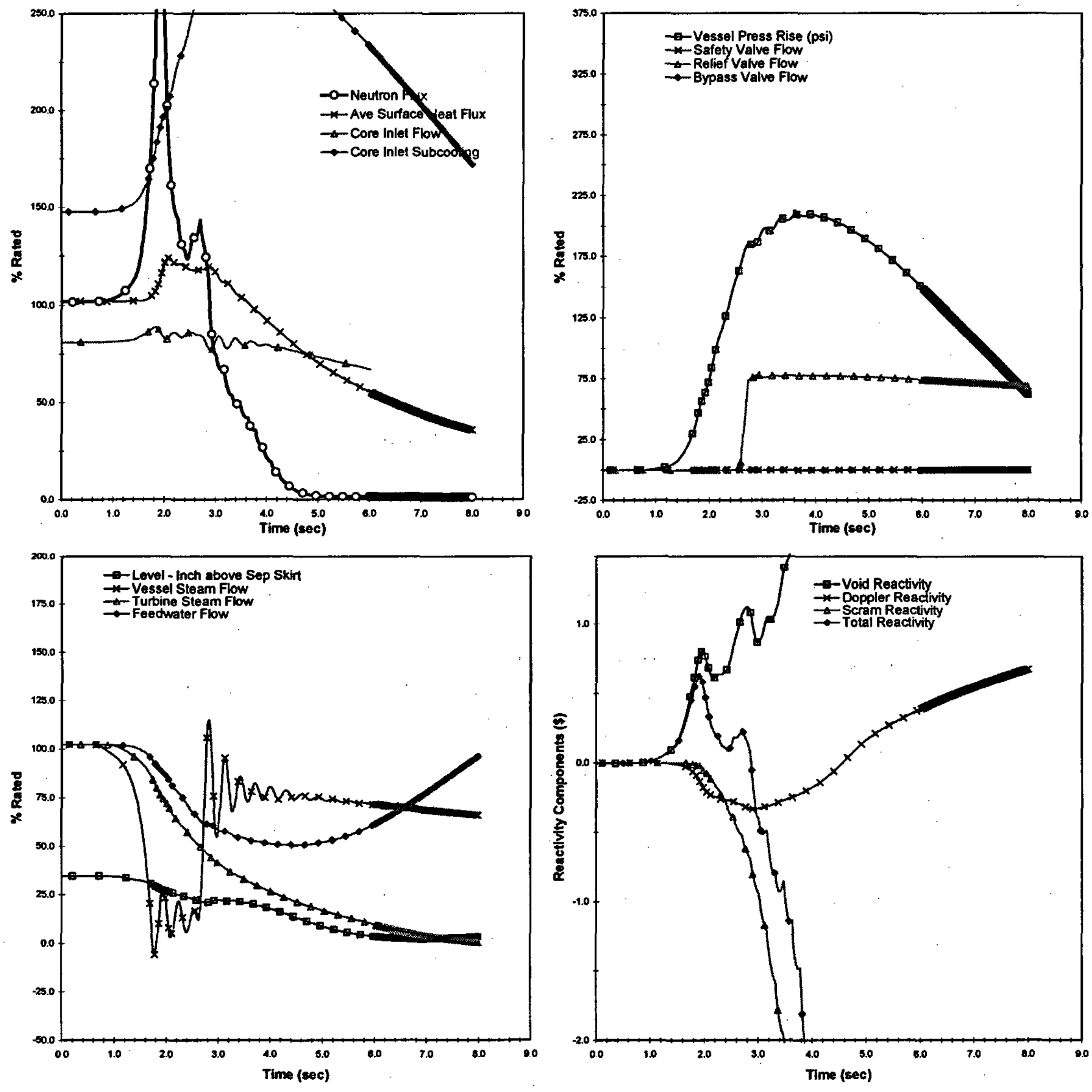


Figure 3-10
Plant Response to MSIV Closure (Flux Scram)



4.0 REACTOR RECIRCULATION SYSTEM

The Reactor Recirculation (RR) system was evaluated for ICF conditions. The major components of the RR system are the pumps, motors, suction and discharge valves, motor-generator sets (including drive motors, fluid couplers, and generators), and the jet pumps. The RR system evaluation included the suction and discharge pressure and temperature, pump speed, drive flow, and head requirements, pump motor current and power requirements, generator current and power requirements, and the drive motor current and power requirements. The effects of aging and degradation mechanisms (e.g., jet pump crudding) were not included in the evaluation.

The results of the evaluation indicate that the capability of the RR system to support operation at 105% of RCF may be marginal during some of the fuel cycle. If so, full 105% core flow may not be available until the end of the fuel cycle when the core differential pressure decreases, which causes the jet pump flow to increase for a given RR pump flow. Rotating equipment limitations are economic in nature and do not affect plant safety.

The RR pump net positive suction head (NPSH) requirements increase in the ICF region. Consequently, it is necessary to either increase the setpoint of the existing automatic cavitation protection interlock or ensure that plant procedures provide manual protection in the ICF region. Because current plant procedures already prohibit operation in the ICF region where there would be no automatic cavitation protection and changing the automatic cavitation interlock setpoint to a higher value would adversely affect plant maneuverability as represented by the power/flow map, no changes are recommended. There is no effect on plant safety.

SLO is not affected by ICF, because the SLO drive flow is limited to a value that does not exceed the value corresponding to 100% of RCF.

The recirculation pump mismatch Technical Specification limits do not change and the flow mismatch limits are not affected.

5.0 REACTOR COOLANT PRESSURE BOUNDARY

5.1 RCPB Piping Evaluation

The JAF reactor coolant pressure boundary (RCPB) piping system including associated branch piping inside containment was evaluated to determine its structural integrity under the MEOD operating conditions. The MEOD conditions primarily affect the pressures, temperatures, and flows for the following RCPB piping systems: RR system, reactor pressure vessel (RPV) Bottom Head Drain Line system, and their associated branch piping (inside containment). The piping system evaluation included the piping supports (e.g., hangers, snubbers, and rigid restraints) and the interfacing piping system components (e.g., RPV nozzles). For these affected piping systems, MEOD temperature, pressure, and flow parameters were compared with the existing piping analysis basis and/or current limiting values. Based on this comparison, the current pipe stress analysis results for the RR lines and the associated branch piping are based on higher temperatures and pressures than those for the MEOD conditions. The flow increase ($\simeq$ 8.1%) in the RR piping has no effect on the RR piping system including Residual Heat Removal (RHR) and Reactor Water Cleanup (RWCU) branch piping because there are no fast closing/opening valves in this system. No new high-energy line pipe break locations are postulated using existing pipe break criteria for RCPB piping systems due to the MEOD conditions. Therefore, the current pipe stress analyses results are adequate for the RR system and associated branch piping.

For the RPV Bottom Head Drain lines, the increase in pressure and temperature due to the MEOD conditions is negligible compared to the CLTP pressure and temperature. Therefore, the current pipe stress analyses results are adequate for the RPV Bottom Head Drain line and associated branch piping.

Therefore, the current piping system evaluations including piping supports and interfacing piping components are adequate for the MEOD conditions.

5.2 Recirculation System Piping Components

A flow-induced vibration evaluation has shown that the safety-related thermowells in the RR system piping are structurally adequate for the MEOD operating conditions. There are no safety-related sample probes in the RR system piping.

Because there is no change in the maximum main steam line flow and the maximum FW line flow for the MEOD operating conditions, there is no effect on the flow-induced vibration of the safety-related thermowells and sample probes in the main steam and FW lines.

6.0 VESSEL OVERPRESSURE PROTECTION

An overpressure analysis is a cycle-specific calculation performed at 102% of CLTP at the maximum core flow, which is unchanged for ARTS/MEOD. The typical sensitivity of operation at the MELLLA condition (81% flow) as compared to the Cycle 16 analysis at the 105% flow condition is provided in Table 6-1.

Table 6-1
Typical Sensitivity of Overpressure Analysis Results

Initial Power/Flow (% of rated)	Peak Steam Dome Pressure (psig)	Peak Steam Line Pressure (psig)	Peak Vessel Pressure (psig)
102 / 105	1306.3	1306.4	1341.8
102 / 81	1303.4	1304.2	1332.9

7.0 THERMAL-HYDRAULIC STABILITY

The stability compliance of GE fuel designs with regulatory requirements of the NRC is documented in Section 4.1 of Reference 2. The NRC approval of the stability performance of GE fuel designs also includes operation in the MEOD region of the power/flow map.

The above NRC acceptance of thermal-hydraulic stability includes the condition that the plant has systems and procedures in place, supported by Technical Specifications, as appropriate, which provide adequate instability protection.

7.1 Stability Option I-D

JAF has implemented stability long-term solution Option I-D. Option I-D is only applicable to plants which can demonstrate that core wide mode instability is the predominant mode and regional mode instability is not expected. Generally, a smaller core size produces higher eigenvalue separation between oscillation modes and tighter core inlet orifice coefficients make regional mode oscillations unlikely. Option I-D has: 1. "Prevention" elements (Exclusion and Buffer Regions), and 2. A "detect & suppress" element (SLMCPR protection provided by the flow-biased APRM flux trip for the dominant core wide mode of coupled thermal-hydraulic/neutronic reactor instability). Solution application consists of calculating an administratively controlled exclusion region (per Reference 4) and demonstrating that the existing flow-biased APRM flux trip line provides adequate SLMCPR protection (per Reference 5). The Option I-D exclusion region is core and fuel cycle dependent and represents a curved line of constant stability margin. The flow-biased APRM flux trip protection is also fuel cycle dependent.

The NRC-approved ODYSY methodology (Reference 6) was applied for the first time to the Cycle 16 stability analysis. ODYSY applications offer the benefit of more accurate simulations of BWR stability events and conditions. The MELLLA upper boundary line was assumed in the region boundary analysis for Cycle 16. Core and hot channel decay ratio calculations were performed to determine the Exclusion and Buffer Regions (See Figure 7-1). In addition, the ODYSY analysis also demonstrated that the core-wide mode is the predominant reactor instability mode for Cycle 16 MELLLA operation. The exclusion region demonstration is affected by operating conditions. The actual region calculation will be performed using the ODYSY code for future operating cycles.

The detect and suppress calculation consists of: 1. Calculation of a 95% probability / 95% confidence level statistically-based hot bundle oscillation magnitude for anticipated core-wide mode reactor instability, and 2. Calculation of the stability-based OLMCPR which provides 95/95 SLMCPR protection. The detect and suppress calculation requires the use of the DIVOM (which is defined as the **D**elta CPR over **I**nitial MCPR **V**ersus the **O**scillation **M**agnitude) curve. Recent TRACG evaluations have shown that the generic core-wide DIVOM curve specified in Reference 5, may not be conservative for current plant operating conditions for plants which have implemented Stability Option I-D. Specifically, a non-conservative deficiency has been identified for high power-to-flow ratios in the generic core-wide mode DIVOM curve. The deficiency results in a non-conservative slope of the associated core-wide DIVOM curve so that the APRM flux trip setpoint is too high. A Part 21 Notification was made on this issue (Reference 7). For Option I-D plants, the applicability of the core wide mode DIVOM curve may be determined by comparing the core average power-to-flow ratio following a simulated flow runback on the rated rod line to approximately 30% of RCF to a value of 66 MWt/Mlbm/hr. If the core average power-to-flow ratio exceeds this value, then the generic core wide mode DIVOM curve is not applicable and appropriate corrective actions should be taken. For JAF Cycle 16, the core average power-to-flow ratio is estimated to be 56.8 MWt/Mlbm/hr and the generic DIVOM slope is valid for Cycle 16 operation.

The SLMCPR protection demonstration is affected by operating conditions. The new APRM flow-biased flux scram line for the ARTS/MEOD operation was determined with the following additional conservatisms in the evaluation:

1. [[
   ]];

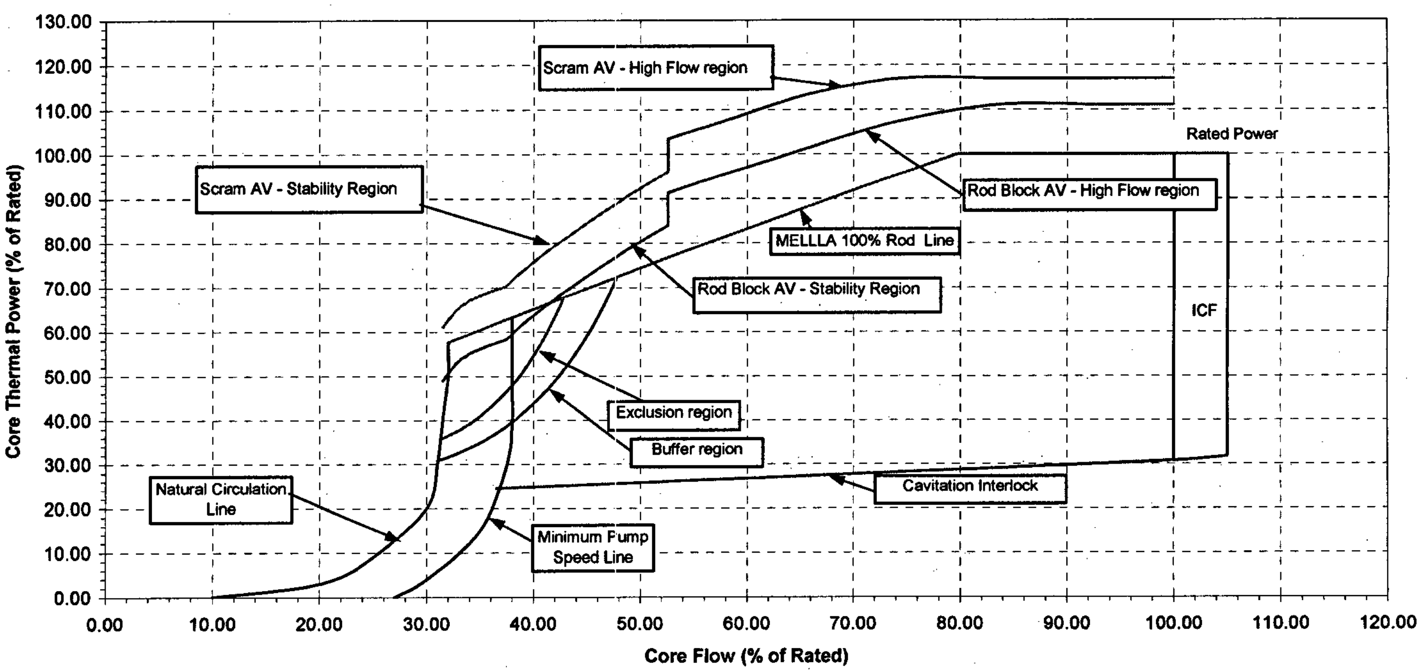
2. The SLMCPR is assumed to be 1.12, and the OLMCPR is assumed to be 1.35; and

3. [[
   ]]

The detect and suppress evaluation shows that adequate SLMCPR protection is provided by the new APRM flow-biased Flux scram line. The AL equation of the new APRM flow-biased flux scram line is: 1.5931 W_C + 13.42 (where W_C is core flow in % of rated). This AL equation, based on I-D stability analysis, can tolerate a 2% rated core power uncertainty up to ~52.6% of RCF. Moreover, this stability-based scram line provides more than 5% rated core power margin at the most limiting state point (i.e., the intersection of the Natural Circulation Line and the High Flow Control Line) to avoid a spurious scram due to a single pump trip. The basis for the more conservative scram power value is based on the margin to the MELLLA boundary. Consequently, MEOD operation is justified for plant operation with stability Option I-D.

The stability-based APRM flow-biased Flux scram line intercepts the MELLLA operating domain, non-stability based APRM flow-biased Flux scram line (defined in Section 1.2.2) at ~52.6% of RCF (~47% of recirculation drive flow). Therefore, it is conservative that the APRM trip system utilizes the non-stability based APRM flow-biased equation at recirculation drive flows greater than 47%, and up to 68.7%. The maximum AV clamps at 117% power for recirculation drive flows greater than 68.7%. The APRM flow-biased Flux scram lines are shown in Figure 7-1.

Figure 7-1
Option I-D APRM Flow-biased Flux Scram and Rod Block AVs



8.0 LOSS-OF-COOLANT ACCIDENT ANALYSIS

The current licensing basis SAFER/GESTR-LOCA analysis for JAF for GE12 and GE14 fuels have been reviewed to determine the effect on the ECCS performance resulting from plant operation in the MEOD domain. The MEOD region permits reactor operation at rated power over a wide range of core flows. The high core flow portion of this operating region (i.e., higher than rated core flow) is known as the ICF region. The low core flow portion of this operating region (i.e., lower than rated core flow) is known as the MELLLA region. The Reference 8 analysis for GE12 fuel considered JAF operation in the ELLLA domain. An analysis with GE14 fuel was performed to determine the effects on the LOCA analysis of operation in the MELLLA domain.

In the ECCS-LOCA analysis at ICF conditions, a slight delay in the onset of early boiling transition can occur for the axial nodes in the upper part of the bundle. This results in a lower calculated peak cladding temperature (PCT) for these nodes. However ICF does not affect the dryout time of the high powered node which determines the overall bundle PCT following a LOCA. Therefore the effect on the ECCS-LOCA results of ICF operation is negligible. Thus the PCTs for the limiting large break cases at rated conditions are applicable to the ICF condition.

The two major parameters that affect the bundle PCT in the design basis ECCS-LOCA calculation which are sensitive to the higher MELLLA load line in the operating power/flow map are the time of boiling transition (BT) at the high power node in the limiting fuel assembly and the core recovery time. Initiation of the postulated LOCA at MELLLA lower core flow conditions may result in earlier BT at the high power node, compared to RCF results, resulting in a higher calculated PCT. Similarly, initiation of the postulated LOCA at lower core flow affects break flow rates and core reflooding times, compared to rated core flow results, which can also result in a higher calculated PCT. These affects on the calculated PCT are acceptable as long as the results remain less than the licensing PCT limits.

An evaluation was performed with GE14 fuel to determine the ECCS-LOCA analysis effects of JAF operation in the MELLLA domain. The SAFER/GESTR-LOCA methodology was used consistent with Reference 8. The limiting Design Basis Accident (DBA) was evaluated to show the effect on the PCT based on limiting MELLLA conditions. The initial conditions for the JAF LOCA analysis used in this determination are listed in Table 8-1. For the limiting GE14 fuel type, the key MELLLA results with both nominal and Appendix K analysis assumptions are presented in Table 8-2 along with the PCT results at rated conditions. These results show that operation in the MELLLA region affects the nominal PCT by + 3°F and the Appendix K PCT by + 93°F.

The Upper Bound PCT is most directly related to changes in the nominal PCT and the Licensing Basis PCT is most directly related to changes in the Appendix K PCT. Therefore, the results in Table 8-2 were used to estimate the effect on the PCT licensing results of operation in the

MELLLA region. The Table 8-2 results for the nominal case show that the PCT is not significantly affected (i.e., + 3°F) by MELLLA. Therefore, the JAF Upper Bound PCT is similarly insensitive to operation in the MELLLA region. The Table 8-2 results for the Appendix K case show an increase of 93°F. Because the current JAF Licensing Basis PCT for GE14 fuel is 1700°F, there is still 500°F of margin to the 2200°F licensing limit. Also because GE14 is the limiting fuel type in the JAF core, this ΔPCT can be conservatively applied to the JAF GE12 results. The current JAF Licensing Basis PCT for GE12 fuel is 1370°F with a 170°F adder for 10 CFR 50.46 reported errors applicable to the JAF ECCS-LOCA analysis. Therefore the increase in the Appendix K PCT due to operation in the MELLLA region still leaves greater than 500°F of PCT margin to the 2200°F licensing limit. Furthermore, as long as the PCT limits are met, the percent of core wide metal-water reaction and maximum local oxidation results are not limiting for jet pump plants such as JAF.

8.1 Conclusions

The evaluation of the sensitivity of the ECCS-LOCA analysis to operation in the MEOD domain shows that the ARTS/MEOD option meets all ECCS-LOCA acceptance criteria. Therefore, there are no ECCS-LOCA analysis related plant operating restrictions due to the implementation of the ARTS/MEOD option.

Table 8-1
DBA LOCA Initial Conditions for JAF ARTS/MEOD

Parameter	Nominal	Appendix K
Core Thermal Power (MWt / % of CLTP)	2536.0 / 100	2587.0 / 102
Vessel Steam Output (Mlbm/hr)	10.98	11.23
Rated Core Flow (Mlbm/hr)	77.0	77.0
Core Flow (% of 77 Mlbm/hr)	79.80	79.80
Vessel Steam Dome Pressure (psia)	1060	1060
Maximum Recirculation Suction Line Break (RSLB) Area (ft^2) [(a)]	4.17	4.17
Bottom Head Drain Line Flow Path Area (ft^2)	0.014	0.014

Notes:

(a) The LOCA DBA break area includes maximum RSLB area and bottom head drain flow path area.

Table 8-2
DBA LOCA Results Comparison for JAF ARTS/MEOD

Single Failure	Nominal GE14 [(a)] (°F)	Appendix K GE14 [(a)] (°F)
100%F [(b)]		
Battery	1056	1554
Low Pressure Coolant Injection (LPCI) IV	989	1356
79.80%F		
Battery	1059	1647

Notes:

(a) The effect on the ECCS-LOCA analysis PCT of operation in the MEOD domain for GE14 is conservatively applicable to GE12.

(b) The effect on the ECCS-LOCA analysis PCT for operation at core flows greater than 100% (ICF) is negligible. Thus the PCTs for the limiting large break cases at rated conditions are applicable to the ICF condition.

9.0 CONTAINMENT RESPONSE

9.1 Introduction

This section discusses the effect of MEOD (considering the effect of FFWTR) on the containment pressure and temperature response and on the containment LOCA hydrodynamic loads (pool swell, vent thrust, condensation oscillation, and chugging) for JAF. The analysis presented here demonstrates margin to the containment pressure and temperature design limits and confirms that the containment hydrodynamic loads currently defined for JAF are not exceeded.

9.1.1 Containment Pressure and Temperature Response

Short-Term Pressure and Temperature

To evaluate the effect of ARTS/MEOD on containment performance, analyses of short-term DBA-LOCA containment response were performed. The short-term containment response covers the blowdown period during which the maximum drywell pressure, drywell temperature, and maximum drywell to wetwell differential pressure occur. Consequently, analyses were performed for various cases that cover the full extent of JAF operation in MEOD. The objective of performing these analyses is to demonstrate that JAF operation in the MEOD region (including the effect of FFWTR) does not result in exceeding the containment pressure and temperature design limits. The results of these analyses are also used for evaluating the various containment hydrodynamic loads.

Long-Term Pressure and Temperature

The long-term pressure and temperature response is not affected by ARTS/MEOD operation or FFWTR. The peak wetwell pressure and temperature and peak suppression pool temperature occur later in the DBA-LOCA and are established by the long-term release of the decay heat and the sensible energy from the reactor vessel to the containment. Because ARTS/MEOD operation and FFWTR operation do not increase the reactor power level nor the vessel operating pressure, neither the decay heat nor the vessel sensible energy is increased. Thus, the DBA-LOCA peak wetwell pressure and temperature and suppression pool temperature, which occur in the long-term, are not impacted by the ARTS/MEOD or FFWTR operation. This also applies to all other long-term duration events.

Analysis Cases

The short-term containment pressure and temperature response for a DBA LOCA was analyzed for four cases. All analyzed cases were performed using the GE M3CPT containment code (References 9 and 10) using the break flow and break enthalpy inputs from analyses using the LAMB computer code (Reference 11). These cases are selected so as to conservatively cover the full extent of the MEOD power/flow boundary. The power and flow values for the four cases include.

- Case 1 which corresponds to 102% of CLTP and 100% of RCF.
- Case 2 which corresponds to 102% of CLTP, with 105% of RCF (i.e., ICF).
- Case 3 which corresponds to 102% of CLTP, with 79.8% of RCF (i.e., on the MELLLA line).
- Case 4 which corresponds to 62% of CLTP, with 36.8% of RCF (Minimum Pump Speed (MPS) on the MELLLA line).

Cases 1, 2, and 3 were analyzed with Normal Feedwater Temperature (NFWT) and with FFWTR. Cases with FFWTR assumed an FW temperature reduction of 100°F. Case 4 was analyzed with NFWT only. [[

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9.1.2 LOCA Containment Hydrodynamic Loads

The JAF LOCA containment hydrodynamic loads assessment included the following:

- Pool swell (PS)
- Vent thrust
- Condensation Oscillation (CO)
- Chugging (CH)

The LOCA hydrodynamic loads were evaluated based on the short-term containment pressure and temperature response analysis.

Plant operation in the ARTS/MEOD region changes the mass flux and the subcooling of the break flow, which may affect the containment short-term LOCA response and subsequently the containment hydrodynamic loads. These loads have been defined generically for Mark I plants as part of the Mark I containment program, and are described in detail in the Mark I Containment Load Definition Report (LDR) (Reference 15). The LDR was reviewed and approved by the NRC in NUREG-0661 (Reference 12). The specific application of these loads to JAF is described in the References 13 and 14. The current containment hydrodynamic loads evaluation for JAF extends this evaluation by considering the entire reactor operating map for JAF including ARTS/MEOD and considering FFWTR.

The impact of ARTS/MEOD (and FFWTR) on the hydrodynamic load definition for SRV actuations is also addressed.

9.2 Evaluation Approach

9.2.1 Analysis Methods and Assumptions

Analysis Methods

Containment Model

The M3CPT code was used to calculate the short-term DBA-LOCA containment pressure and temperature response.

Vessel Blowdown Model- LAMB Break Flow

Vessel blowdown rates were calculated with the LAMB vessel model.

Analysis Assumptions

The assumptions given below are standard for Mark I short-term DBA-LOCA analyses with the GE M3CPT containment model with the use of LAMB generated break flows.

1. The line is considered to be severed instantly at the nozzle safe end to pipe weld. This results in the most rapid coolant loss and depressurization, with coolant being discharged from both ends of the break.
2. The reactor is assumed to scram at the time of accident initiation.
3. Analyses performed to evaluate the response at full rated power use an initial reactor power, which corresponds to 102% of CLTP.
4. The initial suppression pool temperature is at maximum technical specification value for normal operation. Analyses performed to evaluate hydrodynamic loads use a nominal initial suppression pool temperature per Reference 15.
5. Surface steam condensation in the drywell is neglected.
6. Upper bound on vent loss coefficients is used.
7. Thermodynamic equilibrium in the drywell is assumed at all times for recirculation line breaks.
8. The constituents of the fluid flowing through the vents are based on a homogeneous mixture of the fluid in the drywell. The consequences of this assumption result in complete liquid carryover into the wetwell.
9. The flow in the vents is compressible except for the liquid phase.
10. Thermodynamic equilibrium exists between the pool, air, and vapor within the wetwell. The air is fully saturated with vapor at all times.
11. Maximum pool mass corresponding to the maximum initial suppression pool water level and maximum vent submergence is used.

12. Minimum wetwell airspace volume corresponding to suppression pool High Water Level is used.
13. Decay heat used to generate the LAMB break flow is that used for the ECCS/LOCA analysis (ANS 5 decay heat with a 1.2 multiplier).
14. Air bubble back pressure effects are neglected.
15. FW flow is ramped to zero within the first few seconds (standard LAMB assumption for ECCS/LOCA analyses).
16. The initial drywell and wetwell pressures are selected based on the dP system operating at the maximum value permitted by the JAF Technical Specifications.

9.2.2 DBA-LOCA Short-Term Containment Pressure and Temperature

The short-term DBA-LOCA containment pressure and temperature analysis includes a sensitivity analysis and a confirmatory calculation.

Sensitivity Analyses

An initial set of sensitivity analyses were performed at the reactor conditions described in Section 9.1. Cases 1 through 4 were performed with NFWT. Cases 1a through 3a use the same reactor conditions as used for Cases 1 through 3 respectively, but include FFWTR. The sensitivity analyses were performed using the GE M3CPT code with LAMB generated break flows. These codes use the same basic models as used for the short-term DBA-LOCA containment analyses of References 16 and 17. However, for the sensitivity analyses, the critical break flow calculated with the LAMB code was modeled using the Slip break flow model. This is a more conservative model than that used for the References 16 and 17 analyses, which used the homogeneous equilibrium model (HEM) critical break flow model (Reference 18). The LAMB usage of the HEM includes a subcooled critical flow multiplier of 1.25 to ensure a conservatively high blowdown flow. The Slip break flow model was used for the sensitivity calculations because the LAMB break flow based on the Slip model is easier to apply to the containment analysis. The sensitivity analyses with the Slip break flows are only used to establish trends with different reactor operating conditions. The results based on Slip critical break flow are not used to confirm design limits. As described below, design limits were confirmed using the results of a confirmatory calculation using LAMB break flows based on the HEM critical break flow model.

The sensitivity analyses are used to establish trends in results with the different reactor conditions and to establish a limiting reactor condition with respect to the short-term containment response.

Confirmatory Calculation

A confirmatory calculation was performed for the limiting condition (102%P / 105%F, NFWT) established by the sensitivity analyses. The confirmatory calculation was performed using the

M3CPT code and LAMB generated break flow based on the HEM critical break flow model with a subcooled critical flow multiplier of 1.25. The confirmatory calculation was performed using the same approach as that applied for the analyses in References 16 and 17. The confirmatory calculations included one case with the same nominal values of initial drywell and wetwell pressure as used for the Reference 16 analysis (Case 5). A second confirmatory case (Case 5a) used the same initial drywell and wetwell pressure conditions as used for the Reference 17 analysis. The initial conditions for Case 5a correspond to the maximum values allowed by the JAF Technical Specifications.

9.3 Results

9.3.1 Short-Term DBA-LOCA Containment Pressure and Temperature

Sensitivity Analyses

Table 9-1 (Cases with NFWT) and Table 9-2 (Cases with FFWTR) provide a summary of the results of the sensitivity analyses with Slip break flow. The results of the sensitivity analyses show that both peak drywell pressure and temperature for the DBA-LOCA are bounding at the 102%P / 105%F condition with NFWT (Case 2 in Table 9-1). The peak drywell-to-wetwell differential pressure, the time of vent clearing, and the drywell pressure just prior to vent clearing are also included in Table 9-1 because these parameters are used to indicate trends in the pool swell loads and vent-thrust loads. A higher drywell pressure prior to vent clearing with a smaller vent clearing time is indicative of a higher initial drywell pressurization rate and, therefore, a higher pool swell load. Higher values of peak drywell-to-wetwell pressure are indicative of higher vent thrust loads. Based on the sensitivity analyses, operation with ICF (with NFWT) results in a very small increase in the containment response parameters relative to the results obtained with the assumption of rated power and core flow and NFWT. MELLLA operation and/or operation with FFWTR does not result in a higher DBA-LOCA peak drywell pressure or temperature or in DBA-LOCA containment conditions which produce higher LOCA hydrodynamic loads.

Confirmatory Calculation

The results of the confirmatory calculations (Cases 5 and 5a) are given in Table 9-3. Table 9-3 also contains a comparison to the peak calculated drywell pressure values reported in References 16 and 17, to the values reported in the JAF UFSAR and to the JAF containment design limits. The results show that the peak values for the key containment parameters are bounded by results previously reported and well within design limits. Figures 9-1 and 9-2 show the short-term DBA-LOCA containment pressure and temperature response for Case 5a.

A LOCA containment hydrodynamic loads evaluation was also performed using the results of the M3CPT/LAMB confirmatory calculation with nominal initial conditions (Case 5). Per Reference 15, nominal initial conditions were assumed in DBA-LOCA loads evaluation using

M3CPT results. The loads evaluation and results of the evaluation are described in Section 9.3.2.

9.3.2 Containment LOCA Hydrodynamic Loads Evaluation

The LOCA hydrodynamic loads are defined for JAF in the LDR and the Plant Unique Analysis Report (PUAR) (Reference 14). The results of a plant-specific evaluation used to confirm containment adequacy for these loads are documented in the PUAR.

The current containment LOCA load evaluation addresses the vent thrust loads, pool swell loads, CO loads, and chugging loads. The containment response inputs used to evaluate these loads are based on the M3PCT short-term DBA-LOCA analysis from Case 5 in Section 9.2.1. The purpose of the evaluation is to confirm that the LOCA loads defined in References 13 and 15 and evaluated for JAF in Reference 14 remain valid.

The LOCA hydrodynamic loads evaluation and results are summarized below.

Vent Thrust Loads

Vent thrust loads occur as a result of non-condensable gases and steam being discharged from the drywell, via the vents/downcomers, to the suppression pool. Vent thrust loads are calculated using the equations documented in Reference 15 for two conditions: 1) before vent clearing, and 2) after vent clearing. Plant specific vent thrust loads were defined for JAF in Reference 13.

The results of the vent-thrust load calculations using the results of Case 5 confirmed that the calculated vent thrust loads are all bounded by those previously defined in the Plant Unique Load Definition (PULD) (Reference 13).

Pool Swell Loads

Pool swell describes the initial containment response following a LOCA. The DBA event for pool swell for the JAF Mark I containment is a double-ended break of a recirculation suction line. The liquid mass flow, which initially flows from the break, flashes to steam and pressurizes the drywell. The drywell pressurization expels the water in the vents, which forms jets in the suppression pool and causes loads on the structures on the bends in the vent, as well as near the vent exit. Following the expulsion of the water (vent clearing), the non-condensable gases initially in the drywell are forced through the vents/downcomers into the suppression pool and expand as a bubble under the pool surface at each vent/downcomer exit location. The expansion forces the slug of water above the air bubble to accelerate upward, which causes both impact loads on structures initially above the pool surface and drag loads as the water slug flows past the submerged structures. The expansion also produces loads on the suppression pool boundaries. The water slug rises to a peak height at which point the air bubble breaks through the water surface and the water slug collapses.

The loads that occur include the torus vertical loads and shell pressures, impact and drag (i.e., standard and acceleration drag) loads on the vent system and structures, froth and pool fallback loads, bubble drag loads on submerged structures, and the submerged structure jet loads. These loads are controlled by the drywell pressure-time history during pool swell.

The plant-specific pool swell load definition for JAF was provided in Reference 13. The pool swell load definition of Reference 13 was based on the results of plant-specific pool swell tests performed in the quarter-scale test facility (Reference 19). A key parameter used to quantify the severity of the pool swell load is the initial pressurization rate. To evaluate the pool swell load, the value used in the Reference 19 test is compared to the value obtained from the M3CPT analysis results for Case 5. However, because the quarter-scale test conditions and results are scaled down to quarter-scale, the pressurization rate used in the Reference 19 test are scaled up to full-scale for the comparison. The test condition pressurization rate and scale factor for the JAF quarter scale tests (Reference 19) are 31 psi/sec and 0.2627, respectively. From Reference 19, the composite scaling factor then becomes $\sqrt{0.2627}$ so that the pressurization rate in full scale is $31.0 / \sqrt{0.2627} = 60.48$ psi/sec. The pool swell loads evaluation reviews the drywell pressurization rate obtained from Case 5 and compares it to the Reference 19 pool swell test condition scaled up to full-scale.

An initial drywell pressurization rate of 58.9 psi/sec was calculated for Case 5. The results of the confirmatory M3CPT05A calculations for Case 5 therefore confirmed that the Reference 16 pool swell test condition bounds the pressurization rates determined from the confirmatory analysis.

Condensation Oscillation Loads

CO loads result from oscillation of the steam-water interface that forms at the vent exit during the region of high vent steam mass flow rate. This occurs after pool swell. The CO loads include loads on submerged boundaries and submerged structures. Generally, the CO load increases with higher pool temperature and/or higher vent mass flow rate. The basis for the Mark I loads is the LDR which, in turn, is the direct application of test data from the GE Full Scale Test Facility (FSTF) tests (Reference 20). The FSTF tests were designed to simulate LOCA thermal-hydraulic conditions (i.e., vent steam mass flux and pool temperature), which bound all US Mark I plants including JAF. The CO loads can be quantified by the torus wall pressure root-mean-square (RMS) pressure. A correlation of RMS pressure to the key parameters affecting CO pressure (vent steam and liquid mass flux and suppression pool temperature) was used for the evaluation. The RMS pressure which is calculated with this correlation using the results of an M3CPT simulation of the FSTF test was used to establish a Mark I CO load baseline. The CO RMS pressure calculated with the correlation using the results of the M3CPT FSTF simulation was compared to the correlation RMS pressure obtained using the results of the confirmatory M3CPT calculation (Case 5). The comparison confirmed that the correlation RMS pressure based on the FSTF test data bounds that obtained with the confirmatory M3CPT results thus revalidating the JAF CO load definition.

Chugging Loads

Chugging occurs when steam mass flux through the vent is not high enough to maintain a steady steam/water interface at the vent exit. This means that chugging occurs at the tail end of a DBA or intermediate break accident (IBA) or anytime during a small break accident (SBA) with the reactor at pressure.

The design chugging load for JAF is based on the FSTF tests. These tests were run for a range of blowdown and containment conditions developed to bound all Mark I plants. Because the MEOD or FFWTR operation does not expand the range of steam mass flux, suppression pool temperature, and air content beyond the test conditions used to define the chugging load, it is concluded that the current design chugging load is not affected by MEOD or FFWTR operation.

9.3.3 Containment SRV Actuation Loads

The methodology used to define the SRV containment loads is described in Reference 15. Plant specific SRV loads were defined and evaluated for JAF in Reference 14.

The SRV actuation loads can be divided into two main categories: 1) first SRV actuations; and 2) subsequent SRV actuations. The SRV loads for both initial and subsequent actuations can also be divided into two categories. The first category includes the internal pressure loads and thrust loads on the SRV discharge line (SRVDL) and quencher. The secondary category includes the loads resulting from air-bubble formation in the suppression pool following water and air clearing. This second category includes the air-bubble pressure loads on the submerged pool boundaries and air-bubble drag loads on the submerged structures.

The controlling parameters for the SRV loads include: 1) SRVDL and containment geometry; 2) initial water leg length in the SRV discharge line; and 3) SRV flow rate, which is primarily determined by the SRV setpoints, line geometry, and line losses.

Loads due to initial SRV actuation are determined by the SRV setpoints, SRVDL volume, line lengths and friction losses, number of turns, etc. Because all of these parameters, including the SRV setpoints, do not change, loads due to initial SRV actuations are not affected by MEOD or FFWTR.

Loads due to subsequent SRV actuations depend primarily on the maximum SRVDL reflood height at the time of SRV opening and time intervals between openings. The maximum SRVDL reflood height is controlled by the SRVDL geometry and the SRVDL vacuum breaker capacity. The time intervals between SRV openings is controlled by the reactor pressure response, which in turn depends on the rector power level and to a lesser extent on the ECCS flows and suppression pool temperature (source of ECCS). Because the SRVDL geometry, SRVDL vacuum breaker capacity, reactor power level, ECCS flow rate, and suppression pool temperature do not change, loads due to subsequent SRV actuations are not affected by MEOD or FFWTR.

Therefore, operation with ARTS/MEOD or FFWTR has no affect on the currently defined SRV load for JAF.

9.4 Conclusions

Based on the evaluations presented in this section, it is concluded that MEOD including operation with FFWTR does not result in an increase in the peak DBA-LOCA drywell pressure nor result in conditions that would produce higher LOCA hydrodynamic loads. The results of a confirmatory M3CPT calculation using current LAMB/HEM break flows at NFWT conditions confirmed that the DBA-LOCA short-term results reported in References 16 and 17 and in the JAF UFSAR, which were determined to be within design limits, remain bounding. The results of the containment loads evaluation show that all containments loads remain within their defined values given in References 13, 14, and 15.

Table 9-1
Summary of Sensitivity Analysis Results (NFWT)

Power (% Rated) / Flow (% Rated)	Case 1 102P / 100F	Case 2 102P / 105F (ICF)	Case 3 102P / 79.8F (MELLLA)	Case 4 62P / 36.8F
Peak Drywell Pressure (psia)	58.30	58.37	57.91	54.84
Peak Drywell Temperature (°F)	290.16	290.24	289.64	286.15
Peak Drywell - Wetwell Pressure Differential (psid)	25.81	25.82	25.63	23.65
Time of Vent Clearing (sec)	0.1314	0.1314	0.1328	0.1404
Drywell Pressure Prior to Vent Clearing (psia)	24.83 @ 0.1314 sec	24.87 @ 0.1314 sec	24.63 @ 0.1323 sec	23.93 @ 0.1396

Table 9-2
Summary of Sensitivity Analysis Results (FFWTR)

Power (% Rated) / Flow (% Rated)	Case 1a 102P / 100F	Case 2a 102P / 105F (ICF)	Case 3a 102P / 79.8F (MELLLA)
Peak Drywell Pressure (psia)	57.72	57.82	57.46
Peak Drywell Temperature (°F)	289.43	289.56	289.10
Peak Drywell - Wetwell Pressure Differential (psid)	25.34	25.41	25.05
Time of Vent Clearing (sec)	0.1338	0.1331	0.1360
Drywell Pressure Prior to Vent Clearing (psia)	24.48 @ 0.1338 sec	24.43 @ 0.1318 sec	24.16 @ 0.1348 sec

Table 9-3
Summary of M3CPT05A Confirmatory Analysis Results Using LAMB With HEM

Power (% Rated) / Flow (% Rated)	Case 5 102P / 105F (ICF)	Case 5a 102P / 105F (ICF)	Reference 16 102P / 81F (MELLLA)	Reference 17 102P / 81F (MELLLA)	UFSAR Sections 5.2 and 14.6 Initial Core
Feedwater Temperature Assumption	NFWT	NFWT	NFWT	NFWT	
Critical Break Flow Model	HEM	HEM	HEM	HEM	
Initial Drywell Pressure (psia)	16.50	17.70 (a)	16.50	17.70 (a)	
Initial Wetwell Pressure (psia)	14.85	16.00 (a)	14.85	16.00 (a)	
Peak Drywell Pressure (psia)	53.15	54.45	55.9	57.2	
Peak Drywell Pressure (b) (psig)	38.45	39.75	41.2 (d)	42.5	45 (d)
Peak Drywell Temperature (c) (°F)	282.51	285.87			

Notes:

(a) Maximum operating pressures per the Technical Specification

(b) Design Limit for Drywell Pressure = 56 psig, peak calculated value of 45 psig

(c) Design Limit for Drywell Structure Temperature = 309°F

(d) A value of 41.2 psig is reported in UFSAR Section 16.9.3.5.1.3 which was obtained from the Reference 16 Power Uprate analysis.

Figure 9-1
JAF MEOD Short-Term Containment Pressure Response (Case 5a)

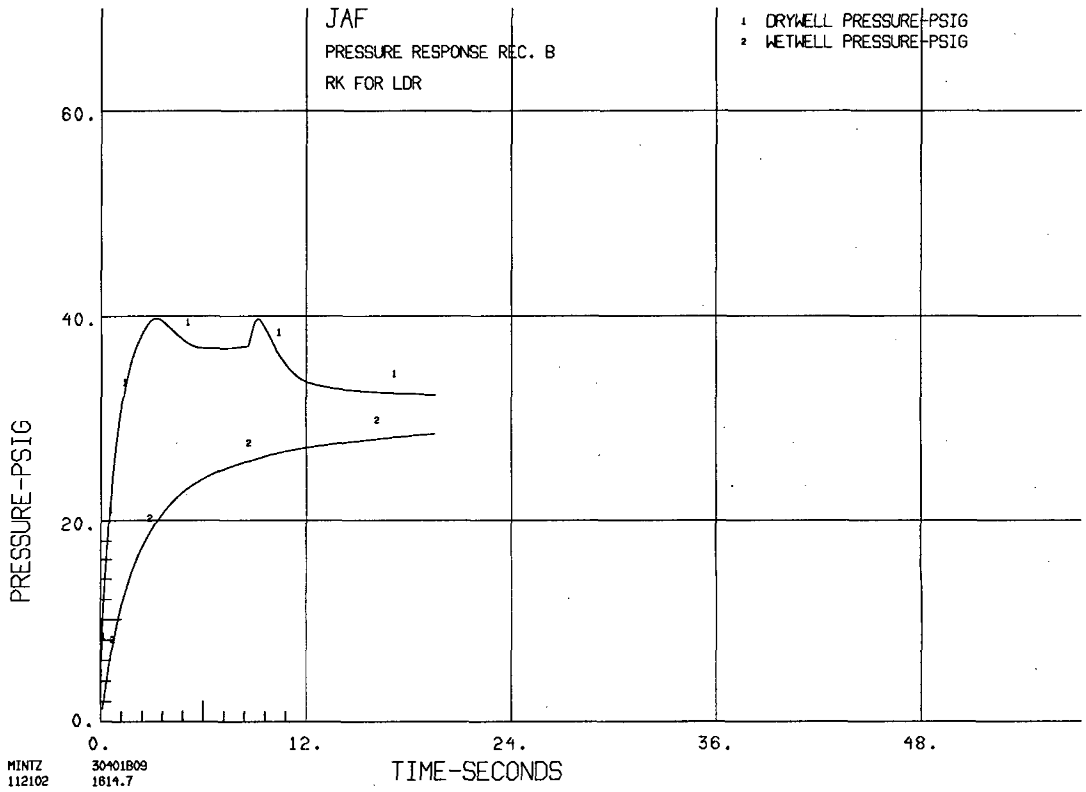
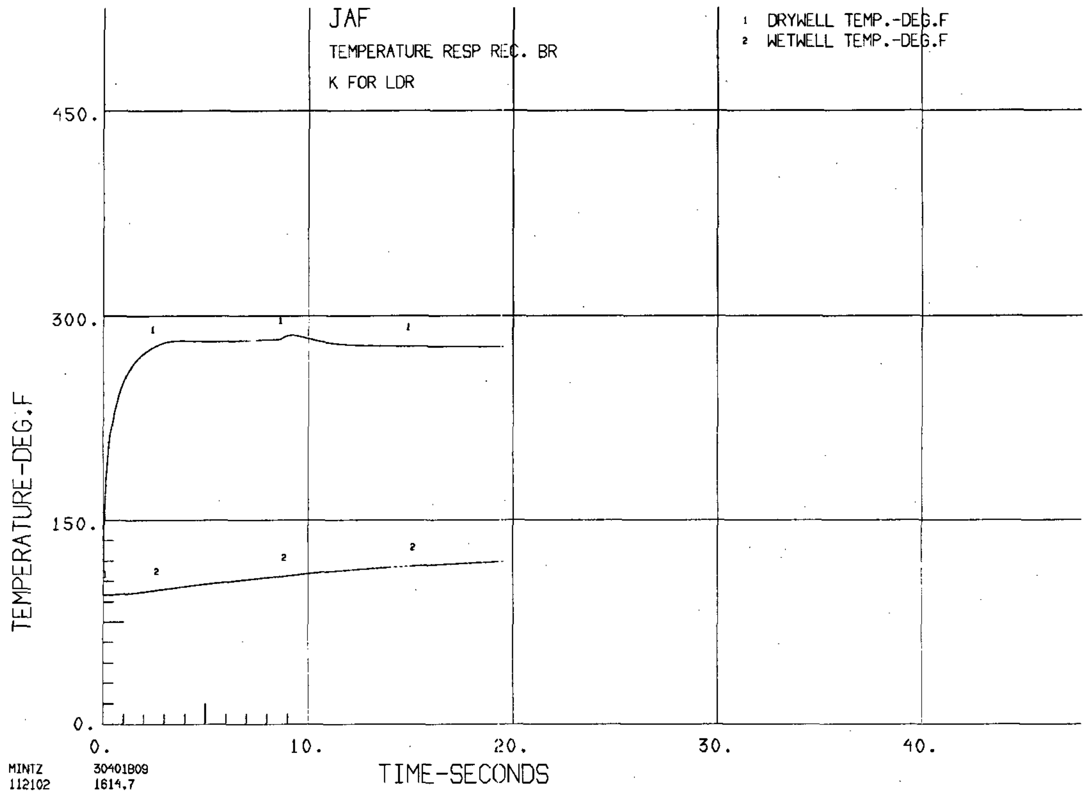


Figure 9-2
JAF MEOD Short-Term Containment Temperature Response (Case 5a)



10.0 REACTOR INTERNALS INTEGRITY

10.1 Reactor Internal Pressure Differences

The ICF condition increases the pressure drop across the reactor internal components and the fuel channels because of the increase in the core flow (105% core flow). Higher core flow results in higher resistance and thereby higher pressure drop across the core plate and other internal components. The reactor internal pressure differences (RIPDs) for the MELLLA condition are bounded by the ICF (105% core flow) and the existing ELLLA (87% core flow) conditions due to the lower core flow in the MELLLA domain (80% core flow). Thus, the RIPD analysis is based on the limiting ICF condition and is also applicable to the MELLLA condition.

The fuel lift margin is also analyzed for the limiting ICF condition and the fuel lift margin at MELLLA conditions are also bounded by ICF as a result of higher core plate pressure drop.

10.1.1 RIPD Analysis Approach and Inputs

The RIPD analysis was performed for the Normal, Upset, Emergency, and Faulted conditions. The RIPD analysis for Normal operating conditions was performed at 100% power and 105% core flow. The analysis input assumptions were for GE12 and GE14 fuels, which are the fuels currently in the JAF core. The RIPDs for the Upset condition were determined by applying conservative adders and multipliers to the steady-state Normal condition pressure differences.

The RIPDs for the Emergency condition were determined based on the limiting emergency event, i.e., the inadvertent opening of all Automatic Depressurization System (ADS) valves. The analysis assumes a bounding condition at 102% power and 105% core flow (the 2% additional power is based on the requirement of Regulatory Guide 1.49). The RIPDs for the Faulted condition were also calculated to analyze the limiting main steam line break event. The analysis was performed at 102% power and 105% core flow. The Faulted condition RIPD calculation also includes an evaluation at the low power cavitation interlock point (22.5% power and 105% flow).

10.1.2 RIPD Analysis Results

The results of the RIPD calculation are shown in Tables 10-1 and 10-2. The RIPDs for the ICF condition increase approximately 12% compared to the previous RIPD licensing basis at the 100% rated core flow condition (Reference 16). The contributing factors to this increase are not only due to the 105% core flow of the ICF condition but also the assumption of the new fuel designs of GE12 and GE14 fuels versus the GE7 fuel design in the Reference 16 analysis. The fuel lift margin results are adequate for all operating conditions.

The ICF RIPD results in Tables 10-1 and 10-2 reflect the limiting GE12 fuel because GE14 is bounded by GE12 as a result of the higher core pressure drop associated with the GE12 fuel design.

10.2 Acoustic and Flow-Induced Loads

The acoustic and flow-induced loads are contributing factors to the JAF design basis load combination in the Faulted condition.

The ICF condition has no effect on the acoustic and flow-induced loads because it is bounded by the ELLLA and MELLLA conditions. The acoustic loads are imposed on the reactor internal structures as a result of the propagation of the decompression wave created by the assumption of an instantaneous RSLB. The acoustic loads affect the shroud, shroud repair, shroud support ring, and jet pumps. The flow-induced loads are imposed on the reactor internal structures as a result of the fluid velocities from the discharged coolant during an RSLB. The flow-induced loads only affect the shroud and jet pumps. The acoustic and flow-induced loads are dependent on the initial pressure and temperature conditions of the fluid in the downcomer region outside the core shroud, and the geometry of the vessel internals, the core shroud, and jet pumps. Thus, the increased subcooling in the downcomer associated with the MELLLA condition would increase the acoustic and flow-induced loads. From ELLLA to MELLLA, the core flow decreases thereby increasing the downcomer subcooling and the critical flow, and the mass flux out of the break in a postulated RSLB. As a result, the acoustic and flow-induced loads in MELLLA conditions increase slightly.

10.2.1 Approach/Methodology

As major components in the vessel annulus region, the core shroud, core shroud support, and jet pumps were evaluated for the bounding RSLB acoustic and flow induced loads representing the ARTS/MEOD conditions.

The flow-induced loads were calculated for an RSLB utilizing the specific JAF geometry and fluid conditions applied to a reference BWR calculation. The loads were calculated by applying scaling factors that account for plant-specific geometry differences (e.g., size of the core shroud, reactor vessel, and recirculation line) and thermal-hydraulic condition differences (e.g., downcomer subcooling) from the reference plant. The reference calculation is based on the GE methods utilized to support the NRC Generic Letter 94-03 that was issued to address the shroud cracks detected at some BWRs.

The acoustic loads applied for JAF shroud and jet pumps represent bounding loads for JAF because bounding subcooling and natural frequencies were applied. Bounding subcooling and natural frequencies for jet pump and shroud were applied. The acoustic loads on the shroud support are generic bounding loads for all BWRs based on GE methods used for flow-induced loads calculations.

For JAF, the most limiting subcooling condition is at the intersection of the minimum core flow and the MELLLA flow control line. The subcooling at this point is applied to the reference BWR calculation, along with the JAF geometry, to determine the specific flow induced loads.

10.2.2 Input Assumptions

The following assumptions and initial conditions were used in the determination of the acoustic and flow induced loads for the ARTS/MEOD operation. GE methodology conservatively assumes a 20% increase in the critical break flow model.

Analytical Assumptions	Bases/Justifications
Initial core thermal power at 102% of CLTP 102P/100F, NFWT	Consistent with JAF current licensing basis
Initial core flow at 80% of rated flow 102P/80F, NFWT, MELLLA	MELLLA power / flow state point at full power
63P/37F, NFWT, Minimum Pump Speed Point on the MELLLA upper boundary line	MELLLA low power / low flow point
52P/37F, Feedwater Temperature Reduction (FWTR), Minimum Pump Speed point on the 100% rod line	100% rated rod line low power / low flow point with the FWTR option
57P/37F, FWTR, Minimum Pump Speed Point on the ELLLA boundary line	ELLLA boundary line low power / low flow point with the FWTR option. Bounding power /flow state point for ARTS/MEOD.

10.3 Reactor Internals Structural Integrity Evaluation

The reactor internal components are subject to loads resulting from operation under steady-state and accident conditions. MELLLA with ICF (i.e., MEOD) causes increased pressure differentials (RIPD) across the reactor internal components for steady-state (Normal), transient (Upset), and accident (Emergency and Faulted) conditions along with increased acoustic and flow induced loads in the accident (Emergency and Faulted) conditions. In addition, fuel lift margins are reduced. The reactor internals evaluation was done on the basis of the combined loading effects of MEOD, and GE12 and GE14 fuel.

The resulting load changes were reviewed to assure that adequate margin exists to accommodate these loads and that the structural integrity of the internal components is maintained under all operating conditions.

The following key RPV internal structure components were reviewed:

Core Support Structure Components:

- Core Plate
- Top Guide
- Control Rod Drive Housing
- Control Rod Guide Tube

- Orificed Fuel Support
- Fuel Channel

Non-Core Support Structure Components:

- Jet Pumps
- Core Spray Line and Sparger
- Access Hole Cover
- Shroud Head & Steam Separator Assembly
- Shroud
- Shroud Repair Components
- Shroud Support

The JAF reactor internals are "non-ASME code" components and there are no specific code requirements that apply. However, ASME Code, Section III criteria and the JAF UFSAR were used as a guide where applicable, consistent with the original design basis of the components. Non-Core Support Structure components are also not required to meet ASME code requirements. However, the internals assessment was performed using the ASME code as a guideline, consistent with the original design basis of the components.

10.3.1 Structural Evaluation Results

For Normal and Upset conditions, the changes in loads are primarily due to increased RIPDs. There is also a small increase in core flow loads on some components. The originally documented reactor internals horizontal and vertical seismic loads, considering the effect of both GE12 and GE14 fuel were used. The bounding loading for each component was used for the evaluations. The temperature change in the lower plenum due to ICF and reduced FW temperature operation is small and results in an insignificant effect on thermal loads. In addition, the fuel lift margin in conjunction with seismic and control rod blade friction considerations remain acceptable.

For Emergency and Faulted conditions, the evaluations considered all RIPD and acoustic and flow induced load changes, as applicable. If the loads on a reactor internals component did not increase above the existing design basis value, no additional evaluation was required and the component was deemed acceptable. For those reactor internal components with higher loads, the loading was assessed consistent with existing design basis analyses.

The shroud tie rod assembly was evaluated for the RIPD loads shown in Tables 10-1 and 10-2. The RIPD loads were combined with other normal, upsat, emergency and faulted condition loads for this evaluation. The revised seismic loads due to GE14 fuel were also included. The evaluation considered the worst case failure scenario for the shroud horizontal welds. This evaluation showed that the tie rod assembly is structurally adequate for the revised RIPD loads.

The flow-induced loads at the 102% CLTP / 100% flow are summarized in Table 10-3. The flow-induced load multipliers for the limiting MEOD conditions are included in Table 10-4. The bounding acoustic loads for JAF are shown in Table 10-5.

For the MEOD evaluation, only the acoustic and flow induced loads due to an RSLB LOCA were evaluated. The acoustic and flow induced loads due to an RSLB are Faulted condition events. Therefore, only the Faulted condition was evaluated. The applicable load combination included both seismic and acoustic or flow-induced loads. The only components that are affected by acoustic and flow induced loads are the shroud, shroud support, and jet pumps.

The core shroud and the tie rod radial restraints were also evaluated for the revised RSLB loads which include the Flow Induced Loads listed in Tables 10-3 and 10-4 and the Acoustic Loads listed in Table 10-5. The evaluation showed that, although some of the restraint loads increased, they are well within the load capacity that was determined in the original analysis. Therefore, the core shroud, the radial restraints and the reactor vessel in the vicinity of the radial restraints are structurally adequate for the revised RSLB loads.

The results of the evaluations determined that all of the internal components are within the allowable stresses and functional criteria of the existing design basis.

In summary, the reactor internals identified in Section 10.3 are acceptable for MEOD operation considering GE12 and GE14 fuels.

10.4 Reactor Internals Vibration

The reactor internals vibration characteristics can be affected by a change in the flow control line, such as the increased rod lines associated with operation to the MEOD upper boundary line.

10.4.1 Approach/ Methodology

To ensure that the flow-induced vibration (FIV) response of the reactor internals is acceptable, a single reactor for each product line and size undergoes an extensively instrumented vibration test during initial plant startup. After analyzing the results of such a test and assuring that all responses fall within acceptable limits of the established criteria, the tested reactor is classified as a valid prototype in accordance with Regulatory Guide 1.20. All other reactors of the same product line and size are classified as non-prototype and undergo a less rigorous confirmatory test.

JAF was designated as prototype plant for BWR4, 218-inch diameter reactors in accordance with Regulatory Guide 1.20. FIV test was performed at JAF and data collected during plant start-up between October 1973 and October 1975. The critical reactor internals were instrumented with vibration sensors at JAF and, the reactor was tested up to 100% core flow at the 100% rod line. This data was used in the current evaluation of JAF for MEOD operation.

To support operation of JAF in the ICF region, the reactor internals measurements were analyzed to determine the acceptability of the flow induced vibration stresses on the reactor internal components due to ICF operation. The evaluation was made at 2536 MWt (CLTP) and at increased core flow of 105 % rated. For MELLLA operation, the rated power output remains the same, but core flow is reduced to 80% of rated as shown in Figure 1-1.

10.4.2 Inputs/Assumptions

The following inputs/assumption were used in the reactor internals vibration evaluation:

Parameter	Input
Plant data selected for flow induced vibration evaluation	JAF was designated as the prototype plant for the BWR4, 218-inch diameter reactors in accordance with Regulatory Guide 1.20. FIV data collected during JAF plant start-up between October 1973 and October 1975 was used. During the startup, the reactor was tested up to 100% core flow at 100% rod line.
Target plant conditions in the MEOD region selected for component evaluation	CLTP value of 2536 MWt and 80% of RCF (MELLLA boundary line) and, at ICF up to 105% of RCF, balanced flow conditions.
GE stress acceptance criterion of 10,000 psi is used for all stainless steel components	Limit is lower than the value allowed by the current ASME Section III design codes for the same material and is bounding for all stainless steel material. The ASME Section III value is 13,600 psi for service cycles equal to 10^{11}.

10.4.3 Analyses Results

The reactor internals vibration measurements report for JAF was reviewed to determine which components are likely to have significant vibration at MEOD conditions.

Because the vibration levels generally increase as the square of the flow, the lower plenum components (Control Rod Guide Tube (CRGT) and Incore Guide Tube (ICGT)) and the jet pumps whose vibrations are dependent on the core flow, will have a 10% increase in vibration due to 5% increase in the rated core flow. The vibration levels of those components were determined to be with in acceptance limits during ICF conditions. For MELLLA operation, flow rates are reduced to 80% of rated with power remaining unchanged. ICF vibration levels bound those at MELLLA conditions.

For the shroud, shroud head, separators, and the steam dryer, the vibrations are a function of steam flow, which at MEOD conditions is bounded by the steam flow at CLTP. For the FW sparger, the vibrations are a function of the FW flow, which at MEOD conditions is bounded by the FW flow at CLTP.

The jet pump sensing lines (JPSLs) were evaluated for possible resonance with the recirculation pump vane passing frequency (VPF) pressure pulsation due to pump speed increase for ICF and decrease for MELLLA conditions. It was determined that all the jet pump sensing lines natural frequencies are well separated from the VPF at ICF conditions (up to 1718 RPM). Therefore, there is no concern with resonance at ICF conditions. Frequency analysis for the exit lines (4, 14, 5, and 15) show that if sustained operation from 900 RPM to 1280 RPM is avoided, there is only a 2.3% chance of potential JPSL resonance with the VPF for these lines. Note that the cracking of a JPSL is not a safety issue and the plant can continue to operate safely with a failed JPSL.

The jet pump riser braces were evaluated for possible resonance due VPF pressure pulsations and it was determined the jet pump riser braces natural frequencies are well separated from the recirculation pump VPF during ICF and MELLLA conditions to have any increased vibrations.

The FIV evaluation is conservative for the following reasons:

- The GE criteria of 10,000 psi peak stress intensity is more conservative than the ASME allowable peak stress intensity of 13,600 psi for service cycles equal to 10^{11};
- The modes are absolute summed; and
- The maximum vibration amplitude in each mode is used in the absolute sum process, whereas in reality the vibration amplitude fluctuates.

Therefore, the FIV will remain within acceptable limits.

10.4.4 Conclusion

This section demonstrates that, from an FIV viewpoint, the reactor internals structural mechanical integrity is maintained to provide JAF safe operation in the MEOD domain. The potential for JPSL resonance during recirculation pump reduced speed operation as discussed above is a plant operation concern, not a safety concern.

10.5 Feedwater Temperature Reduction

The JAF FFWTR report (Reference 21) identifies the mass and energy releases from Subcompartment (Annulus) Pressurization (AP) loads at CLTP conditions. Operation in the MEOD domain reduces the maximum FWTR allowed at JAF.

10.5.1 Approach/ Methodology

The approach used was to calculate the mass and energy releases from AP loads at MEOD conditions for various reductions in FW temperatures, compare them with the results from Reference 21, and determine the maximum FWTR.

10.5.2 Inputs/Assumptions

The key input, which forms the basis for assessing the effect of MEOD with partial FWTR on the mass and energy releases from postulated AP loads, is the decrease in FW temperature from the normal FW temperature of 424°F. Additional input data is listed below:

- Steam dome pressure at off-rated power is 1059 psia.
- Off-rated reactor thermal power is 2587 MWt (102% of CLTP).
- Moody subcooled critical flow (slip flow) is assumed.

10.5.3 Analyses Results

The maximum allowable FWTR at the MELLLA conditions (102%P / 79.8%F) was determined to be 35°F, such that the mass and energy release would not exceed the mass and energy release reported in Reference 21.

Table 10-1
Summary of RIPD Results (Normal and Upset Conditions)

Components	Normal Condition		Upset Condition	
	100P / 100F Reference 16	100P / 105F ICF	102P / 100F Reference 16	102P / 105F ICF
Core Plate and Guide Tube	24.29	27.36	26.69	29.76
Shroud Support Ring and Lower Shroud	31.10	34.39	33.50	36.79
Upper Shroud	6.81	7.08	10.22	10.61
Shroud Head	6.96	7.84	10.44	11.76
Shroud Head to Water Level, irreversible	9.9	10.58	14.79	15.88
Shroud Head to Water Level, elevation	1.04	0.89	1.3	1.34
Channel Wall – Core Average Power Bundle	8.83	9.08	11.73	< 10.7
Channel Wall – Maximum Power Bundle	12.23	11.89	15.13	< 13.5
Channel Wall – Central Average Power Bundle	10.41	10.03	13.31	12.93
Top Guide	0.63	0.64	< 1.1	0.71
Steam Dryer	0.37	0.36	0.6	0.47

Table 10-2
Summary of RIPD Results (Emergency and Faulted Conditions)

Components	Emergency Condition [a]	Faulted Condition		
	102P / 105F	102P / 100F Reference 16	102P / 105F	22.5P / 105F
Core Plate and Guide Tube	29.5	27.3	32.0	33.0
Shroud Support Ring and Lower Shroud	43.0	49.3	54.0	55.0
Upper Shroud	15.1	27.1	30.0	31.0
Shroud Head	15.5	27.5	29.5	31.0
Shroud Head to Water Level, Irreversible	17.7	29.6	32.0	32.0
Shroud Head to Water Level, Elevation	1.5	2.3	1.5	2.6
Channel Wall – Core Average Power Bundle	10.7	N/R [b]	11.9	9.7
Channel Wall – Maximum Power Bundle	13.5	N/R	14.5	10.2
Top Guide	1.6	2.9	3.5	4.4
Steam Dryer	N/C [c]	5.4	N/C	<10.0

Notes:

(a) No Emergency condition was calculated in Reference 16.

(b) Not Reported.

(c) Not Calculated.

Table 10-3
Summary of Baseline Flow-induced Loads Results

Item	Component	Parameter	Unit	Maximum Loads[a]
1	Shroud	Baseline Force	lbf	235,100
2		Baseline Moment at the Shroud Centerline	in-lbf	17,310,000
3	Jet Pump	Baseline Force	lbf	15,752
4		Baseline Moment at the Jet Pump Centerline	in-lbf	895,000

Note:

(a) Loads at rated conditions 102P / 100F.

Table 10-4
Summary of Flow-induced Load Multipliers

Item	Component	Operating Conditions [a]	Vessel Pressure (psia)	Downcomer Subcooling (Btu/lbm)	Load Multiplier [b]
1		102P / 100F, NFWT	1058	21.63	[[
2		102P / 80F, NFWT, MELLLA	1058	27.24	
3	Shroud / Jet Pump	63P / 37F, NFWT, Minimum Pump Speed Point on the MELLLA upper boundary line	1055	46.73	
4		52P / 37F, FWTR, Minimum Pump Speed point on the 100% rod line	1055	49.34	
5		57P / 37F, FWTR, Minimum Pump Speed Point on the ELLLA boundary line	1055	53.14	]]
Item	**Component**	**Operating Conditions [a]**	**Vessel Pressure (psia)**	**Downcomer Subcooling (Btu/lbm)**	**Load Multiplier [c]**
6		102P / 100F, NFWT	1058	21.63	[[
7		102P / 80F, NFWT, MELLLA	1058	27.24	
8	Shroud	63P / 37F, NFWT, Minimum Pump Speed Point on the MELLLA upper boundary line	1055	46.73	
9		52P / 37F, FWTR, Minimum Pump Speed point on the 100% rod line	1055	49.34	
10		57P / 37F, FWTR, Minimum Pump Speed Point on the ELLLA boundary line	1055	53.14	]]

Notes:

(a) MELLLA does not include FWTR.

(b) Loads multipliers [[ ]] in critical break flow assumption.

(c) Loads multipliers [[ ]] in critical break flow assumption.

Table 10-5
Summary of Acoustic Loads Results

Item	Component	Parameter	Unit	Maximum Loads
1	Shroud	Total Lateral Force	lbf	2.292E6
2		Moment at the Base of the Shroud Centerline	in-lbf	2.866E8
3	Shroud Support	Total Vertical Force	lbf	2.20E6
4		Moment at the Shroud Support Plate Outside Edge Nearest the Break	in-lbf	3.236E8
5	Jet Pump	Total Lateral Force	lbf	2.927E4
6		Moment at the Center of the Base of the Jet Pump	in-lbf	1.777E6

11.0 ANTICIPATED TRANSIENT WITHOUT SCRAM

11.1 Approach/Methodology

The basis for the current ATWS requirements is 10 CFR 50.62. This regulation includes requirements for an ATWS Recirculation Pump Trip (RPT), an Alternate Rod Insertion (ARI) system, and an adequate Standby Liquid Control System (SLCS) injection rate. The purpose of the ATWS analysis is to demonstrate that these systems are adequate for plant changes associated with operation in the MEOD region. This is accomplished by performing a plant-specific analysis in accordance with the approved licensing methodology (Reference 22), to demonstrate that the ATWS acceptance criteria are met for operation in the MEOD region.

The expansion of plant operation to MEOD conditions affects the peak vessel pressure, and the peak long-term containment response (suppression pool temperature and containment pressure). The MEOD analysis assumed that the SRVs opened at the upper Analytical Limit of the SRV Electric Lift subsystem, and that the two lowest set SRVs were OOS. The analysis assumed the Cycle 16 core and an initial power level of 2536 MWt (100% of CLTP) with the corresponding MELLLA minimum core flow of 80% of RCF.

Two limiting ATWS events for JAF were re-evaluated at the most limiting MEOD power and flow point (100% of CLTP and 80% of RCF) with ARI assumed to fail, thus requiring the operator to initiate SLCS injection for shutdown. These limiting events were:

(1) Closure of all MSIVs (MSIVC); and

(2) Pressure Regulator Failure (Open) to Maximum Steam Demand Flow (PRFO).

The following ATWS acceptance criteria were used to determine acceptability of the JAF operation in the MEOD region:

(1) Fuel integrity:

- Maximum clad temperature < 2200° F
- Maximum local clad oxidation < 17%

(2) RPV integrity:

- Peak RPV pressure < 1500 psig (ASME service level C)

(3) Containment integrity:

- Peak suppression pool bulk temperature < 220°F
- Peak containment pressure < 62 psig

The adequacy of the margin to the SLCS relief valve lifting as described in NRC Information Notice 2001-13, "Inadequate Standby Liquid Control System Relief Valve Margin," was also assessed.

11.2 Input Assumptions

The following initial conditions and assumptions were used in the analysis:

Analytical Assumptions	Bases/Justifications
The reactor is operating at 2536 MWt (100% of CLTP).	Consistency with JAF current licensing basis.
Initial core flow is 80% of RCF.	Lowest core flow at rated power range to maximize the initial void fraction in the coolant, and thus more severe pressurization transient consequences.
Both beginning-of-cycle (BOC) and EOC nuclear dynamic parameters were used in the calculations.	Consistency with generic ATWS evaluation bases.
Dynamic void and Doppler reactivity are based on JAF Cycle 16 data.	ATWS analyses are fuel-cycle independent, thus utilization of JAF Cycle 16 fuel parameters are appropriate.
Sodium Pentaborate Solution Concentration is 10% by weight.	Minimum solution concentration to meet ATWS requirements.
Two SRVs OOS, specified as the valves with the lowest setpoints.	Consistency with the Technical Specifications.
SRV setpoints correspond to the upper Analytical Limit of Electric Lift subsystem	Consistency with the system design and Technical Requirements Manual.

The initial operating conditions are included in Table 11-1.

11.3 Analyses Results

A parametric study was performed for the ATWS overpressure response. Both the limiting MSIVC and PRFO events were evaluated at BOC and EOC conditions. The PRFO event at the BOC condition was determined to be the bounding case for the peak vessel pressure for the ATWS conditions. Table 11-2 summarizes the key short-term results for the PRFO at BOC case. The peak vessel bottom pressure response for this limiting event is below the ATWS vessel overpressure protection criterion of 1500 psig. Therefore, the vessel overpressure criteria for ATWS is met.

Table 11-3 shows that the highest calculated peak suppression pool temperature is 190°F, which is below the ATWS limit of 220°F. Table 11-4 shows that the highest calculated peak containment pressure is 13.4 psig, which is below the ATWS limit of 62 psig. Thus, the containment criteria for ATWS is met.

Coolable core geometry is assured by meeting the 2200°F peak cladding temperature and the 17% local cladding oxidation acceptance criteria of 10 CFR 50.46. [[

]]

Finally, there is adequate margin to prevent the SLCS relief valve from lifting (per NRC Information Notice 2001-13).

11.4 Conclusions

Results of the ATWS analysis performed to support operation at the MEOD conditions show that the maximum values of the key performance parameters (fuel cladding temperature, peak vessel pressure, suppression pool temperature, and peak containment pressure) remain within the applicable limits.

Table 11-1
Initial Conditions for ATWS Analyses

Parameters	Current Analysis
Dome Pressure (psia)	1055
Core Flow (Mlb/hr / % rated)	61.45 / 80
Core Thermal Power (MWt / %NBR)	2536 / 100
Steam/Feed Flow (Mlb/hr / %NBR)	10.976 / 100
Feedwater Temperature (°F)	424
Initial Void Reactivity Coefficient BOC value(c/%)	-14.2
Initial Void Reactivity Coefficient EOC value (c/%)	-11.8
Core Average Void Fraction BOC Value (%)	57.5
Core Average Void Fraction EOC Value (%)	45.1
SRV Opening Analytical Limts for Electric Lift (psig)	1152 (2 valves) 1157 (2 valves) 1162 (7 valves)
Sodium Pentaborate Solution Concentration in the SLCS Storage Tank (% by weight)	10.0
Nominal Boron 10 Enrichment (atom %)	34.7
SLCS Injection Location	Lower Plenum Standpipe
SLCS Injection Rate (gpm)	50.0
SLCS Liquid Transport Time (sec)	30
Initial Suppression Pool Temperature (°F)	95
Initial Suppression Pool Mass (Mlbm)	6.62
Service Water Temperature (°F)	95
High Dome Pressure ATWS-RPT Setpoint (psig)	1155*
Number of SRVs OOS (current Technical Specification requirement)	2
Average SRV Opening Analytical Limit for Electric Lift (psig)	1161

* Technical Specification Allowable Value

Table 11-2
Summary of Key ODYN Parameters for Bounding Short-term ATWS Calculation

	PRFO at BOC
Peak Vessel Bottom Pressure (psig)	1493
Time of Peak Vessel Pressure (sec)	31.4
Peak Neutron Flux (% rated)	367
Time of Peak Neutron Flux (sec)	19.3
Peak Vessel Heat Flux (% rated)	146
Time of Peak Heat Flux (sec)	22.8

Table 11-3
Peak Suppression Pool Temperature

Event	BOC	EOC
MSIVC	171°F	190°F
PRFO	175°F	189°F

Table 11-4
Peak Containment Pressure

Event	BOC	EOC
MSIVC	9.4 psig	13.4 psig
PRFO	10.2 psig	13.2 psig

12.0 STEAM DRYER AND SEPARATOR PERFORMANCE

The ability of the steam dryer and separator to perform their design functions during MEOD operation was evaluated. MEOD decreases the core flow rate for operation near the MELLLA boundary, resulting in an increase in separator inlet quality for constant reactor thermal power. MEOD also increases the core flow rate for operation near the maximum ICF boundary, resulting in a decrease in separator inlet quality for constant reactor thermal power. These factors, in addition to the core radial power distribution, affect the steam separator-dryer performance. Steam separator-dryer performance was evaluated to determine the effect of MEOD on the steam dryer and separator operating conditions, the entrained steam (i.e., carryunder) in the water returning from the separators to the reactor annulus region, the moisture content in the steam leaving the RPV into the main steam lines, and the margin to dryer skirt uncovery.

The evaluation concluded that the performance of the steam dryer and separator remains acceptable in the MEOD region. The moisture content at the MELLLA and ICF conditions increases less than 0.01 wt% compared to the moisture content at rated flow and CLTP, and remains below 0.1 wt%. The carryunder in the MEOD region increases less than 0.1 wt% compared with the carryunder at rated flow and CLTP, and remains below 0.35 wt%. However, the actual moisture performance of the steam separators and dryer will be determined by plant testing in the MEOD region (see Section 13). This testing will provide performance data that can be used to establish operating limitations, if required. If necessary, radial power peaking or core flow can be adjusted, individually or collectively, to maintain the moisture content at or below the desired value. The moisture content specification may also be increased by an evaluation of the affected systems and components.

13.0 TESTING

Required pre-operational tests (i.e., APRM and recirculation system flow calibrations) will be performed in preparation for operation at the MEOD conditions with the ARTS improvements. Routine measurements of reactor parameters (e.g., APLHGR, LHGR, MAPLHGR, MLHGR, MCPR) will be taken within lower power test conditions in the MELLLA region for the MELLLA power ascension and in the current operating domain prior to ascending into the ICF region. Core thermal power and fuel thermal margin will be calculated using accepted methods to ensure current licensing and operational practice are maintained.

Measured parameters and calculated core thermal power and fuel thermal margin will be utilized to project those values at the CLTP test conditions. The core performance parameters will be confirmed to be within limits to ensure a careful monitored approach to CLTP in the MELLLA region and then to the ICF region.

Test Condition A	Power/Flow Map region between +0% and –5% of MELLLA Boundary that extends up from the 50% core flow line to the core flow line that results in the 90% of CLTP on the MELLLA Boundary
Test Condition B	Power/Flow Map region within 95% and 100% of CLTP and between +0% and –5% of the MELLLA Boundary.
Test Condition C	Power/Flow Map region within ±2.5% of the FCL that extends up to CLTP at Maximum Core Flow and between 70% and 90% core flow.
Test Condition D	Power/Flow Map region within 95% and 100% of CLTP and between +0% and –5% of the Maximum Core Flow.

Initial MELLLA testing will be performed in Test Condition A. Power increase beyond Test Condition A will be along this constant FCL to Test Condition B. The ICF testing will be initiated in Test Condition C and, by following a constant FCL power ascension up to Test Condition D, will be completed in Test Condition D.

The APRMs will be calibrated prior to MEOD implementation. The APRM flow-biased scram and rod block setpoints will be calibrated consistent with the ARTS/MEOD implementation and all APRM trips and alarms will be tested. The flow-biased setpoints of the RBM will also be confirmed.

Acceptable plant performance in the MEOD power-flow range will be confirmed by inducing small flow changes through the recirculation flow control system. Control system changes are not expected to be required for MEOD operation, with the possible exception of tuning following evaluation of testing. Subsequently, the recirculation system flow instrumentation calibration will be confirmed within Test Conditions B and D.

Steam dryer and moisture separator performance will be evaluated by measuring the main steam line moisture content. The evaluations will be conducted within Test Conditions B and D. Testing during the current operating cycle will establish the moisture carry-over fraction at CLTP and 100% of RCF. Other test condition power/flow operating points may be tested as deemed appropriate prior to the Test Condition B and D tests to demonstrate the test methodology or to confirm that acceptable steam moisture content at limiting operating conditions achieved before MEOD implementation.

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