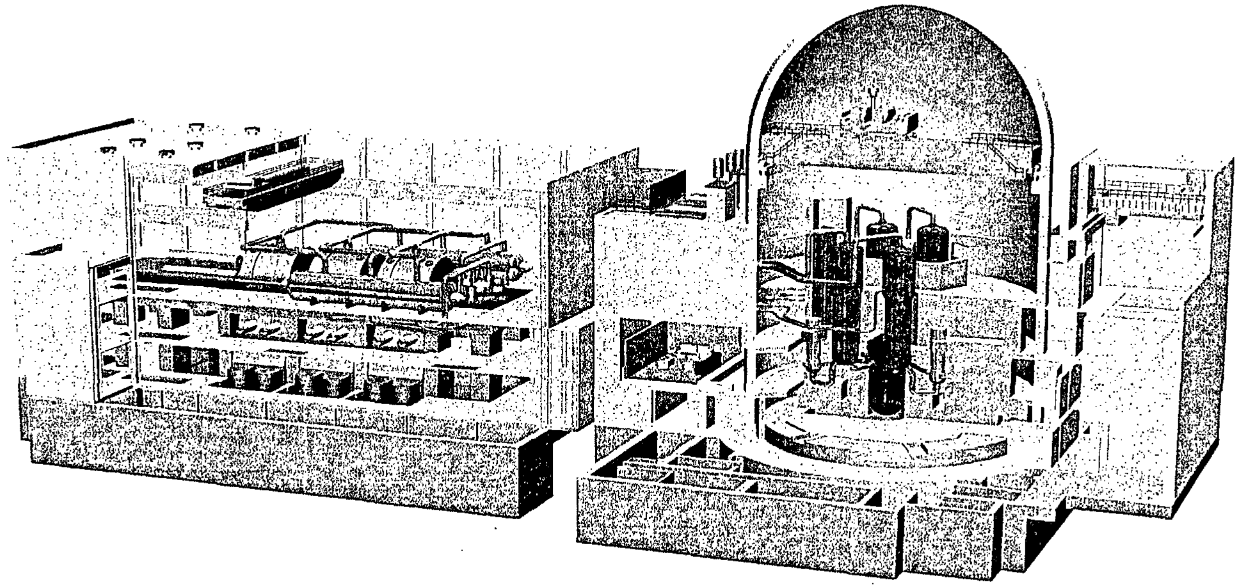
Non-Proprietary

US-APWR Topical Report

Non-LOCA Methodology



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ABSTRACT

This report presents the non-LOCA methodology that will be used in the Standard Review Plan (SRP) Chapter 15 safety analysis for MHI-designed pressurized water reactors such as the US-APWR. The contents of this document include description of the computer code, code validation, acceptance criteria and event specific methodology with sample transient analysis. The methodology for the analysis of radiological consequence is not described in this topical report.
The purpose of submitting this topical report during the US-APWR pre-application phase is to provide information to the NRC to facilitate efficient and timely review of the accident analysis to be provided in the Design Certification Document (DCD) as part of the Design Certification License Application.

The report provides an overview of the applicable methodology and the description of the specific models incorporated in the following MHI codes used to analyze non-LOCA accidents, as well as a discussion of the bases for applying these codes/methods to the US-APWR. Validation of the principal models of these codes by comparison with computer codes that have been approved by the NRC is presented

- MARVEL-M Plant system transient analysis code
- TWINKLE-M Multi-dimensional neutron kinetics code
- VIPRE-01M Subchannel thermal hydraulics analysis and fuel transient code

The event classification and associated acceptance criteria that will be used by MHI for each non-LOCA event included in the DCD are presented, ordered by the broader event categories defined by the SRP Chapter 15 and Regulatory Guide 1.206.

The following six events were selected to represent the spectrum of key analytical methods (combinations of codes), key SRP accident categories (15.1, 15.2, 15.3, 15.4, 15.6, and 15.7), and specialized models used by MHI in the non-LOCA accident analysis for the US-APWR.

- Uncontrolled RCCA Bank Withdrawal at Power
- Complete Loss of Forced Reactor Coolant Flow
- Spectrum of RCCA Ejection
- Steam System Piping Failure
- Feedwater System Pipe Break
- Steam Generator Tube Rupture

A detailed description of event sequences, method of analysis, analysis assumptions and sample transient results are provided in the topical report for each of these events. Appendices provide additional analyses to support selected methodology assumptions.

On the basis of the information in this topical report, it is concluded that the applied codes and methodologies are appropriate for US-APWR safety analyses. Also, it is concluded that the information provided in this topical report supports its purpose to provide key technical information related to the computer codes and methodology as well as the sample plant responses of the US-APWR related with the representing non-LOCA safety analysis to the NRC during the pre-application phase to facilitate an efficient and timely review of the Design Certification Application.

ACKNOWLEDGEMENTS

Mr. Tadakuni Hakata's contribution to the development and documentation of the MARVEL-M code is gratefully acknowledged.

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List of Acronyms

1-D	one-dimensional
3-D	three-dimensional
AOOs	Anticipated Operational Occurrences
APWR	Advanced Pressurized Water Reactor
ASME	American Society of Mechanical Engineers
BOC	Beginning of Cycle
CFR	Code of Federal Regulations
CVCS	Chemical and Volume Control System
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
ECCS	Emergency Core Cooling System
EFWS	Emergency Feed Water System
EOC	End of Cycle
EPRI	Electric Power Research Institute
GDC	General Design Criteria
HFP	Hot Full Power
HZP	Hot Zero Power
MHI	Mitsubishi Heavy Industries, Ltd
MSIV	Main Steam Isolation Valve
Non-LOCA	Non Loss of Coolant Accident
NRC	US Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
PAs	Postulated Accidents
PCMI	Pellet Cladding Mechanical Interaction
PCT	Peak Clad Temperature
PWR	Pressurized Water Reactor
RCCA	Rod Cluster Control Assembly
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RIA	Reactivity Initiated Accident
RIE	Reactivity Initiated Event
RPS	Reactor Protection System
RTDP	Revised Thermal Design Procedure
RWSP	Refueling Water Storage Pit
SAFDL	Specified Acceptable Fuel Design Limits
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SIS	Safety Injection System
SRP	Standard Review Plan

1.0 INTRODUCTION

The purpose of this topical report is to present the non-LOCA computer codes and methodologies that are adopted by Mitsubishi Heavy Industries, Ltd. (MHI) for the analysis of all non-LOCA events in the Standard Review Plan (SRP) Chapter 15, except LOCA and dose evaluation, for MHI-designed pressurized water reactors such as the US-APWR. The MHI non-LOCA methodology using the following codes is very similar to the conventional non-LOCA methodology used for currently operating US PWRs:

- MARVEL-M Plant system transient analysis code
- TWINKLE-M Multi-dimensional neutron kinetics code
- VIPRE-01M Subchannel thermal hydraulics analysis and fuel transient code

The MARVEL-M, TWINKLE-M and VIPRE-01M codes are MHI improved versions. The MARVEL code [References 1 & 2] and the TWINKLE code [Reference 3] were originally developed by Westinghouse Electric Corporation in the 1970s and used for licensing analysis for their US PWRs. The VIPRE-01 [Reference 4] code was originally developed by Battelle Pacific Northwest Laboratories, under the sponsorship of the Electric Power Research Institute and is also used for licensing analysis in the US.

Under a licensing agreement between Westinghouse and MHI, the MARVEL and TWINKLE codes were made available to MHI, and have been applied to licensing analysis for Japanese PWRs. For the non-LOCA safety analysis in the US, MHI uses the 4-loop MARVEL-M code [References 5]. The primary changes to TWINKLE-M are the increase in the maximum number of mesh points and adding the ability for the user to change certain fuel thermal properties. However, the underlying solution method remains unchanged.

The VIPRE-01M code is a MHI modified version of original VIPRE-01 code that includes some additional options concerning DNB correlations and fuel thermal properties. The topical report "Thermal Design Methodology" [Reference 6] submitted by MHI describes the modifications and validations related to the VIPRE-01M code.

MHI performs the non-LOCA safety analysis for the SRP Chapter 15 events using these computer codes and methodologies. This report describes:

- Section 2 – Computer Codes and MHI Modifications
- Section 3 – Validation of Models Utilizing Modified Codes
- Section 4 – Acceptance Criteria for SRP Chapter 15 Non-LOCA Events
- Section 5 – Non-LOCA Methodology for Typical Non-LOCA Events
- Section 6 – Sample Non-LOCA Event Analyses

2.0 COMPUTER CODE DESCRIPTION

As described in Section 1.0 the following computer codes are used by MHI for the non-LOCA safety analysis:

- MARVEL-M   Plant system transient analysis code
- TWINKLE-M   Multi-dimensional neutron kinetics code
- VIPRE-01M   Subchannel thermal hydraulics analysis and fuel transient code

Sections 2.1, 2.2, and 2.3 provide an overview of the plant system and mathematical models and detailed descriptions of the modifications associated with the MHI versions of each code. MHI modified these codes under Mitsubishi's Quality Assurance Program (QAP) [Reference 7].

2.1 MARVEL-M Code

2.1.1 Introduction

The MARVEL-M code is the same as the original MARVEL code from the viewpoint of constitutive and principal models. The main differences between the original MARVEL code and MARVEL-M are the extension from 2-loop simulation to 4-loop simulation and the addition of a built-in RCP model. The other refinements such as a pressurizer surge line node, a hot spot heat flux simulation model, and improved numerical solution and conversion techniques are described.

<u>History of MARVEL-M Development</u>

The use of digital computer techniques for safety design and safety evaluation of nuclear power plants started in the 1960s. The single loop LOFTRAN code was developed by Westinghouse Electric Corporation in the 1960s for control and protection analysis for pressurized water reactors. Limitations associated with the original single loop LOFTRAN code led to the development of BLKOUT [Reference 8], a code designed for long-term, multi-loop transient analysis for PWRs. The BLKOUT code was an improvement over single loop LOFTRAN because the BLKOUT code incorporated simulation of two loops. Additionally, the BLKOUT code utilized perfect mixing in the reactor vessel inlet, which was a reasonable assumption for analyzing long-term transients. The code was used for accident analysis, such as loss of all AC power to the station auxiliaries, loss of normal feedwater, and also for system design studies, such as the auxiliary feedwater system sizing studies.

In the early 1970s the BLKOUT code was modified to handle shorter time steps necessary for the computation of fast transients. This modified version of BLKOUT eventually evolved into the 2-loop MARVEL code. The MARVEL models accounted for the effect of multiple loops more precisely than the BLKOUT code, such as the multiple azimuthal flow channels in the reactor vessel and mixing model (no mixing -partial -complete) in the reactor vessel, but otherwise adopted methods similar to LOFTRAN.

Three or four loop plants were modeled in MARVEL as a 2-loop simulation by assuming that the other loops are operated in the same way as either of the two modeled loops. The code was used for safety analysis for multi-loop reactor plant transient response where the reactor coolant loops behave in a non-uniform manner, such as start-up of an inactive reactor coolant loop and steam line break.

The original MARVEL code was licensed to Mitsubishi Heavy Industries Ltd (MHI) with other computer codes including nuclear and thermal-hydraulic codes under the licensing agreement between Westinghouse and MHI in 1971. Some model improvements were subsequently made by MHI, such as a hot-spot heat flux model similar to the Westinghouse FACTRAN code [Reference 10]. Since then the MARVEL code has been used extensively for licensing safety analysis for various types of transients and design basis accidents, including Startup of an Inactive Reactor Coolant Loop, Loss of Reactor Coolant Flow, Inadvertent Rod Withdrawal, the Steam Line Break Accident, Feedwater Line Break Accident, and Steam Generator Tube Rupture Accident. The program has also been utilized as a tool for control studies and operating plant analysis for PWRs by MHI in Japan. In addition, certain realistic models were added as options to facilitate benchmarking and comparison with data obtained from operating plants.

In the 1990s, the original MARVEL code was expanded to a 4-loop version and added a reactor coolant pump model. This version of the code is denoted MARVEL-M. This evolution of the MARVEL-M code is graphically depicted below.

The MARVEL-M code is applicable to licensing safety analysis and control system studies and other applications for current PWR plants and for the APWR both for Japan and the US.

MARVEL-M Code Applicability to US-APWR Safety Analysis

The digital computer programs used for design and safety analysis in the Westinghouse pressurized water reactors were developed based on state-of-the-art technical and engineering knowledge at the time, and were influenced by the evolution of nuclear reactors developed by Westinghouse. As described above, the essential models of the MARVEL-M code are the same as the original MARVEL code developed by Westinghouse and approved by the NRC. The MARVEL-M code uses analytical models that are similar to those utilized in other codes, such as the LOFTRAN code, which have been used to license and continue to be used for safety analyses of their US PWRs. Due to the similarities between the US-APWR design and the current generation of US PWRs licensed by Westinghouse and the similarities between the codes used to analyze the transient response of the plant to an accident, it is concluded that the MARVEL-M code is applicable for performing the non-LOCA accident analysis for the US-APWR.

This topical report presents the overview of the MARVEL-M code in Section 2.1.2. Section 2.1.3 provides details on the improvement and refinement of certain original MARVEL models. Section 2.1.4 presents the realistic models incorporated as options in the MARVEL-M code for post-event analysis of a steam generator tube rupture event in Japan. Note that these realistic models are not used for original plant licensing. Section 2.1.5 discusses the precautions and limitations regarding the use of the MARVEL-M code.

2.1.2 General Description - Overview

MARVEL-M simulates reactor coolant loops and their associated systems; as well as the reactor core, pressurizer, control and protection system, safeguards system and others. The MARVEL-M code is applicable to 2-, 3-, and 4-loop PWR plants. A schematic diagram of the reactor systems simulated in 4-loop MARVEL-M is shown in Figure 2.1-1.

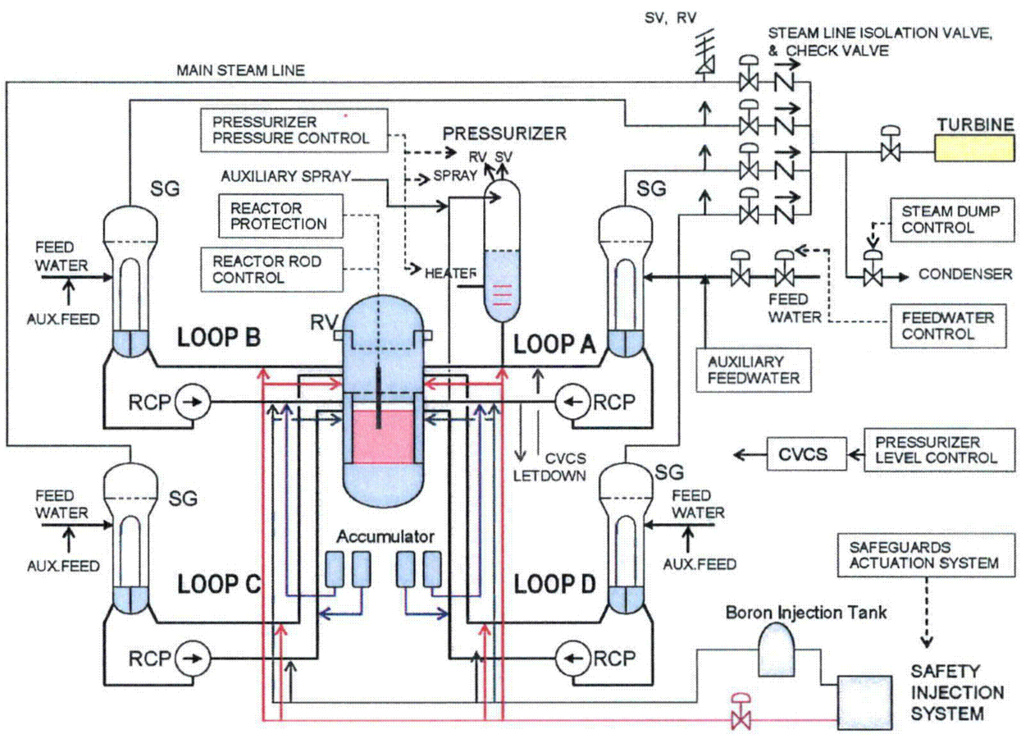


Figure 2.1-1 PWR Plant Systems Modeled in MARVEL-M Code

2.1.2.1 Reactor Core Model

(1) Neutron Kinetics

The instantaneous power is calculated from the sum of the contribution of the instantaneous fission rate and contributions from decay heat. The rate of change of fission power is calculated by a space-independent one-energy one point neutron kinetics model with six delayed neutron groups. Reactivity is calculated as the sum of contributions due to moderator density variations (or temperature variation), boron concentration, the fuel Doppler effect, and rod motion.

The reactivity variation due to fuel Doppler effect can be calculated by the change in the fuel effective temperature and the Doppler coefficient of reactivity, or Doppler power coefficient and fuel power expressed by the normalized average fuel temperature rise. The reactivity changes due to changes in the core coolant and fuel properties in the axial meshes are calculated from their coefficients of reactivity and the changes in core properties weighted by flux squared based on the perturbation theory approximation.

The reactor core model consists of up to four azimuthal flow sections corresponding to the coolant loops. If the core coolant properties are non-uniform, the azimuthal average of the core properties can be calculated using a user input reactivity weighting factor.

(2) Fuel Thermal Kinetics Models

The fuel rod kinetics are modeled by two equal volume concentric pellet nodes and one cladding node. The heat transfer coefficient from the clad surface to coolant is calculated using one of two different modes of heat transfer (1) sub-cooled convection or (2) local nucleate boiling. The total heat input to the coolant is the sum of the heat transferred from the cladding to the coolant and the heat generated in the coolant.

(3) DNBR Evaluation Model

MARVEL-M has ability to calculate the value of DNBR during a transient using a simple calculation model. The model employs user-input values of the DNBR at nominal core conditions and at selected DNBR limits represented by operating parameters of core inlet temperature, pressure and power levels. The code may accept input defining DNBR dependency on reactor coolant flow. The simplified DNBR model closely agrees with design calculations when the core operating conditions do not exceed the design flux distribution or core protection limits. When conditions exceed the limitations for the simplified model, DNBR analysis is performed by a more detailed external calculation code.

2.1.2.2 Reactor Coolant System Model

(1) Nodal Representation of the Reactor Coolant System

The thermal and hydraulic characteristics of the reactor coolant system are described by time and space dependent differential equations. They are reduced to nodal differential forms, the solutions of which are easily managed by means of finite differences. The system is divided into the following nodes or flow sections.

Table 2.1-1 Reactor Coolant System Flow Sections (4-Loop Model)

In each section, the mass and energy balance equations are solved by integration over time steps by the finite difference method, ignoring the momentum balance for forced reactor coolant flow.

Although not typically used for licensing analysis for non-LOCA accidents, MARVEL-M models two phase flow as a homogenous equilibrium mixture (except for the pressurizer). This model is acceptable if the predicted volumetric void fraction is within the fluid flow regime where a homogenous mixture is expected. If the predicted volumetric void fraction becomes significant, even within the homogenous regime, the user must check whether the boiling affects the validity of the analysis for the intended purpose.

Each individual reactor coolant loop has a reactor coolant pump and a steam generator. The reactor coolant flows are variable and flow reversal is permitted unless the overall reactor vessel inlet flow becomes negative.

In the steam generators, the heat transfer rate from each flow section to the secondary side is calculated based on the log-mean temperature difference so that the power transferred is computed accurately for a limited number of tube flow sections and over a wide range of primary flow conditions including natural circulation conditions. The heat transfer coefficient and heat transfer area are treated as variables and defined as functions of the representative operating parameters.

The reactor coolant circulating through the coolant loops enter the reactor vessel through the inlet nozzles. The coolant flows downward through the downcomer into the reactor vessel lower plenum, then turns and flows upward to the reactor core. After passing through the core the coolant enters the reactor vessel upper plenum and leaves the vessel through the outlet nozzles.

A small fraction of the flow entering the reactor vessel bypasses the reactor core. The flow is not considered effective for removing core power and is modeled by two paths. A small fraction of the bypass flow directly enters the reactor vessel upper head region through the cooling spray nozzles from the top of the downcomer. The coolant in the upper head is stagnant. The rest of the bypass flow goes up a flow channel from the lower plenum to the upper plenum without core heating.

(2) Mixing Model in Reactor Vessel

The coolant from the coolant loops is mixed in the reactor vessel lower plenum before entering the core. The coolant leaving the core is also mixed in the reactor vessel upper plenum. The mixing of the loop coolant is, however, known to be imperfect from the results of mixing tests conducted in the 1970s. If the coolant loop operation is not uniform, the core may be subject to operation with azimuthally tilted temperatures and nuclear fluxes. Figure 2.1-2 shows the reactor coolant flow paths and mixing in the reactor vessel.

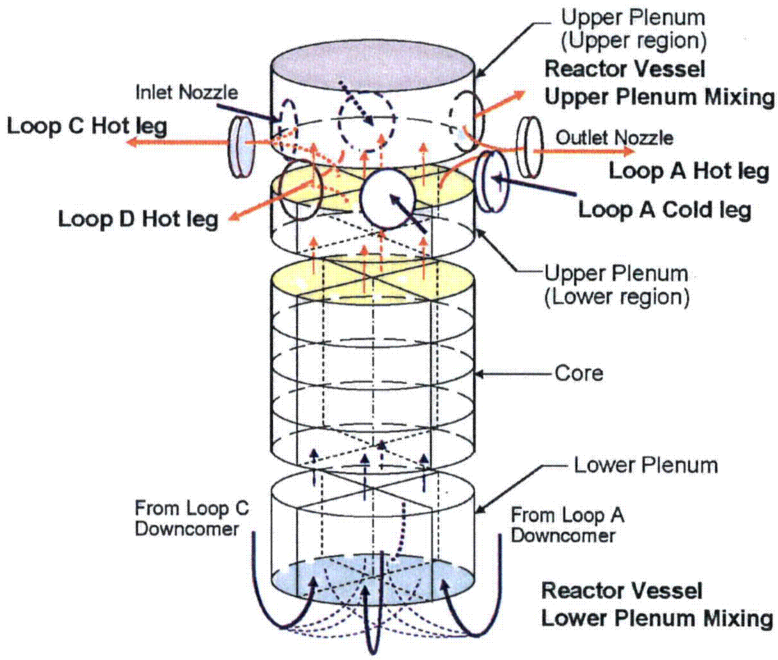


Figure 2.1-2 Reactor Coolant Flow and Mixing in Reactor Vessel

In order to simulate reactor conditions caused by imperfect mixing in the reactor vessel plenums, a maximum of four parallel flow channels can be provided in the reactor vessel as shown in Figure 2.1-2. The number of parallel flow channels is set at the actual number of reactor coolant loops in the application. Cross flow between the flow channels is first assumed in the reactor vessel inlet downcomer sections, if the loop operation is unbalanced, so that flow rates leaving the downcomer are uniform. Mixing of reactor coolant flows between the flow channels are assumed to occur in the reactor vessel lower plenum and the upper plenum in order to account for imperfect mixing of the coolant in the reactor vessel according to user input mixing factors.

The code input mixing factor, *FMXI*, is calculated by :

$$FMXI = (1 - f_{mi}) \cdot \frac{Nloop}{(Nloop - 1)} \qquad (1)$$

where

$Nloop$ = Total number of reactor coolant loops

f_{mi} = The fraction of the coolant flow emerging from an inlet nozzle which flows up the azimuthal part (per loop) of the core nearest the inlet nozzle.

FMXI = 0 means no mixing, while *FMXI = 1* means perfect mixing. The mixing in reactor vessel upper plenum is specified by a factor *FMXO* of the user input. The factor *FMXO* is defined such that *FMXO=0* means no mixing and *FMXO=1* means perfect mixing. It is calculated by the following equation:

$$FMXO = (1 - f_{mo}) \cdot \frac{Nloop}{(Nloop - 1)} \qquad (2)$$

where

f_{mo} = The fraction of vessel outlet flow leaving through an outlet nozzle which comes from the azimuthal part of the core nearest the outlet nozzle.

(3) Pressurizer Model

The change in the reactor coolant mass contained in the reactor coolant system (excluding the pressurizer) causes an outsurge or an insurge to the pressurizer through a pressurizer surge line which connects the pressurizer and the reactor coolant system hot leg. The pressurizer pressure is determined based upon an isentropic process during the steam expansion and contraction. An option is available that allows the isentropic process to change to a saturation process with an input time delay. After the pressurizer has emptied, the reactor coolant system pressure is determined by the compressibility of the coolant, in most cases a two phase mixture, in some part of the reactor coolant system.

2.1.2.3 Reactor Coolant Flow Transient Model

The manner in which primary system flows respond to a disturbance is important because such coolant flow removes the core heat and transfers it to the secondary fluid in the steam generators. An important phenomenon is the rapid flow decrease upon loss of reactor coolant pumping power – known as "flow coastdown". A flow decrease occurs also when the frequency of the electric power supply decreases causing the reactor coolant pump speed to follow the frequency decay. Mechanical failures such as pump rotor seizure cause rapid flow reduction, which are analyzed as the locked rotor accident. After a complete pump loss the post-accident plant operation must rely on the plant's capability for heat removal by natural circulation.

MARVEL-M is provided with a reactor coolant pump hydraulic and kinetics model so that flow transients can be computed by the pump model in conjunction with the existing reactor coolant system hydraulic models. After the reactor coolant pumps are stopped, the reactor coolant flow becomes natural circulation. The natural circulation model is incorporated in the code.

2.1.2.4 Steam Generator and Secondary System Model

(1) Heat Transfer Coefficient from the Primary to the Secondary
The overall heat transfer coefficient in the steam generators consists of the four major thermal resistances: the primary convection film, the tube metal, the fouling, and the secondary side boiling heat transfer. Those resistances, except for the fouling resistance, are modeled to vary depending on changes in the appropriate parameters.

(2) Steam Generator Water Level and Heat Transfer Area
At no load conditions, the water level in the steam generators is calculated from the liquid volume contained in the steam generator shell and the geometry of the shell side volume. At power the water level is a complex function of liquid volume and other variables in the steam generator since the existence of void in the heating region in the shell side raises the water level considerably. The water level at power is modeled as a function of the water mass and the boiling rate in the secondary shell based on steam generator design calculations.

The effective heat transfer area is reduced when part of the steam generator tubes is uncovered. The effective heat transfer area is computed from the ratio of the water level to the height of the steam generator U-tube bundle. The heat transfer from the uncovered part of the steam generator tubes to the vapor in the shell side is assumed to be negligible and is not modeled.

(3) Steam Generator Secondary-Side Thermal Kinetics Equation
The steam generator secondary side contains a two-phase fluid. Assuming that most of the secondary fluid is at saturated conditions, and ignoring sub-cooling in the liquid in the preheat region and the downcomer, a thermal kinetics equation is derived from mass, volume and energy balance equations as a lumped saturated equilibrium mixture of vapor and liquid.

(4) Main Steam Lines and Steam Flow Distribution
MARVEL-M has the capability of simulating up to four steam generators and steam lines. The main steam lines from each steam generator are connected together at a common steam header, each via an isolation valve and a check valve.

If the operating conditions of the steam generators are different from each other, the steam outputs are unbalanced and the steam flow distribution is calculated from the steam pressure of each steam generator and the pressure losses through the steam lines to meet the total steam flow. A steam relief valve and up to three safety valves on each steam line are modeled; these valves are opened when the steam pressures increase above their respective set pressures.

2.1.2.5 Safety Systems and Miscellaneous Models

(1) Reactor Protection System - Reactor Trip
The reactor protection system is provided to protect the reactor core and plant design limits. The MARVEL-M code simulates the following reactor trips, which automatically insert the control rods to shut down the reactor when the trip signals reach or exceed their respective

setpoint. (Setpoints are usually set at the protection limit, which includes measurement error and channel error.)

- High Neutron Flux Trip
- High Flux Rate Trip
- Overtemperature ΔT Trip
- Overpower ΔT Trip
- Low Pressurizer Pressure Trip
- High Pressurizer Pressure Trip
- High Pressurizer Level Trip
- Low Steam Generator Water Level Trip
- High Steam Generator Water Level Trip
- Low Reactor Coolant Loop Flow Trip
- Turbine Trip

Overtemperature ΔT Trip and Overpower ΔT Trip protect the core operating limits, that define a region of permissible operation in terms of power, pressure, axial power distribution, and coolant temperatures.

The trip serves to protect the core against DNB and core exit boiling, accounting for all the adverse instrumentation setpoint errors and the time delays in signal measurement and processing. When the reactor coolant loop ΔT exceeds the calculated ΔT setpoint, the reactor is tripped.

Overpower ΔT Trip protects the reactor against excessive core thermal power. The protection line for the condition is a function of coolant temperatures and axial power distribution. When the reactor coolant loop ΔT exceed the calculated ΔT, the reactor is tripped.

(2) Safety Injection System (SIS)

The Safety Injection System is provided to deliver borated emergency core cooling water to the reactor coolant system to assure core cooling and reactivity control for accidents such as the main steam line break. The safety injection function is modeled in MARVEL-M, but the recirculation function used in the LOCA analysis is not modeled.

The Safety Injection System is equipped with two (or more) safety injection pumps, which take suction from the refueling water storage tank and deliver borated water to the reactor through injection lines. A boron injection tank may be modeled in the cold leg injection line to promptly deliver highly concentrated boric acid. The injection system also includes accumulators, pressurized with nitrogen and connected to each cold leg, which also deliver borated water to the reactor. The gas-pressurized accumulators function as a passive injection system, discharging automatically when the reactor coolant system pressure decreases below the accumulator pressure.

(3) Safety Injection System Actuation System

The following Safety Injection System Actuations are modeled in the code

- Low Pressurizer Pressure
- Low Pressurizer Pressure in Coincidence with Low Pressurizer Level
- High Steam Flow in Coincidence with Low-Low T_{avg}
- Steam Line Differential Pressure
- Steam Line Low Pressure
- Manual Safety Injection

(4) Steam Line Isolation and Feedwater Isolation

The main steam line isolation valves are closed by a steam line isolation signal, which is generated from coincidence of high steam flow, safety injection, low T_{avg}, and/or high containment pressure signals. Manual actuation of steam line isolation is available.

A safety injection signal closes all control valves and trips the main feedwater pumps to isolate feedwater lines and close discharge valves. A low T_{avg} signal coincident with a turbine trip also actuates feedwater isolation to avoid excessive cooldown of the primary side due to continued addition of cold feedwater to the steam generators.

(5) Other Models

Thick metal effects, the Rod Control System, the Steam Dump Control System, and the Chemical and Volume Control System are also modeled.

2.1.2.6 Perturbations

Perturbations in many parameters and systems can be simulated in the code.
Examples include but are not limited to the following:

- Reactivity
- Core Power
- Reactor Coolant Loop Flows
- Steam Flow
- Feedwater Flow and Feedwater Enthalpy
- Steam and Feedwater Isolation Valves
- Pressurizer Spray, Relief Valves and Auxiliary Spray
- Reactor Trip
- Steam Line Break
- Feedwater Line Break
- Reactor Coolant System Small Break (Including SG Tube Rupture)
- Safety Injection System Operation
- Chemical and Volume Control System Operation
- Turbine Runback and Trip
- Malfunction of Reactor Control Systems

2.1.3 Theoretical Models of MARVEL-M Improvement

This section describes the mathematical model improvements in the MARVEL-M code, including the models associated with the reactor coolant system loops and reactor coolant pump hydraulic kinetics model. This section also describes model refinements such as hot-spot fuel thermal kinetics and pressurizer surge line models.

2.1.3.1 Four Loop Reactor Coolant System Model

MARVEL-M code has the ability to simulate up to four reactor coolant loops. The hydraulic and thermal models of the individual reactor coolant flow sections and the models of the steam generators and pressurizer are the same as the original MARVEL code. The algorithm for core mixing in the reactor vessel in MARVEL-M is changed, although the basic model for each reactor coolant loop has remained the same as the original MARVEL code. The algorithm for the steam lines has changed to incorporate the expansion in the number of coolant loops that can be simulated by the code.

(1) Equation for Reactor Coolant Flow Sections

The reactor coolant system thermal kinetics equations are derived using a nodal approximation similar to the original MARVEL code. The nodes and flow sections are illustrated in Figure 2.1-3.

Figure 2.1-3 Reactor Coolant System Flow Model

One of six different flow models is used for each flow section. The six models include transport delay (denoted by *SLUG*), mixing (*MIXG*), a steam generator heat transfer section (*HEEX*), a core heated section (*HEAT*), a reactor vessel outlet plenum (*MIXS*), and an inactive coolant volume (*MIXD*). These models are functionally the same as the original MARVEL code.

(2) Dead Volume (Reactor Vessel Head Volume) (*MIXD*)
There is a plenum in the reactor upper head of the reactor vessel as shown in Figure 2.1-4. The volume is modeled as a control volume called the Dead Volume (*VDEAD*) as shown at the top of Figure 2.1-5.

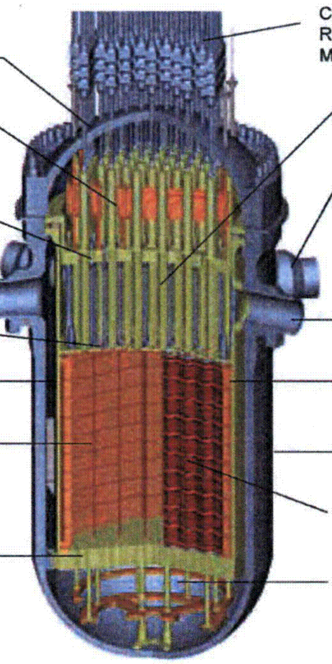


Figure 2.1-4 Reactor Vessel Inner Structure

A small fraction (*BYPASS*) of the coolant entering the reactor vessel bypasses the reactor core. A small fraction of the bypass flow (*FDEAD*) is diverted and flows directly into the vessel head plenum (Upper Head Dead Volume) through small cooling spray nozzles. A small fraction of upper plenum flow (*FUPH*) goes up into the area through control rod guide tubes in the central area. A small rate of fluid flow exits from the dead volume to the upper plenum through control rod guide tubes in the peripheral region and is mixed with upper plenum coolant. The flow pattern is caused by the pressure difference profile across the upper support plate during forced reactor coolant flow conditions. The flow paths are modeled as shown in Figure 2.1-5.

The coolant in the reactor vessel head plenum is stagnant and is virtually inactive during short-term transients. The flow model determines the steady-state temperature in the upper head volume using the formula below.

$$\text{Initial Enthalpy} = (FDEAD * H_{cold} + FUPH * H_{hot}) / (FDEAD + FUPH) \tag{3}$$

H_{cold} and H_{hot} are coolant enthalpy at cold leg and hot leg.

The user-defined fraction *FUPH* is smaller than *FDEAD* and the initial upper head temperature is generally close to the cold-leg temperature. The value of *FUPH* can be varied by the user to reflect different upper head temperature assumptions.

During plant cooldown or depressurizing transients the reactor coolant in the dead volume may flash and form a steam phase at the top, separated from the liquid. It may act as an alternate pressurizer to define the reactor coolant system pressure after the pressurizer is emptied. The downward flow leaving the dead volume is assumed to be single-phase liquid until the void fraction becomes very large.

When the total mass in the reactor coolant system starts increasing, the pressurizer may begin to refill with water and, at the same time, the upper head vapor phase starts decreasing. During refilling of the boiled-off part of the dead volume, flow may occur from the reactor vessel upper plenum (below the core support plate) into the dead volume. The net incoming flow then refills the boiled-off part. This process may occur very slowly since the refilling requires condensation of the vapor existing in the dead volume. The processes in the pressurizer and the upper head depend on the reactor coolant condition. The described behavior of the dead volume is modeled in the code.

(3) Pressure Gradient in Reactor Coolant System

The core and reactor coolant system thermal-hydraulic model does not simulate the pressure gradient in the system except for the pressurizer and the reactor coolant system (at the hot-leg pressurizer surge line connection), although fluid properties in the flow sections are treated by two dimensions of specific enthalpy and pressure from sub-cooled to two phase flow (homogeneous).

Although, the reactor coolant system fluid in a PWR is normally pressurized and sub-cooled by the pressurizer, boiling of the coolant fluid may occur locally during upon specific transients due to excessive power increase or rapid depressurization.

The pressure gradient of the reactor coolant system is of importance for computation of pressure in the core for DNBR evaluation and the RCS maximum pressure, which usually occurs at the reactor coolant pump discharge with the pump running. The pressure differences between the pressurizer and those points are compensated for by adding pressure differences computed taking account of the pressure losses and the elevation effect.

2.1.3.2 Flow Mixing in Reactor Vessel (4-Loop Model)

The reactor coolant fluid that circulates in the reactor coolant loops is introduced into the reactor vessel through the inlet nozzles. Thus the mixing in the reactor vessel inlet and outlet plenum is imperfect. Therefore, in order to take this into consideration in the analysis of the reactor vessel thermal kinetics behavior, an azimuthal as well as an axial analysis is necessary. In this program, the azimuthal effect is considered by using a maximum of four separate flow channels for each loop simulated. Although this model is not detailed enough to describe the exact thermal and hydraulic behaviors, by use of certain code inputs, representative or conservative prediction of reactor nuclear and thermal transients can be made. The flow model is shown in Figure 2.1-5.

Figure 2.1-5 Reactor Vessel Flow Model

(1) Mixing in the Downcomer and Reactor Vessel Lower Plenum

The flow sections *V(14,1)* to *V(14,4)* in Figure 2.1-5 correspond to the flow volumes of the annular volume between the vessel wall and the core barrel, including inlet nozzles, as seen in Figure 2.1-4. (These flow sections will be called the downcomers from here on.) Reactor coolant flow enters the downcomers through the reactor vessel inlet nozzles. When the direct vessel safety injection to reactor vessel option is used, the safety injection flow is introduced to the corresponding downcomer volumes. The flow lines are not shown in Figure 2.1-5.

i) Cross flows in downcomers

When loop flows are unbalanced, cross flows are assumed between downcomer flow sections *V(14,1)* to *V(14,4)*. Cross flows are determined to satisfy the following conditions:

1. Cross flows may only exist between each downcomer section and the adjacent downcomer sections.
2. Cross flows occur so that coolant flow rates at the downcomer exit are uniform.

3. The cross flows are proportional to the differences between downcomer inlet flows.

ii) Mixing in lower plenum
Mixing in the reactor vessel lower plenum is assumed to occur, if specified in the code input, at the point where the coolant enters the lower plenum *V(15,1)* to *V(15,4)*. The mixing factor FMXI is defined by user input as follows:

$$\begin{aligned} & FMXI = 0 \text{ ; no mixing} \\ 0 < \ & FMXI < 1 \text{ ; partial mixing} \\ & FMXI = 1 \text{ ; perfect mixing} \end{aligned} \tag{7}$$

(2) Mixing in the Reactor Vessel Upper Plenum

Mixing in the reactor vessel upper plenum is assumed to exist in the flow volume *V(22,1)*(refer to Figure 2.1-5). Partial mixing, if assumed, is simulated by taking the reactor vessel outlet flow partly from *V(22,1)* and partly from the flow volume *V(21,i)*. (These are the volumes above the top of the active core and below the outlet nozzle center.)

The user-defined mixing factor *FMXO* for the upper plenum is:

$$\begin{aligned} FMXO &= 0 \text{ ; no mixing} \\ 0 < FMXO &< 1 \text{ ; partial mixing} \\ FMXO &= 1 \text{ ; perfect mixing} \end{aligned} \tag{14}$$

2.1.3.3 Reactor Coolant Pump and Flow Transient Model

(1) Reactor Coolant Flow Transient Equations [Reference 11]
The fundamental flow transient equations are based on a momentum balance around each reactor coolant loop and across the reactor vessel, flow continuity, and the reactor coolant pump (RCP) characteristics with or without electrical power supply. The conservation equation for driving head, elevation head, and head losses is written for a multi-loop nuclear reactor system as follows:

$$H_{RV} + [H_{PUMP}]_i + [H_{LP}]_i + [H_{SG}]_i - \Delta P_{RV} - [\Delta P_{LP}]_i - \Delta P_{KE-RV} - [\Delta P_{KE-LP}]_i = 0 \quad , i=1 \text{ to } Nloop \qquad (22)$$

where
H_{RV}, H_{LP}, H_{SG} = head generated by the difference in elevation and density of fluid in reactor vessel, reactor coolant loop, and steam generator
H_{PUMP} = reactor coolant pump head
ΔP_{RV}, ΔP_{LP} = pressure loss in reactor vessel and reactor coolant loop, including reactor coolant pump and steam generator
Note; if H_{PUMP} is input, ΔP_{LP} does not include pressure loss by RCP.
ΔP_{KE-RV}, ΔP_{KE-LP} = pressure head developed by fluid kinetic energy in reactor vessel and reactor coolant loop (including contributions from reactor coolant system pipes, pumps, and steam generator)

The RCP head, H_{PUMP}, is derived from the homologous curves of the RCP.

When the electrical power supply to a RCP motor is lost, the pump head, H_{PUMP}, is eventually lost. If the RCPs in the other loops continue to operate, the flow in the loop with the stopped RCP is reversed due to the reversed head between the reactor vessel inlet and outlet nozzle of the loop caused by the H_{PUMP} associated with the operating RCPs. For this condition, the stopped RCP acts as a flow resistance and the RCP performance for the reversed flow has to be prepared.

If all RCPs stop, the forced coolant flow eventually changes to natural circulation flow, if the elevation heads, H_{RV}, H_{LP}, H_{SG}, are developed. It should be noted that the head generated in the steam generator U-tubes is conservatively recommended not to be included because the head in each U-tube could be unstable.

The pressure heads developed by the fluid kinetic energy are expressed as:

$$\Delta P_{KE-RV} = \frac{1}{g}\sum\left(\frac{L}{A}\right)_{RV} \cdot \frac{dW_{RV}}{dt} \qquad (23)$$

$$[\Delta P_{KE-LP}]_i = \left[\frac{1}{g}\sum\left(\frac{L}{A}\right)_{LP} \cdot \frac{dW_{LP}}{dt}\right]_i \quad , i=1 \text{ to } Nloop \qquad (24)$$

where
g =acceleration due to gravity

$\left(\frac{L}{A}\right)_{RV}$, $\left(\frac{L}{A}\right)_{LP}$ = length of flow path over cross-section area of flow path for reactor vessel and loop.

W_{RV}, W_{LP} = fluid flow rate in reactor vessel and in RCS loop

The pressure drop in the reactor or other sections of the coolant loops vary as a function of flow squared and the density. If the nominal full-flow conditions are known, the pressure drops are written as:

$$\Delta P_{RV} = \Delta P^0_{CORE}\left(\frac{W_{CORE}}{W^0_{CORE}}\right)^2\left(\frac{\rho^0_{CORE}}{\rho_{CORE}}\right) + \Delta P^0_{RVI}\left(\frac{W_{RVI}}{W^0_{RVI}}\right)^2\left(\frac{\rho^0_{RVI}}{\rho_{RVI}}\right) + \Delta P^0_{RVO}\left(\frac{W_{RVO}}{W^0_{RVO}}\right)^2\left(\frac{\rho^0_{RVO}}{\rho_{RVO}}\right) \quad (25)$$

$$[\Delta P_{LP}]_i = [\Delta P_{HL} + \Delta P_{SG} + \Delta P_{CL} + \Delta P_{PUMP}]_i = \left[\sum_j \Delta P^0_j\left(\frac{W_j}{W^0_j}\right)^2\left(\frac{\rho^0_j}{\rho_j}\right)\right]_i$$

$$, J=HL,\ SG,\ CL,\ PUMP,\quad i=1 \text{ to } Nloop \quad (26)$$

where

superscript: ^{0} denotes nominal value

subscripts: $_{CORE,\ RVI,\ RVO,\ HL,\ SG,\ CL,\ PUMP}$ = core, reactor vessel inlet plenum (downcomer), reactor vessel upper plenum, hot leg, steam generator, cold leg (including cross-over leg) and reactor coolant pump.

In Equation (26), $\left(\frac{W_j}{W^0_j}\right)^2$ is defined as having the same sign in the parenthesis in order to take account for the direction of the flow.

It should be noted that the pressure losses in Equations (25) and (26) are computed by mass flow squared compensated by the density, which is acceptable for turbulent flow, but is underestimated for lamina flow. Pressure loss corrections using Reynolds number are available in the code as an option (see Section 2.1.4 Realistic Model).

From basic conservation laws, the sum of the loop flows must equal the reactor vessel flow.

$$W_{RV} = \Sigma W_{LP} \quad (27)$$

For the steady state condition, the reactor coolant flows are determined by Equations (22) to (27), assuming $\Delta P_{KE} = 0$.

(2) Reactor Coolant Pump Model

The sum of various torques in the RCP must equal the pump motor torque:

$$T_{KE} + T_H + T_W + T_R = T_M \quad (28)$$

Where

T_{KE} = pump kinetic torque

T_H = pump hydraulic torque

T_W = pump windage and friction torque

T_R = retardation torque due to eddy current in the stator of RCP

T_M = pump motor torque

The pump motor torque, $T_{M,}$ is given by the speed-torque curve of the pump motor. The torque generated by an induction motor is a function of the difference between motor speed and the synchronous speed.

Flow Coastdown

When the electrical power to a pump motor is interrupted, the motor torque, T_M in Equation (28), becomes zero and a flow coastdown results, which is characterized by a decreasing reactor coolant flow in each affected loop.

$$T_{KE} + T_H + T_W + T_R = 0 \qquad (29)$$

In the fundamental Equation (22), the reactor coolant pump head, H_{PUMP}, decreases according to the decrease of the pump speed.

During the flow coastdown the pump head is generated by the inertia of the pump and the coupling between the pump and the system fluid is considered.

The kinetic energy of rotating parts of the reactor coolant pump is:

$$KE = \frac{1}{2g} I_P \varpi^2 \qquad (30)$$

where

I_P = moment of inertia of rotating parts of reactor coolant pump

ϖ = angular speed of rotating parts of reactor coolant pump

The kinetic energy is dissipated into several losses and the power developed by the inertia is given by differentiating Equation (30). Dividing by the speed gives the total torque developed by the depletion of the kinetic energy.

$$T_{KE} = \frac{1}{g} I_P \cdot \frac{d\varpi}{dt} \qquad (31)$$

The hydraulic torque is defined as

$$T_H = \frac{W \Delta P_P}{\varpi \eta \rho} \qquad (32)$$

where

η = hydraulic efficiency of reactor coolant pump

ρ = fluid density

The pump head H_{PUMP} is determined from head versus flow characteristics of the pump which depends on the pump speed. The effect of a change in pump speed on the head-flow curve is defined according to the following affinity law:

$$H_{PUMP} = f_{HF}(\varpi, W) \qquad (33)$$

The above equations can be solved exactly if the pump characteristics, reactor system pressure losses, friction, windage and retardation torques are known.

Pump Motor Power Frequency Decay

If the frequency of the pump power decays, the motor torque, T_M in Equation (28), decreases depending on the pump motor speed-torque characteristics, causing decrease of the pump speed and the pump hydraulic torque. Because the RCPs use synchronous AC motors, the reactor coolant flow decreases by about the same rate as the pump speed and frequency decay. The frequency decay rate is determined from the electrical network strength against failures in some of the power generating stations in the network.

Reactor Coolant Pump Locked Rotor

If a RCP rotating part is locked and the rotation of the impeller instantly stops, the reactor coolant pump hydraulic torque is lost and the loop coolant flow rapidly decreases. Eventually the loop flow is reversed due to the head of the intact reactor coolant pump in the other loops. The flow change is calculated using the reactor coolant pump hydraulic characteristics with the rotor locked. In such a case, the reactor coolant flow in the core also decreases rapidly causing a rapid reduction in core heat removal.

All the models described by the above equations have been incorporated in the MARVEL-M code and coupled with the reactor coolant system models. The code can compute flow transients from the various causes, allowing different flows in up to four loops. If all the reactor coolant pumps stop, the flow transient proceeds to natural circulation condition continually.

The flow transients are integrated with the other nuclear and thermal-hydraulic performance. The flow models are applicable to the reactor transient analysis for partial loss of flow, complete loss of flow (including due to pump motor power frequency decay) and locked rotor.

(3) Natural Circulation Elevation Head Model

When the electrical power to a pump motor is interrupted, the reactor coolant pump torque becomes zero and the reactor flow coasts down and the pump eventually stops rotating.

If the elevation heads, H_{RV}, H_{LP}, H_{SG}, are developed, natural circulation flow is established. The natural flow conditions are calculated by the overall balance equation, Equation (22), using the relevant equations for the associated variable terms.

The driving forces in the case of natural circulation are calculated from the difference in the fluid densities around the circuit as follows:

$$H_{RV} = \int_{RVI}^{RVO} \rho \cdot dz = (Z_{CORE} + Z_{PL})\rho_{RVI} - Z_{CORE}\rho_{CORE} - Z_{PL}\rho_{RVO} \quad (34)$$

$$[H_{LP}]_i = \int_{RVO}^{RVI} \rho \cdot dz = [Z_{SG}(\rho_{SGO} - \rho_{SGI}) + Z_{TUBE}(\rho_{SGC} - \rho_{SGH}) + Z_{CL}(\rho_{CL}^{SG} - \rho_{CL}^{RCP})]_i \quad , i=1 \text{ to } Nloop \quad (35)$$

where

Z_{CORE} = core height (active fuel region)

Z_{PL} = height of reactor vessel outlet nozzle centerline above top of core (active fuel region)

Z_{SG} = height of steam generator tube sheet above the hot leg centerline

Z_{TUBE} = height of average steam generator U-tube

Z_{CL} = height of reactor coolant cross-over piping (cold leg reactor vessel inlet nozzle centerline above bottom of cross-over centerline)

ρ_{RVI}, ρ_{RVO} = fluid density in reactor vessel inlet plenum (downcomer) and upper plenum, respectively

ρ_{CORE} = average fluid density in core

ρ_{SGI} = average of fluid density in hot-leg piping, rising part to SG, and fluid density in SG hot leg side plenum

ρ_{SGO} = average of fluid density in SG cold leg side plenum and cold-leg piping from SG to the level of reactor vessel inlet nozzle centerline

ρ_{SGC} = average fluid density in steam generator cold leg side tubes

ρ_{SGH} = average fluid density in steam generator hot leg side tubes

ρ_{CL}^{SG}, ρ_{CL}^{RCP} = fluid density in the cross-over leg. Superscripts of SG, RCP denote steam generator side and reactor coolant pump side, respectively

(4) Solution of flow transient equations

Equations (22) with the relevant equations for the various associated variable is reduced to the following set of simultaneous equations for changes in the loop flow,

$$\frac{1}{g}\left(\frac{L}{A}\right)_{LP}\cdot\left[\frac{\Delta W_{LP}}{\Delta t}\right]_i+\frac{1}{g}\left(\frac{L}{A}\right)_{RV}\cdot\frac{\Delta W_{RV}}{\Delta t}$$

$$=\left[H_{PUMP}\right]_i+H_{RV}+\left[H_{LP}\right]_i+\left[H_{SG}\right]_i+\Delta P_{RV}+\left[\Delta P_{LP}\right]_i \quad ,\ i=1\ to\ Nloop \qquad (36)$$

When the reactor coolant pump in a reactor coolant loop is running, the head of the reactor coolant pump $[H_{PUMP}]_i$ is calculated from Equations (29) to (33). If some pumps are not running, the idle reactor coolant loop flow for the loop associated with the idle pump is reversed, and the pump head is replaced with a pressure loss. When all the reactor coolant pumps are not operating, all the pump heads, $[H_{PUMP}]_i$ are replaced with pressure losses and the reactor coolant flow transitions to natural circulation. The natural circulation flow in the multiple loops depends primarily on the power generation in the reactor core and the heat removal in the loops at the steam generators. The flow transition from the forced circulation to the natural circulation is calculated using Equation (36).

In solving Equation (36) iterative computations are performed to satisfy all the relevant equations. All anticipated and postulated reactor coolant system flow transients are computed by Equation (36) with the boundary conditions specified in the input data.

2.1.3.4 Secondary Steam System Model (4-Loop Model)

(1) Distribution of Steam Flows

The main steam lines from each steam generator are connected together at a common steam header, each via an isolation valve and a check valve, as illustrated in Figure 2.1-6.

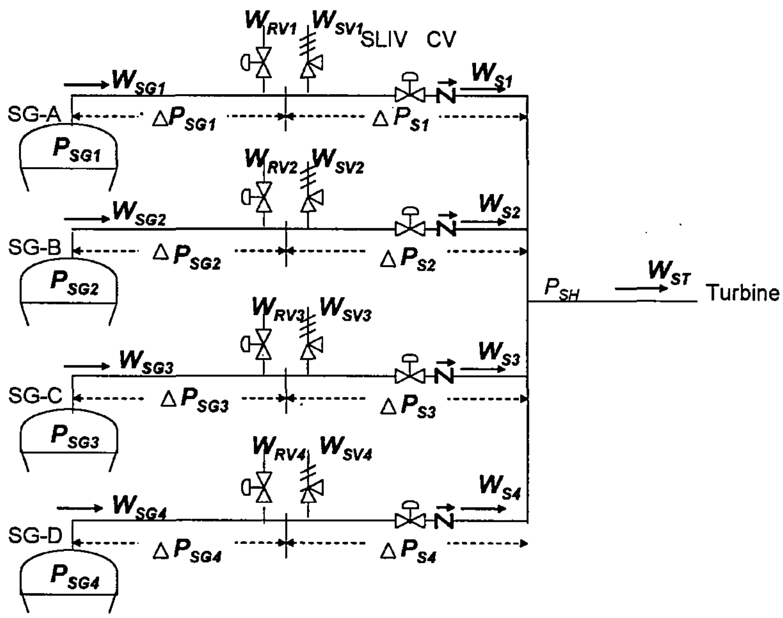


Figure 2.1-6. Steam Line Models

If the operating conditions of the steam generators are different from each other, the steam outputs from the steam generators are unbalanced. The steam flow distribution is then dependent upon the steam pressure of each steam generator and upon the pressure losses through the steam lines. The steam flow distribution can be obtained by solving the following basic equations:

$$\left[P_{SG} + \Delta P^{0}_{SG} \cdot (\overline{W}_{SG} + \overline{W}_{RV} + \overline{W}_{SV})^2 + \Delta P^{0}_{S} \cdot (\overline{W}_{S})^2\right]_i = \left[P_{SH}\right]_i \quad , i=1 \text{ to } Nloop \tag{37}$$

$$\overline{W}_{ST} = \sum_{i=1,N} \left[\overline{W}_S\right]_i \tag{38}$$

$$\left[P_{SH}\right]_i = P_{SH} \tag{39}$$

where $\overline{W}_{SG}, \overline{W}_{RV}, \overline{W}_{SV}$ are steam generator, relief valve and safety valve mass flow rates normalized by the rated loop steam flow rate.

To solve the four non-linear equations of (37) for a multi-loop plant, the following predictions of the steam generator pressures at the next time step are introduced for stable computation, since changes in steam flow change steam generator pressures and vice versa.

For a 3- and 4-loop plant, simultaneous second-order equations are solved by iterative computation to converge the entire solution.

Steam line header pressures are obtained using the following equation:

$$[P_{SH}]_i = \left[P_{SG} + \Delta P^0_{SG} \cdot (\overline{W}_{SG} + \overline{W}_{RV} + \overline{W}_{SV})^2 + \Delta P^0_S \cdot (\overline{W}_S)^2\right]_i \tag{41}$$

When $[P_{SH}]_i$ becomes higher than the set pressure of the secondary relief valves P_{SRV} or secondary safety valves P_{SSV}, the valves open and steam is relieved. As long as the relief or safety valve capacity is sufficient, the pressure at the relief or safety valve is maintained at their set pressures with pressure accumulation based on user input for the valve characteristics.

$$[P_{SH}]_i = \left[P_{SV} + \Delta P^0_S \cdot (\overline{W}_S)^2\right]_i \tag{42}$$

Relief and Safety valve flows are computed from:

$$[P_{SV} - P_{SG}]_i = \left[\Delta P^0_{SG} \cdot (\overline{W}_{SG} + \overline{W}_{RV} + \overline{W}_{SV})^2\right]_i \tag{43}$$

If some of the steam line isolation valves are closed, the steam flow becomes zero and the steam flow to the turbine W_{ST} is taken from the steam generators with the steam isolation valve open.

Note:

In solving equations (42) to (43), pressure losses $[\Delta P_S]_i$ (between the valve connection and common header) are currently included in $[\Delta P_{SG}]_i$ since most of the line pressure losses are dominated by the pressure losses at the steam generator exit nozzle with integral flow restrictors and the pressure losses are determined by $[\Delta P_{SG}]_i + [\Delta P_S]_i$ when the safety and relief valves are closed. When common header pressure P_{SH} becomes higher than the set pressure of the safety valves P_{SSV} or relief valves P_{SRV}, the safety or relief valve is opened. This is conservative for the maximum steam line pressure. Once the valve is opened, flow from the valve to the common header is greatly reduced due to the reduction of pressure losses $[\Delta P_S]_i$.

(2) Steam Safety and Relief Valves
Multiple safety valves with slightly different set pressures are provided in each steam line. The simulation of those safety valves includes a maximum of three valves with different set pressures and the valve pressure accumulation when opened.

Relief valves can be controlled by automatic proportional controllers to maintain steam pressure at the set-point. Manual control is also simulated. Steam released from safety and relief valves is released to the atmosphere. The MARVEL-M code has the capability to integrate atmospheric relief flow for use in the radiological assessment of certain accidents.

2.1.3.5 Other Model Refinement

MARVEL-M contains some other additional refinements beyond what has been described in the previous sections. Items (1) through (4) are MHI refinements made in the 1970s, while item (5) is a refinement of the numerical solution methodology adopted in the MARVEL-M code.

(1) Pressurizer Surge Line Model
A flow section has been added in the pressurizer surge line to the original MARVEL code between the reactor coolant system hot leg connection and the pressurizer. This is to more realistically model pressurizer insurge water enthalpy. If the pressurizer surge line is not simulated, hot leg coolant water directly enters the pressurizer during pressurizer insurge. This may result in overpredicting cooling of the pressurizer liquid phase and may cause a larger pressurizer pressure reduction for a subsequent outsurge. This refinement resulted from the observation of a transient test during a reactor plant preoperational test.

(2) Hot-Spot Fuel Thermal Kinetics Model
A hot-spot fuel thermal kinetics model is provided in the original MARVEL code. The model was similar to the fuel thermal kinetics model for the average channel.

A more detailed hot-spot fuel thermal kinetics model is included in the MARVEL-M code. The basic model is the same as the FACTRAN code by Westinghouse, which was approved by the NRC [Reference 10]. The FACTRAN code has the ability to model of up to 10 radial sections in the fuel pellet, cladding and clad surface heat transfer coefficient to compute the transient fuel temperature and heat flux. FACTRAN also has the capability to handle post-DNB transition film boiling heat transfer, Zircaloy-water reaction, and partial melting of the pellet material. The use of the model added to MARVEL-M is limited to the computation of the heat flux transients at the surface of the cladding at a hot-spot. The normalized hot-spot heat flux can be used as an option (the largest heat flux between the average channel and the hot spot is used) to calculate DNBR using the simplified DNBR model in MARVEL-M. The fuel pellet thermal properties can be input by the user.

(3) Core Void Simulation
Boiling can occur in the reactor core when the core power increases excessively or if the core coolant temperature exceeds the saturation temperature. The void causes insurge to the pressurizer, resulting in an increase in pressurizer pressure. The MARVEL-M code has an internal model to calculate the void fraction in the core. The MARVEL-M code has added a scheme to accept void transients calculated by an external detailed thermal-hydraulic code, which can compute void formation taking into account sub-cooled boiling, detached boiling, as well as bulk boiling. The VIPRE-01M code can be used for that purpose. This feature is

only used to assure that the RCS pressure is conservatively high for the rod ejection accident where local void formation in the core could impact the peak pressure.

(4) Feedline Break Blowdown Simulation
Licensing feedline break analysis uses the water release rate computed by the Moody correlation. During the water release, when the steam generator is depressurized below the feedwater saturation pressure, feedwater contained in the feedline flashes and a mixture of steam and water can be released into the steam generator shell side. This phenomenon is simulated by a flow section connected to the steam generator secondary side.

(5) Conversion of Reactor Coolant System Volume Balance by Pressure Search

2.1.4 Realistic Models

The original MARVEL code was developed for transient and accident analysis (excluding LOCA events) and control system design studies for pressurized water reactor plants. The models are sufficiently accurate for the purposes of design and licensing safety analysis. The code has also been used for various other applications, such as analysis for operating instruction development, support for plant transient tests (start-up tests), and post-event analysis of operating plant events (e.g. the steam generator tube rupture event at Mihama Unit 2 in 1993). Through those analyses the MARVEL code has been refined and various models have been added in order to simulate real plant transient behavior. Selected model refinements for these realistic analyses are described in this section, although they are typically not used for licensing safety evaluation of reactor plants. Use of these models is optional and is controlled by user input in the MARVEL-M code.

2.1.4.1 Steam Generator Tube Rupture

A steam generator tube rupture event (a single double ended tube rupture) occurred at the Mihama Unit No.2 (a 2 loop PWR plant designed by Westinghouse and constructed by Mitsubishi for Kansai Electric). During the actual event many systems were actuated and operated (e.g., Low Pressurizer Pressure reactor trip, Low-Low Pressurizer Pressure Safety Injection, manual actions for reactor coolant system (RCS) cooldown using a secondary relief valve, depressurization of the RCS by auxiliary pressurizer spray, termination of the SG leak in the failed SG by termination of SI, etc.). Many reactor system transient behaviors were also observed, including emptying the pressurizer, recovery of pressurizer water level, natural circulation in the primary loops, the failed SG secondary side transients such as steam pressure increase to actuate steam relief valves repeatedly, and SG water level increase. The reactor plant was safely shut down without significant radioactivity release.

(1) Event analysis by MARVEL-M code
The event was analyzed using the MARVEL-M code to ascertain the adequacy of the reactor system operation and to aid in the response to detailed regulatory questions and examination.

The MARVEL-M code and the models for SGTR event were evaluated and verified to be able to analyze such an accident with sufficient accuracy.

(2) Realistic SGTR Models
To obtain better agreement between the MARVEL-M analysis and the trend records of the event several realistic models were developed and added to the MARVEL-M code during the post-accident analysis of the event.
The essentially important models are:
- Pressure transient after pressurizer emptied and water level recovery
- Failed SG secondary response (SG steam pressure, water level, etc.)
- Tube leak model

(a) Reactor Vessel Upper Head Model
After the pressurizer is emptied RCS pressure is maintained by the stagnant fluid in the upper head volume, where vapor phase is formed and acts as an alternate pressurizer. High Pressure Safety Injection also acts to maintain the system pressure. Essential models of the upper head are already incorporated in the code as described in this document. To obtain closer agreement between the reactor coolant system pressure and the pressurizer level transient, the following realistic models were investigated and added to the MARVEL-M code. Some Japanese PWRs used to be designed to trip the electric cross-tie breakers between the non-safeguards busses and the safeguard busses on a Safety Injection signal. That causes loss of power supply to the reactor coolant pumps, resulting in natural circulation flow.

i) During natural circulation, flow through the cooling spray nozzle in the upper head may exist due to elevation heads caused by temperature differences between the downcomer and reactor vessel upper plenum.
ii) Condensing heat transfer may occur between the vapor phase and the upper head metal and liquid in the upper head volume.

(b) Realistic steam generator tube leak flow model
The following conservative realistic model to compute SGTR leak flow has been added to the MARVEL-M code. The initial primary-to-secondary leak flow at steam generator tube rupture is computed as the critical flow, and the leak flow transitions to orifice-type flow (not critical) later when the pressure difference between the primary and secondary sides is reduced. In the calculation the pressure losses along the tube from the tube inlet or outlet to the break point is included. A conservative, but realistic SGTR analysis can be performed with a user-defined discharge coefficient of 1.0 for critical flow.

(c) Reverse Heat Transfer Coefficient
When the steam generator shell-side temperature is higher than the primary side temperature, heat is transferred from the shell side to the primary side. A constant value for the reverse heat transfer coefficient can be input. An internal calculation can also be selected, which uses McAdam's correlation.

(d) The steam in the failed SG and steam line can be compressed because of the increase in SG level due to leakage from the primary side. Steam in the steam line may be condensed by condensate heat transfer to the pipe wall.

(e) Coolant leaking from the primary side may not be completely mixed with the secondary water. A two-node model for the water portion of the steam generator secondary is added to account for cooling of a portion of the steam generator water when the RCS temperature is below the steam generator temperature.

2.1.5 Precautions and Limitations for the Use of the MARVEL-M Code

2.1.5.1 Range of Operating Variables

The program is designed to be run within the following ranges of operating variables.

- Reactor Coolant System Temperature and Pressure
  Temperature : 50°F to approximately1100°F
  Pressure : 50 psia to critical pressure (about 3200 psia)
- Pressurizer Water Level
  From empty to full including water discharge (After the pressurizer is emptied, the program may analyze the system behavior until the coolant in the reactor vessel inactive volume (dead volume) is boiled off.)
- Steam Generator
  Steam Pressure : 14 psia to 1500 psia
  Water Inventory : Empty to moderately high level
- Reactor Coolant Loop Flow
  Forward, reverse and natural circulation flows are computed. Two phase flows are also permitted as a homogeneous equilibrium mixture of vapor and liquid
- Reactor Core Kinetics
  Reactor power : neutron source level to overpower level
  Reactivity : sub-critical to super-prompt critical.

The program is intended to cover a very wide range of operating parameters. However, when the plant operating variables deviate excessively from the normal operating conditions, care must be used in interpreting the results in context with the accuracy and limitations of the code models over the regions where the variables are extreme.

2.1.5.2 Applicability of the Code to the Scenarios of Licensing Analysis

The MARVEL-M code is used for multiple transients and accidents. The code is provided with the models for most of the scenarios in the design basis transients and accidents for pressurized water reactor plants except for Loss of Coolant Accident. However, other appropriate codes should be used for specific transients and accidents in part or as a whole, since the following models are not sufficiently detailed for certain specific transients.

- Space independent one point neutron kinetics equations are used.

- A simplified DNBR calculation is modeled, but detailed DNBR calculation should be performed by an external code.
- Two phase flows in the reactor coolant system are modeled assuming homogeneous equilibrium mixture of vapor and fluid, except for the pressurizer and the upper head volume where vapor and liquid are separated and not at equilibrium.

The following events should not be evaluated with MARVEL-M. The use of another appropriate code is recommended.

(1) Transients

(a) Transients that are classified as reactivity initiated events (RIE), e.g. Inadvertent Rod Withdrawal from Sub-critical Condition, should be analyzed by a code developed for the specific purposes. (The TWINKLE-M code is used for the US-APWR.)

(b) Transients for which the minimum DNBR is heavily dependent on changes in reactor coolant flow. For example, the Loss of Flow should use a thermal-hydraulic code which can compute the local fuel kinetics and DNBR correctly using the output of plant operating variables by MARVEL code. (The VIPRE-01M code is used for the US-APWR.)

(2) Accidents

(a) Reactivity Initiated Accidents, e.g. RCCA Ejection, should be analyzed using a spatial neutron kinetics code. (TWINKLE-M code is used for the US-APWR.)

(b) DNBR calculations for the large steam line break from a shutdown or hot standby condition should be calculated by an appropriate external thermal-hydraulic code with capability of computing DNBR, in conjunction with a spatial neutron kinetics code if a large transient distortion of the flux distribution is to be taken into account.

(c) LOCA should be analyzed by LOCA codes.

(3) Conservatism in Models

Safety analysis for Chapter 15 for a safety analysis report of a reactor plant should be performed with adequate conservatism to assure the safety of the reactor plant for Anticipated Operational Occurrences (AOOs) and Postulated Accidents (PAs). The conservatism or safety margins should be assured by the computer models, assumptions and safety criteria, and input data used for the analysis, depending on the scenarios of the transients and accidents.

The MARVEL-M code may be regarded as a code between best estimate (BE) and evaluation model (EM). Key conservatisms in the models are:

(a) The pressurizer pressure calculation is based on the isentropic process of the vapor phase for short-term that gives conservatively higher pressure increase for insurge transients.

(b) The steam generator secondary side thermal model is based on equilibrium of the vapor and liquid at saturation, neglecting the sub-cooling in the downcomer and the preheat region. This model is generally conservative: i.e. sub-cooled water, if modeled, could absorb some energy following a loss of load, loss of normal feedwater flow and feedline break and also for a SG tube rupture accident.

(c) MARVEL-M computes the pressurizer pressure and the reactor coolant system pressure at the connection of the pressurizer surge line. The pressure differences between the surge line connection, core, and the maximum pressure point (usually at the discharge of the reactor coolant pump) are corrected at each time step by adding a conservative bias. For the purpose of calculating fluid properties, the RCS pressure is assumed to be constant around the RCS loop, because the RCS coolant is subcooled in a PWR.

(d) The model of the reactor coolant mixing in the reactor vessel allows conservative mixing by selecting the mixing factors bounding the best estimation based on the experimental data.

The MARVEL-M code may be regarded as a realistic conservative evaluation code as a whole. The transient and accident analyses performed using the MARVEL-M code are expected to give sufficiently conservative results by using conservative assumptions and conservative values of plant data depending on the scenarios of the transient and accident.

2.2 TWINKLE-M Code

The TWINKLE-M code is the multi-dimensional spatial neutron kinetics code which solves two-group transient diffusion equations using a finite-difference technique. The code uses six delayed neutron groups and contains the detailed fuel-clad-coolant heat transfer model for calculating mesh-wise Doppler and moderator feedback effects. The code is used to predict the kinetic behavior of a reactor for the transients that cause a major perturbation in the spatial neutron flux after steady state initialization.

Aside from basic cross-section data and thermal-hydraulic parameters, the code accepts the following types of basic input parameters: inlet temperature, pressure, flow, boron concentration, control rod motion, and others. Various outputs are produced (for example, channel-wise power, axial offset, enthalpy, volumetric surge, point-wise power, and fuel temperatures).

The original TWINKLE code was approved by the NRC (WCAP-7979-P-A) [Reference 3] as a multi-dimensional neutron kinetics analysis code. The code was licensed to MHI under the licensing agreement between Westinghouse and MHI. Since then the code has been applied to licensing analysis for Japanese PWRs. At the beginning, a conservative methodology based on one-dimensional kinetics with the assumption of a constant hot channel factor during transient was applied to the RCCA ejection for fuel enthalpy evaluation.

In 1993 the fuel failure threshold for the RIE (Reactivity Initiated Event) for Japanese PWRs was required to be lowered and expressed as a function of local fuel burnup. This change was in response to the results of the RIE fuel failure testing at NSRR (Nuclear Safety Research Reactor) experiments. In order to comply the new threshold, MHI introduced a more realistic methodology for the RCCA ejection from hot zero power condition. This methodology employed a time-dependent hot channel factor based on the three-dimensional kinetics model in the TWINKLE code. The maximum number of spatial mesh points had to be increased to allow a full three-dimensional core representation. In addition, a discontinuity factor consistent with a core simulator ANC code [Reference 12] was incorporated to present the local power distribution more accurately. This new version of TWINKLE is now referred to as TWINKLE-M.

MHI has used the TWINKLE-M three-dimensional kinetics code to validate the use of a one-dimensional model for plant licensing in Japan. This is supported by the following:

- The TWINKLE code was originally approved by the NRC as a multi-dimensional kinetics code.
- The TWINKLE three-dimensional static calculation is in good agreement with the ANC.
- The TWINKLE three-dimensional kinetics is used as a reference solution for a recent nodal kinetics code SPNOVA developed by Westinghouse. The SPNOVA code was approved by the NRC [Reference 13].

MHI has had significant experience in using the 1-D and 3-D capabilities of the TWINKLE-M code over many years.

Section 3 presents validation of the three-dimensional calculation. This information supports the use of TWINKLE-M 1-D and 3-D capabilities for licensing new reactors in the US.

2.3 VIPRE-01M Code

VIPRE-01M is the MHI version of VIPRE-01, which is a subchannel analysis code that is developed to perform thermal-hydraulic analyses in reactor cores. Using the original VIPRE-01 code as the basis, MHI incorporated certain added functions for more flexible design applications. VIPRE-01M is used to evaluate reactor core thermal limits related to the minimum DNBR, reactor core coolant conditions, and fuel temperature and heat flux in normal and off-normal conditions.

The original version of VIPRE-01 was developed by Battelle Pacific Northwest Laboratories, under the sponsorship of Electric Power Research Institute (EPRI). Its basic components are from the well-known COBRA code series. VIPRE-01 divides the reactor core into a number of flow channels. The size of each flow channel could be as small as the flow area surrounded by four fuel rods (fuel rods and/or control rod guide thimble) situated on a square lattice, or be formed by a number of fuel rod bundles. Conservation equations of mass, momentum (in axial and lateral directions), and energy are solved to determine axial mass flux distributions, lateral flow rate per unit length, and enthalpy distributions. Fluid properties are functions of the local enthalpy and a uniform but time-varying system pressure. Transient thermal behavior of the fuel rod is also analyzed in association with the determined thermal-hydraulic analysis results.

Specific constitutive models which prescribe optional flow resistance, turbulent mixing, and subcooled as well as saturated boiling, are selected in VIPRE-01M analyses to provide adequate results for the purpose of the applications.

VIPRE-01M has incorporated mainly the following features into the original VIPRE-01.

- DNB correlations for design applications
- Fuel thermal properties for design applications
- Options for hot spot PCT analysis

The original solution methods and constitutive models are not changed at all. Therefore, the VIPRE-01M code is virtually identical to the original VIPRE-01. The conclusion of validation for the original VIPRE-01 code by EPRI still remains valid.

The details concerning calculation models, additional DNB correlation, fuel properties of the VIPRE-01M code and validation to transient analysis are described in Reference 6.

3.0 CODE VALIDATION

3.1 MARVEL-M Code

The MARVEL-M code is the same as the original MARVEL code from the viewpoint of constitutive and principal models. The main differences between the original MARVEL code and MARVEL-M are the extension from 2-loop simulation to 4-loop simulation and the addition of a built-in RCP model. The other refinements such as a pressurizer surge line node, a hot spot heat flux simulation model, and improved numerical solution and conversion techniques are described in Section 2.1.3.5.

This section provides a comparison of the calculated results between the MARVEL-M code and the 4-loop LOFTRAN* code in order to validate the adequacy of the modifications included in the MARVEL-M code. A code-to-code comparison is sufficient for this purpose because the LOFTRAN code has been used extensively in the licensing analysis of currently operating nuclear plants in the US for the accidents that are affected by the new MARVEL-M models.

*MHI has the source code, as well as sample input and output, for the 4-loop LOFTRAN code under a licensing agreement with Westinghouse.

The uncontrolled RCCA bank withdrawal at power has been chosen because both the LOFTRAN and MARVEL-M codes use simplified internal DNBR calculations to demonstrate the adequacy of the reactor protection system for uniform transients. The loss of flow accidents (partial loss of flow, complete loss of flow, and locked rotor) have been chosen because both LOFTRAN and MARVEL-M use an internal reactor coolant pump model for calculating the loop and total core flow transient. Comparison of other key parameters for these accidents such as nuclear power, core thermal power, RCS average temperature, and pressurizer pressure further confirm that the reactivity, pressure, and power models in the MARVEL-M code remain valid.

3.1.1 Uncontrolled RCCA Bank Withdrawal at Power

(1) Event Description
The Uncontrolled RCCA Bank Withdrawal at Power event initiates from nominal power operation. The event produces a positive reactivity insertion, and nuclear power increases until a reactor protection system setpoint is reached.

The Uncontrolled RCCA Bank Withdrawal at Power has been chosen because both the LOFTRAN and MARVEL-M codes calculate power and RCS parameters to demonstrate the adequacy of the reactor protection system for uniform transients. In this way, the transient validates the overall adequacy of the point kinetics model, fuel heat transfer model, and RCS thermal hydraulic model. Parameters of interest include reactor power, core average heat flux, RCS average temperature, and pressurizer pressure.

The maximum control rod insertion reactivity case (75 pcm/sec) is selected because it results in the maximum perturbation to the parameters of interest.

(2) Analysis Assumption
Analysis assumptions and calculation conditions in the typical 4-loop plant with a 17x17, 257 fuel assembly (17x17-257FA) core are as follows.

(a)	Initial condition	Nominal power, Nominal T_{avg}, Nominal RCS pressure
(b)	Reactor trip	118% nominal power
(c)	Insertion reactivity rate	75 pcm/sec
(d)	Feedback reactivity	Minimum feed back
(e)	Trip reactivity	-4%ΔK/K
(f)	Pressure control system	Off

(3) Results and Conclusions

Comparison results of MARVEL-M and LOFTRAN are shown in Figures 3.1.1-1 through 3.1.1-4. The results demonstrate that the two codes have equivalent capabilities and are in close agreement.

It is concluded that the MARVEL-M code is suitable for use in analyzing uniform non-LOCA transients assuming constant RCS flow that challenge the reactor protection system.

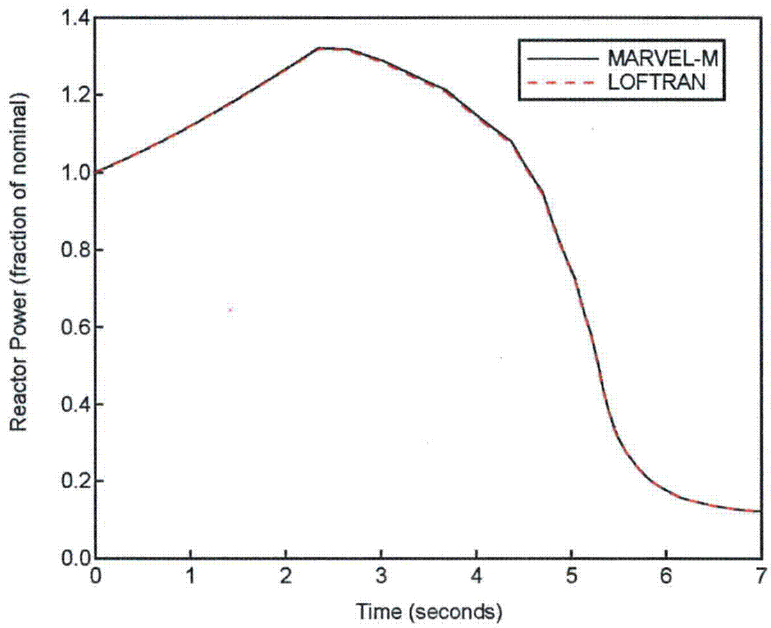


Figure 3.1.1-1 Reactor Power, Uncontrolled RCCA Bank Withdrawal at Power Comparison with MARVEL-M and LOFTRAN

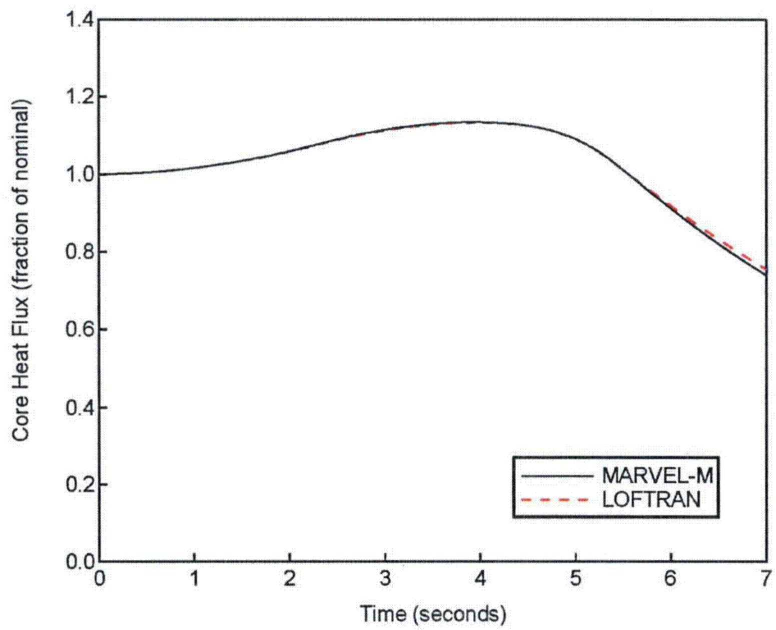


Figure 3.1.1-2 Core Heat Flux, Uncontrolled RCCA Bank Withdrawal at Power Comparison with MARVEL-M and LOFTRAN

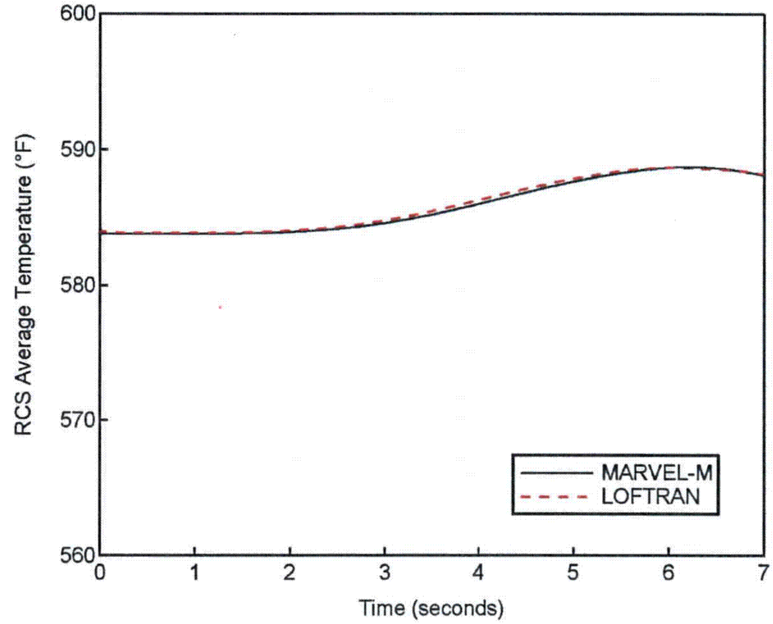


Figure 3.1.1-3 RCS Average Temperature, Uncontrolled RCCA Bank Withdrawal at Power Comparison with MARVEL-M and LOFTRAN

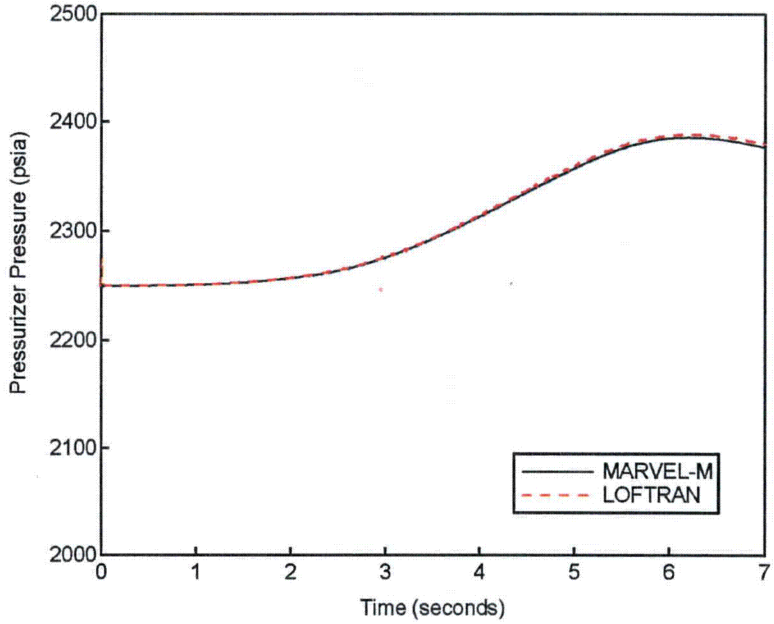


Figure 3.1.1-4 Pressurizer Pressure, Uncontrolled RCCA Bank Withdrawal at Power Comparison with MARVEL-M and LOFTRAN

3.1.2 Partial Loss of Forced Reactor Coolant Flow

(1) Event Description
The Partial Loss of Forced Reactor Coolant Flow event initiates from nominal power operation. The event includes cases where either one or two RCPs coast down, resulting a DNBR decrease due to core flow reduction until a reactor protection setpoint is reached.

The Partial Loss of Reactor Coolant Flow has been chosen because both the LOFTRAN and MARVEL-M codes use an internal reactor coolant pump model to calculate the loop and total core flow transients. In this way, the transient validates the adequacy of the MARVEL-M expansion from 2-loop to 4-loop simulation and the built-in RCP model. Two RCPs coasting down are analyzed in this case. Parameters of interest include reactor power, core average heat flux, loop flow rate, and pressurizer pressure.

(2) Analysis Assumption
Analysis assumptions and calculation conditions in the typical 4-loop plant with a 17x17-257FA core are as follows.

(a) Initial condition	Nominal power, Nominal T_{avg}, Nominal RCS pressure
(b) Reactor trip	Low Reactor Coolant Loop Flow
(c) RCP coast down number	Two RCPs
(d) Feedback reactivity	Minimum Density feedback and maximum Doppler feedback
(e) Trip reactivity	-4%ΔK/K

(3) Results and Conclusions
Comparison results of MARVEL-M and LOFTRAN are shown in Figures 3.1.2-1 through 3.1.2-4. The results demonstrate that the two codes have equivalent capabilities and are in close agreement.

It is concluded that the MARVEL-M code is suitable for use in analyzing a flow coastdown in one or more loops for the purpose of calculating time-dependent parameters input to the VIPRE-01M code (RCS flow rate and reactor power) for heat flux at the hot channel and DNBR calculations.

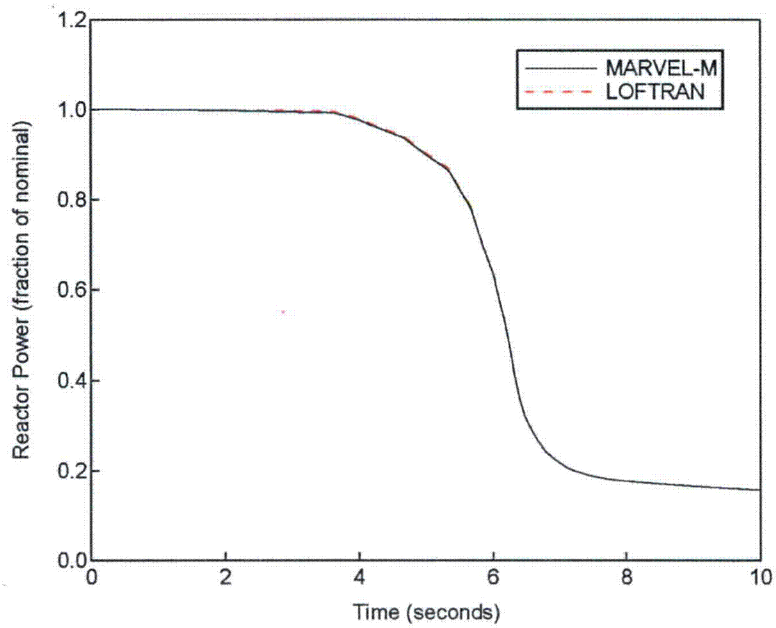


Figure 3.1.2-1 Reactor Power, Partial Loss of Forced Reactor Coolant Flow Comparison with MARVEL-M and LOFTRAN

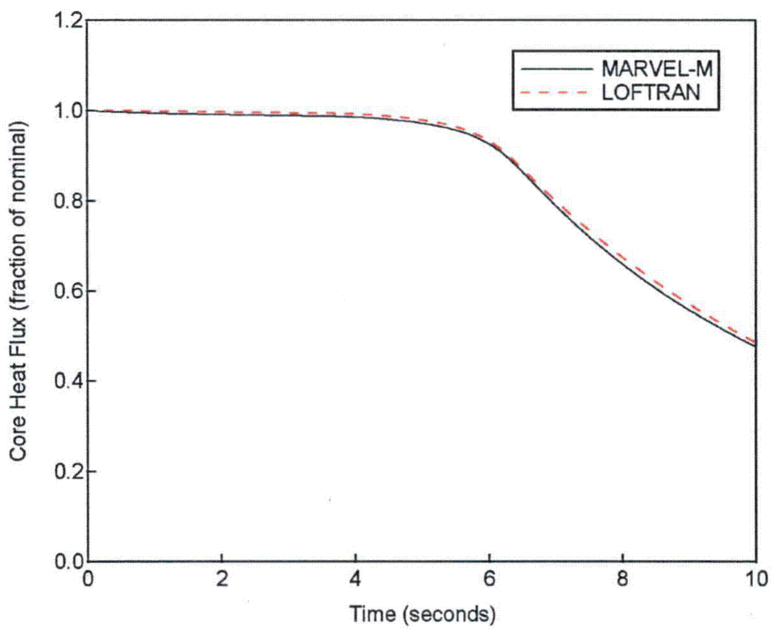


Figure 3.1.2-2 Core Heat Flux, Partial Loss of Forced Reactor Coolant Flow Comparison with MARVEL-M and LOFTRAN

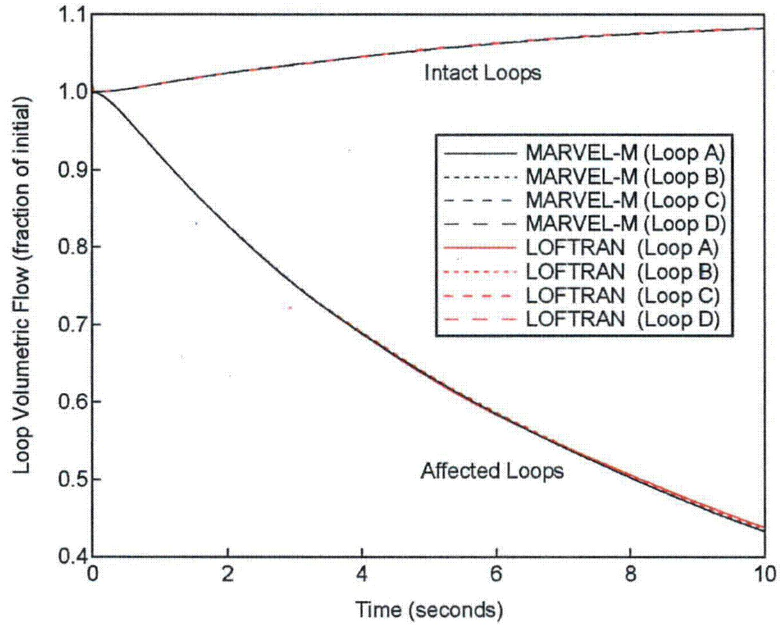


Figure 3.1.2-3 Loop Volumetric Flow, Partial Loss of Forced Reactor Coolant Flow Comparison with MARVEL-M and LOFTRAN

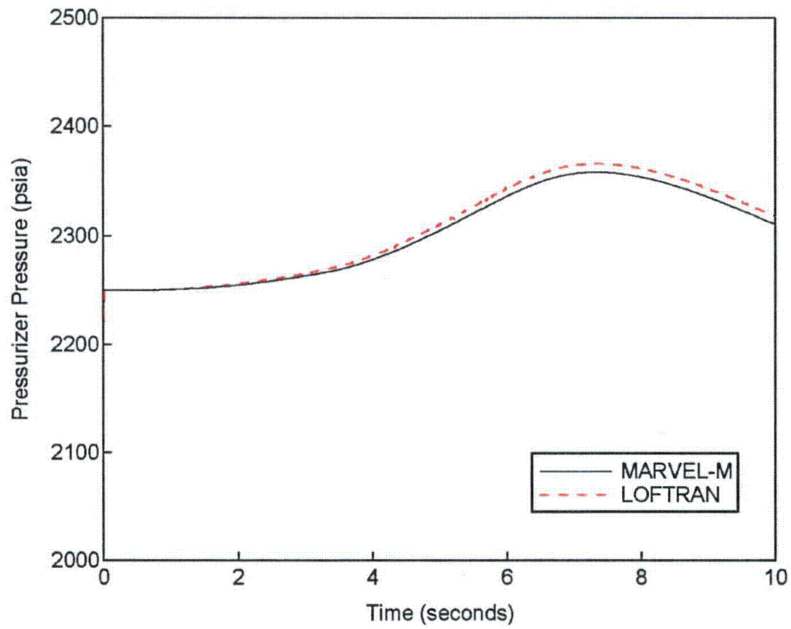


Figure 3.1.2-4 Pressurizer Pressure, Partial Loss of Forced Reactor Coolant Flow Comparison with MARVEL-M and LOFTRAN

3.1.3 Complete Loss of Forced Reactor Coolant Flow

(1) Event Description

The Complete Loss of Forced Reactor Coolant Flow event initiates from nominal power operation. All RCPs coast down and DNBR decreases due to core flow reduction until a reactor protection setpoint is reached.

The Complete Loss of Flow event has been chosen because both the LOFTRAN and MARVEL-M codes use an internal reactor coolant pump model to calculate the loop and total core flow. In this way, the transient validates the adequacy of the built-in RCP model for the purpose of calculating parameters used in calculating DNBR. The parameters of interest include reactor power, core heat flux, loop flow rate, and pressurizer pressure.

(2) Analysis Assumption

Analysis assumptions and calculation conditions in the typical 4-loop plant with a 17x17-257FA core are as follows.

(a) Initial condition	Nominal power, Nominal T_{avg}, Nominal RCS pressure
(b) Reactor trip	Low RCP speed
(c) RCP coast down	All RCPs
(d) Feedback reactivity	Minimum Density feedback and maximum Doppler feedback
(e) Trip reactivity	-4%ΔK/K

(3) Results

Comparison results of MARVEL-M and LOFTRAN are shown in Figures 3.1.3-1 through 3.1.3-4. The results demonstrate that the two codes have equivalent capabilities and are in close agreement.

It is concluded that the MARVEL-M code is suitable for use in analyzing a uniform flow coastdown in all loops for the purpose of calculating time-dependent parameters input to the VIPRE-01M code (RCS flow rate and reactor power) for heat flux at the hot channel and DNBR calculations.

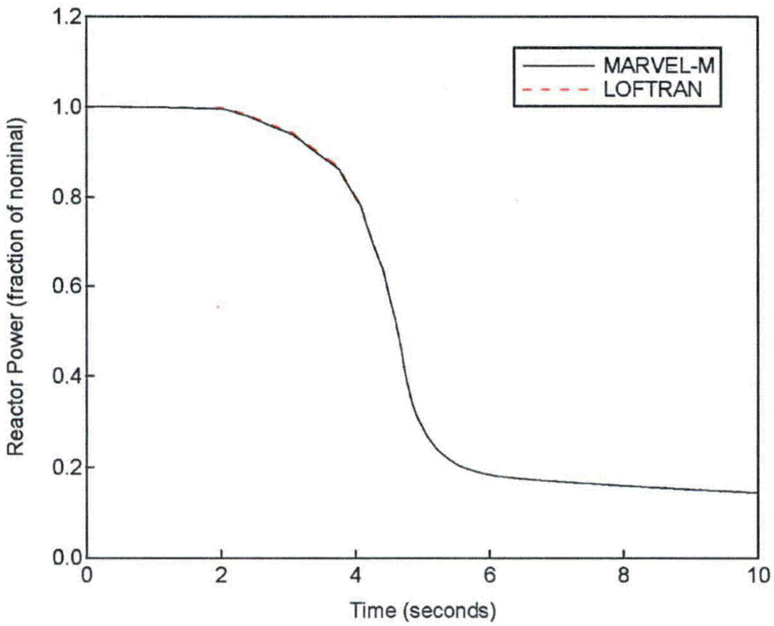


Figure 3.1.3-1 Reactor Power, Complete Loss of Forced Reactor Coolant Flow Comparison with MARVEL-M and LOFTRAN

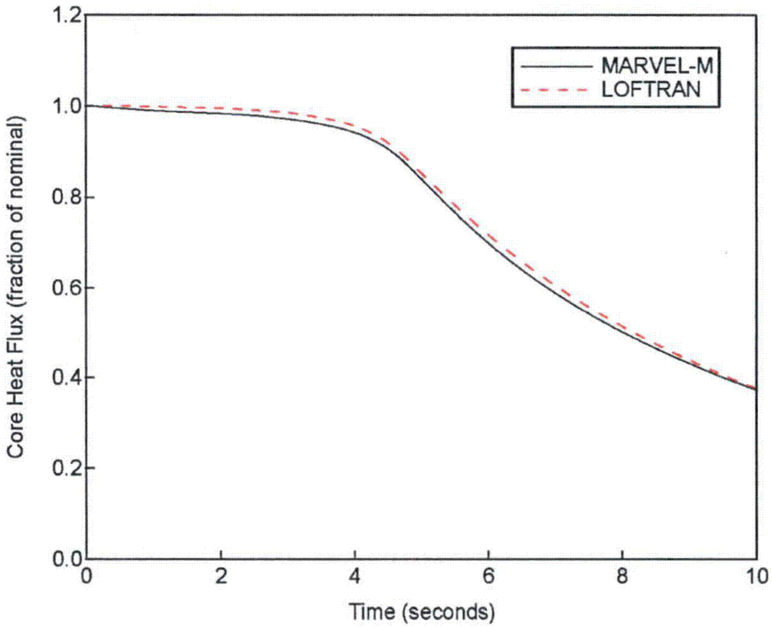


Figure 3.1.3-2 Core Heat Flux, Complete Loss of Forced Reactor Coolant Flow Comparison with MARVEL-M and LOFTRAN

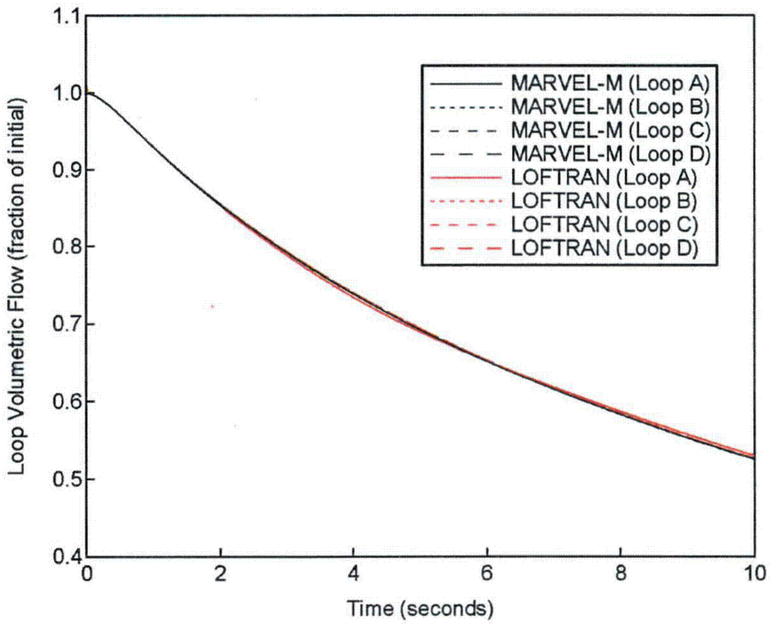


Figure 3.1.3-3 Loop Volumetric Flow, Complete Loss of Forced Reactor Coolant Flow Comparison with MARVEL-M and LOFTRAN

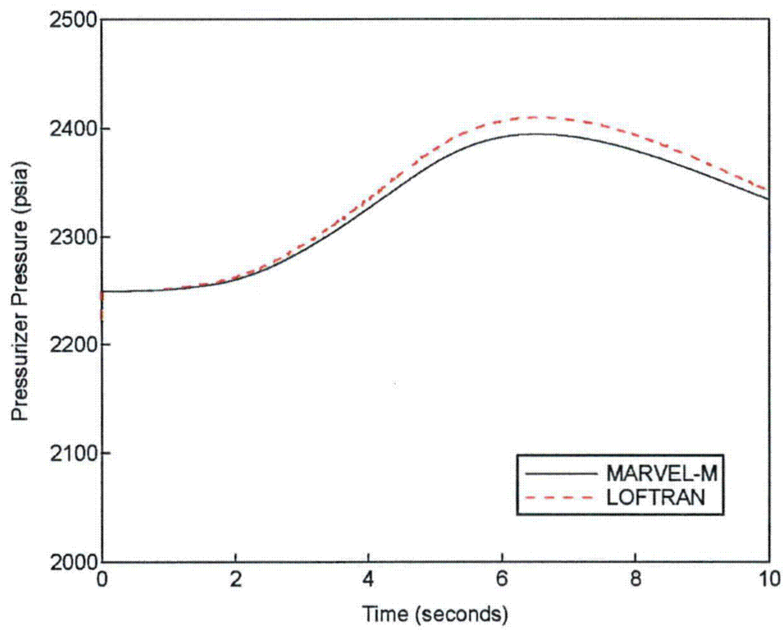


Figure 3.1.3-4 Pressurizer Pressure, Complete Loss of Forced Reactor Coolant Flow Comparison with MARVEL-M and LOFTRAN

3.1.4 Reactor Coolant Pump Shaft Seizure

(1) Event Description

The Reactor Coolant Pump Shaft Seizure event initiates from nominal power operation. In this event, one RCP shaft seizes (Locked Rotor) and DNBR decreases due to rapid flow reduction in the core until a reactor protection system setpoint is reached.

The Reactor Coolant Pump Shaft Seizure accident has been chosen because both the LOFTRAN and MARVEL-M codes use an internal reactor coolant pump model to calculate the loop and total core flow. In this way, the transient validates the extension from 2-loop to 4-loop simulation and the built-in RCP model. Parameters of interest include reactor power, core average heat flux, loop flow rate, and pressurizer pressure.

(2) Analysis Assumption

Analysis assumptions and calculation conditions in the typical 4-loop plant with a 17x17-257FA core are as follows.

(a)	Initial condition	Nominal power, Nominal T_{avg}, Nominal RCS pressure
(b)	Reactor trip	Low Reactor Coolant Loop Flow
(c)	RCP Shaft Seizure	One RCP locked at 0 seconds
(d)	Feedback reactivity	Minimum Density feedback and maximum Doppler feedback
(e)	Trip reactivity	-4%ΔK/K

(3) Results and Conclusions

Comparison results of MARVEL-M and LOFTRAN are shown in Figures 3.1.4-1 through 3.1.4-4. The results for loop flow rate and reactor power are in close agreement.

Pressurizer pressure of the MARVEL-M code is slightly lower than that of the LOFTRAN code due to the difference in average core heat flux. Although those minor differences exist, both codes have equivalent capability for this accident.

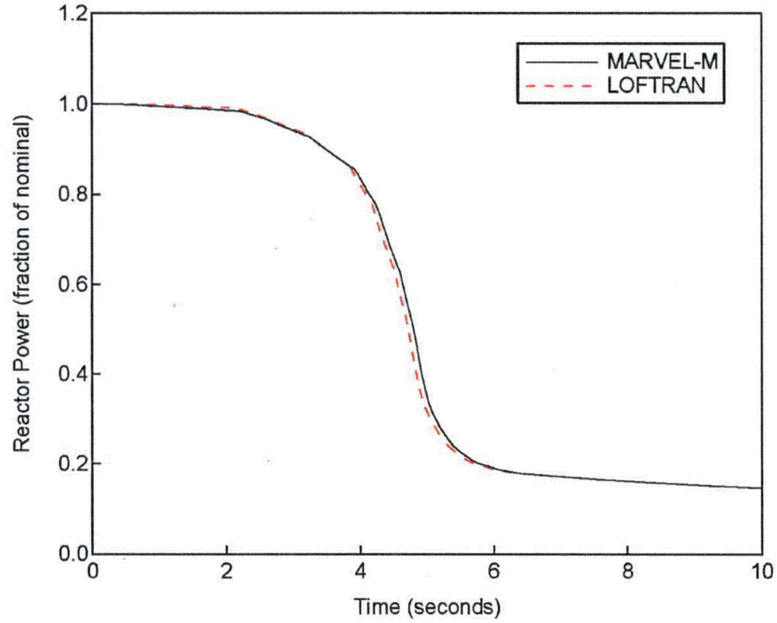


Figure 3.1.4-1 Reactor Power, Reactor Coolant Pump Shaft Seizure Comparison with MARVEL-M and LOFTRAN

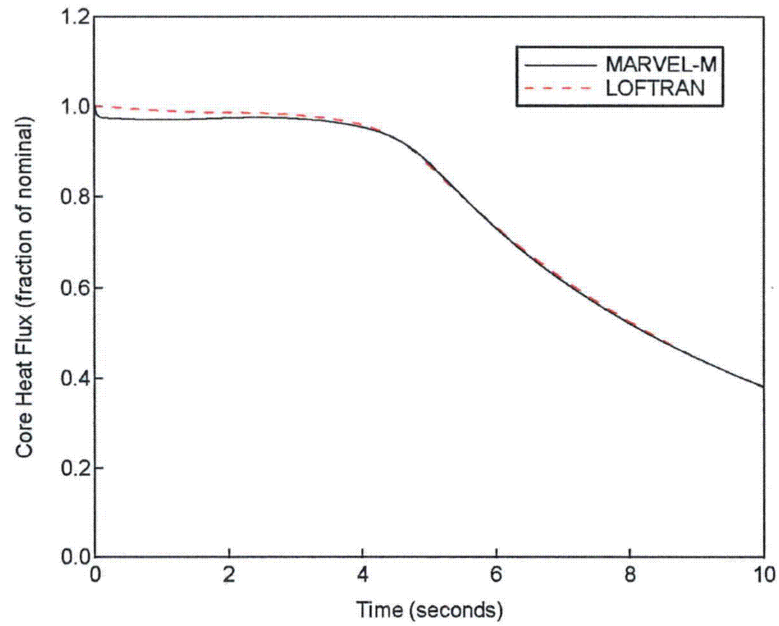


Figure 3.1.4-2 Core Heat Flux, Reactor Coolant Pump Shaft Seizure Comparison with MARVEL-M and LOFTRAN

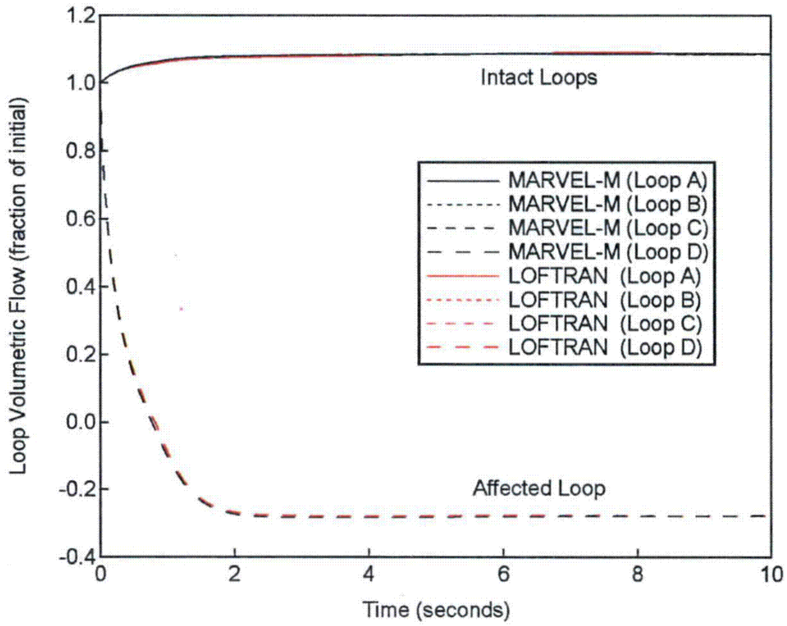


Figure 3.1.4-3 Loop Volumetric Flow, Reactor Coolant Pump Shaft Seizure Comparison with MARVEL-M and LOFTRAN

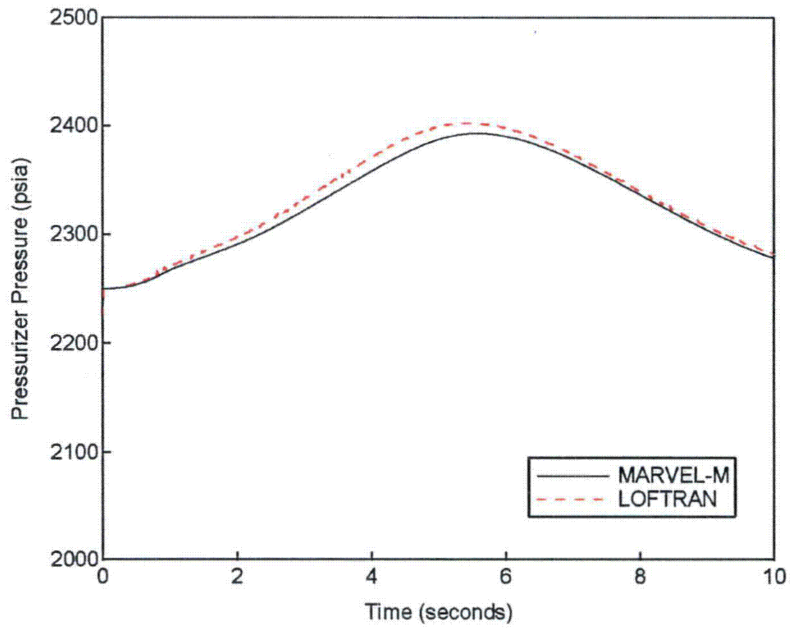


Figure 3.1.4-4 Pressurizer Pressure, Reactor Coolant Pump Shaft Seizure Comparison with MARVEL-M and LOFTRAN

3.2 TWINKLE-M Code

The solution methods and constitutive models of the TWINKLE-M code have not changed from the original version, but the maximum number of spatial mesh points is expanded from 2,000 points to a variable number input by the user.

First, the three-dimensional calculation by the TWINKLE-M code is verified by comparing the power distribution with that from the core simulator ANC code. A two-by-two (2 x 2) mesh per assembly in the radial direction are used by both codes allowing confirmation that the expanded number of mesh points in TWINKLE-M has been properly implemented.

For three-dimensional transient analyses, it is desirable to use as coarse a mesh as possible while maintaining sufficient accuracy. A second objective of the validation is to compare the results of a two-by-two (2 x 2) coarse mesh simulation of the rod ejection accident to a four-by-four (4 x 4) fine mesh simulation of the same accident with the same cross-section data using TWINKLE-M for both.

Section 3.2.1 describes a TWINKLE-M to ANC comparison for cases with and without an ejected rod under steady-state conditions. Section 3.2.2 compares the sensitivity of the TWINKLE-M results to different mesh size assumptions.

3.2.1 Comparison with Core Design Code

In this section the validity of the three-dimensional capabilities of the TWINKLE-M code are confirmed by comparing core power distribution and other parameters with the ANC. Three cases are defined to cover the range of conditions for which TWINKLE-M will be used in the non-LOCA accident analysis. The first case is hot full power with fuel temperatures at their full power values and all rods are fully withdrawn. The second case is hot zero power with uniform temperature distribution and RCCAs at the zero power insertion limit. Both cases are characterized by radial and axial power distributions in the normal operating condition. The third case is representative of a highly peaked radial power distribution characteristic of one RCCA ejection accident from the hot zero power condition.

It is important to note key differences between TWINKLE-M and ANC that are relevant to the comparison of the two codes. The TWINKLE-M code solution methodology is based on a finite difference technique, whereas ANC uses a nodal methodology. This is important because the two methods treat the core-reflector boundary condition differently. The TWINKLE-M code uses a multiplier to the diffusion coefficient for reflector regions in order to more accurately predict core power for peripheral core regions. In addition, ANC is a steady state 3-D core simulator code whereas TWINKLE-M is a 3-D core transient analysis code. As a result, all of the comparison cases are done at steady-state conditions.

(1) Analysis Assumptions
The end-of-cycle hot zero power condition is selected for this validation because the hot channel factor after RCCA ejection becomes largest for every core condition. A control and shutdown rod location in the typical 4-loop plant with a 17x17, 257 fuel assembly (17x17-257FA) core is shown in Figure 3.2.1-1. And analysis assumptions and calculation conditions are as follows:

(a) Core condition
Case 1: 24 month equilibrium core, beginning-of-cycle (BOC)
Case 2,3: 24 month equilibrium core, end-of-cycle (EOC)

(b) Initial condition
Case 1: Hot full power
Case 2,3: Hot zero power

(c) RCCA position
Case 1: All RCCAs out
Case 2: Bank-D is fully inserted. Bank-C and B are partially inserted.
Case 3: One RCCA from Bank-D is ejected from core.
The rest are same as Case 2

(d) Mesh division
2 x 2 meshes per assembly in the radial direction
( ) in the axial direction for the active core region

(2) Results and Conclusions

Radial power distribution comparison between ANC and TWINKLE-M for the hot full power case and the hot zero power RCCA insertion limit case are shown in Figure 3.2.1-2 and 3.2.1-3, respectively. A similar comparison for the one RCCA ejected case is shown in Figure 3.2.1-4. Axial power distribution comparison for all the cases is shown in Figure 3.2.1-5.

In the full power case, the radial and axial power distributions for both codes are in good agreement with small differences in some assemblies. In the zero power cases, the maximum error in the assembly average power distribution appears at RCCA locations, which is expected due to the limitations associated with the differences between the two codes modeling methodologies. Additionally, the average axial power distributions for both codes are in good agreement.

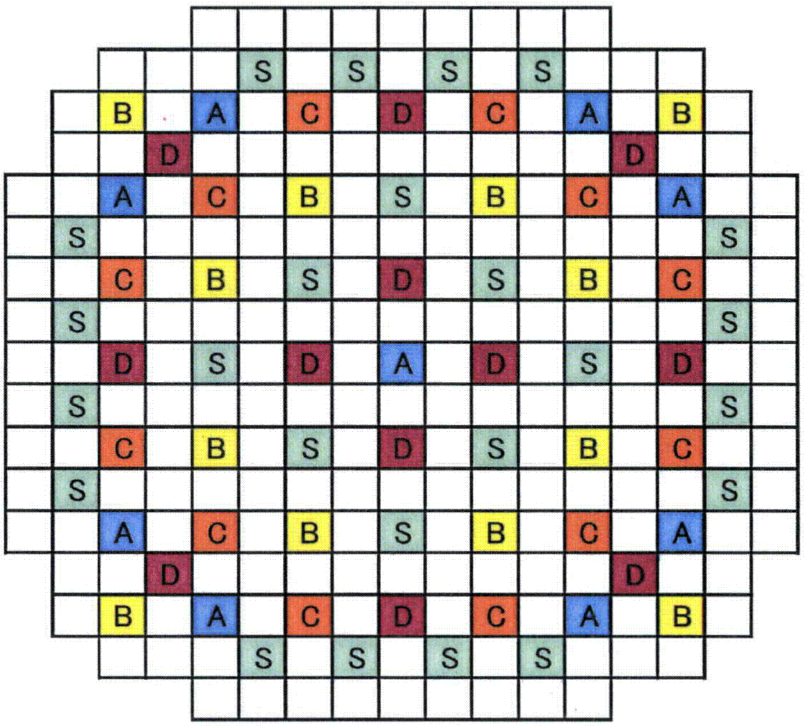
The results of ejected worth, hot channel factor and axial offset shown in Table 3.2.1-1 also demonstrate agreement between the codes.

These results indicate the validity of the three-dimensional TWINKLE-M calculation of core power utilizing the expanded number of mesh points.

Table 3.2.1-1 Results of RCCA Ejection Comparison with ANC and TWINKLE-M

	Ejected worth (pcm)	Hot channel factor	Axial offset (%)
ANC	603	27.3*	90.4
TWINKLE-M	600	27.7	90.0

* Node average value (Maximum is 29.5 rod wise)



Control group bank A	9
Control group bank B	12
Control group bank C	12
Control group bank D	12
Shutdown group bank S	24

Figure 3.2.1-1 Control and Shutdown Rod Location
(17x17-257FA Core, 4-Loop Plant)

TWINKLE-M

				0.43	0.50	0.49	0.51	0.50	0.51	0.49	0.50	0.43				
		0.40	0.64	1.31	1.10	0.93	1.04	1.06	1.04	0.93	1.10	1.31	0.64	0.40		
	0.40	1.09	1.01	1.26	1.03	1.08	1.10	1.06	1.10	1.08	1.03	1.26	1.01	1.09	0.40	
	0.64	1.01	0.95	1.02	1.14	1.28	1.15	1.09	1.15	1.28	1.14	1.02	0.95	1.01	0.64	
0.43	1.31	1.26	1.02	1.26	1.14	1.06	1.13	1.18	1.13	1.06	1.14	1.26	1.02	1.26	1.31	0.43
0.50	1.10	1.03	1.14	1.14	1.04	1.21	1.35	1.15	1.35	1.21	1.04	1.15	1.14	1.03	1.10	0.50
0.49	0.93	1.08	1.28	1.06	1.21	1.36	1.10	1.17	1.10	1.36	1.21	1.06	1.28	1.08	0.93	0.49
0.51	1.04	1.10	1.15	1.13	1.35	1.10	1.18	1.09	1.18	1.10	1.35	1.13	1.15	1.10	1.04	0.51
0.50	1.06	1.06	1.09	1.18	1.15	1.17	1.09	0.92	1.09	1.17	1.15	1.18	1.09	1.06	1.06	0.50
0.51	1.04	1.10	1.15	1.13	1.35	1.10	1.18	1.09	1.19	1.10	1.35	1.13	1.15	1.10	1.04	0.51
0.49	0.93	1.08	1.28	1.06	1.21	1.36	1.10	1.17	1.10	1.36	1.21	1.06	1.28	1.08	0.93	0.49
0.50	1.10	1.03	1.14	1.15	1.04	1.21	1.35	1.15	1.36	1.21	1.04	1.15	1.14	1.03	1.10	0.50
0.43	1.31	1.26	1.02	1.26	1.15	1.06	1.13	1.18	1.13	1.06	1.15	1.26	1.02	1.26	1.31	0.43
	0.64	1.01	0.95	1.02	1.14	1.28	1.15	1.09	1.15	1.28	1.14	1.02	0.95	1.01	0.64	
	0.40	1.09	1.01	1.26	1.03	1.08	1.10	1.06	1.10	1.08	1.03	1.26	1.01	1.09	0.40	
		0.40	0.64	1.31	1.10	0.93	1.04	1.06	1.04	0.93	1.10	1.31	0.64	0.40		
				0.43	0.50	0.49	0.51	0.50	0.51	0.49	0.50	0.43				

ANC

				0.43	0.51	0.49	0.52	0.51	0.52	0.49	0.51	0.43				
		0.40	0.65	1.23	1.08	0.94	1.03	1.06	1.03	0.94	1.08	1.23	0.65	0.40		
	0.40	1.03	1.01	1.24	1.04	1.08	1.12	1.08	1.12	1.08	1.04	1.24	1.01	1.03	0.40	
	0.65	1.01	0.97	1.03	1.15	1.27	1.17	1.11	1.16	1.27	1.15	1.04	0.97	1.01	0.65	
0.43	1.23	1.24	1.04	1.24	1.16	1.08	1.15	1.19	1.15	1.08	1.16	1.24	1.03	1.24	1.23	0.43
0.51	1.08	1.04	1.15	1.16	1.07	1.22	1.33	1.16	1.33	1.22	1.07	1.16	1.15	1.04	1.08	0.51
0.49	0.94	1.08	1.27	1.08	1.22	1.34	1.11	1.17	1.11	1.34	1.22	1.08	1.27	1.08	0.94	0.49
0.52	1.03	1.12	1.16	1.15	1.33	1.11	1.15	1.07	1.15	1.11	1.33	1.15	1.17	1.12	1.03	0.52
0.51	1.06	1.08	1.11	1.19	1.16	1.17	1.07	0.90	1.07	1.17	1.16	1.19	1.11	1.08	1.06	0.51
0.52	1.03	1.12	1.17	1.15	1.33	1.11	1.15	1.07	1.15	1.11	1.33	1.15	1.16	1.12	1.03	0.52
0.49	0.94	1.08	1.27	1.08	1.22	1.34	1.11	1.17	1.11	1.34	1.22	1.08	1.27	1.08	0.94	0.49
0.51	1.08	1.04	1.15	1.16	1.07	1.22	1.33	1.16	1.33	1.22	1.07	1.16	1.15	1.04	1.08	0.51
0.43	1.23	1.24	1.03	1.24	1.16	1.08	1.15	1.19	1.15	1.08	1.16	1.24	1.04	1.24	1.23	0.43
	0.65	1.01	0.97	1.04	1.15	1.27	1.16	1.11	1.17	1.27	1.15	1.03	0.97	1.01	0.65	
	0.40	1.03	1.01	1.24	1.04	1.08	1.12	1.08	1.12	1.08	1.04	1.24	1.01	1.03	0.40	
		0.40	0.65	1.23	1.08	0.94	1.03	1.06	1.03	0.94	1.08	1.23	0.65	0.40		
				0.43	0.51	0.49	0.52	0.51	0.52	0.49	0.51	0.43				

TWINKLE-M / ANC

				1.00	0.99	0.99	0.99	0.98	0.99	0.99	0.99	1.00				
		0.99	0.98	1.06	1.02	1.00	1.01	1.01	1.01	1.00	1.02	1.06	0.98	0.99		
	0.99	1.06	0.99	1.02	0.98	1.00	0.98	0.98	0.98	1.00	0.98	1.02	0.99	1.06	0.99	
	0.98	0.99	0.98	0.99	0.99	1.01	0.99	0.98	0.99	1.01	0.99	0.99	0.98	0.99	0.98	
1.00	1.06	1.02	0.99	1.01	0.99	0.98	0.98	0.99	0.98	0.98	0.99	1.01	0.99	1.02	1.06	1.00
0.99	1.02	0.98	0.99	0.99	0.98	0.99	1.01	0.99	1.01	0.99	0.98	0.99	0.99	0.98	1.02	0.99
0.99	1.00	1.00	1.01	0.98	0.99	1.01	0.99	1.00	0.99	1.01	0.99	0.98	1.01	1.00	1.00	0.99
0.99	1.01	0.98	0.99	0.98	1.01	0.99	1.03	1.01	1.03	0.99	1.01	0.98	0.99	0.98	1.01	0.99
0.98	1.01	0.98	0.98	0.99	0.99	1.00	1.02	1.02	1.02	1.00	0.99	0.99	0.98	0.98	1.01	0.98
0.99	1.01	0.98	0.99	0.98	1.01	0.99	1.03	1.02	1.03	0.99	1.01	0.98	0.99	0.98	1.01	0.99
0.99	1.00	1.00	1.01	0.98	0.99	1.01	0.99	1.00	0.99	1.01	0.99	0.98	1.01	1.00	1.00	0.99
0.99	1.02	0.98	0.99	0.99	0.98	0.99	1.01	0.99	1.02	0.99	0.98	0.99	0.99	0.98	1.02	0.99
1.00	1.06	1.02	0.99	1.01	0.99	0.98	0.98	0.99	0.98	0.98	0.99	1.01	0.99	1.02	1.06	1.00
	0.98	0.99	0.98	0.99	0.99	1.01	0.99	0.98	0.99	1.01	0.99	0.99	0.98	0.99	0.98	
	0.99	1.06	0.99	1.02	0.98	1.00	0.98	0.98	0.98	1.00	0.98	1.02	0.99	1.06	0.99	
		0.99	0.98	1.06	1.02	1.00	1.01	1.01	1.01	1.00	1.02	1.06	0.98	0.99		
				1.00	0.99	0.99	0.99	0.98	0.99	0.99	0.99	1.00				

Legend			
		～	±3%
	±3%	～	±5%
	±5%	～	±8%

Figure 3.2.1-2 Radial Power Distribution Comparison with ANC and TWINKLE-M Case 1, BOC HFP All RCCAs Out

TWINKLE-M

				0.52	0.68	0.73	0.77	0.74	0.77	0.73	0.68	0.52				
		0.35	0.64	1.21	1.30	1.29	1.41	1.37	1.41	1.29	1.30	1.21	0.64	0.35		
	0.35	0.65	0.97	1.22	0.92	0.61	1.27	0.69	1.27	0.61	0.92	1.22	0.97	0.65	0.35	
	0.64	0.97	0.44	0.74	1.16	1.18	1.38	1.15	1.38	1.18	1.16	0.74	0.44	0.97	0.64	
0.52	1.21	1.22	0.74	0.59	1.09	0.75	1.14	1.50	1.14	0.75	1.09	0.59	0.74	1.22	1.21	0.52
0.68	1.30	0.92	1.16	1.09	1.02	1.44	1.46	1.13	1.46	1.44	1.02	1.09	1.16	0.92	1.30	0.68
0.73	1.29	0.61	1.18	0.75	1.44	1.53	1.03	0.73	1.03	1.53	1.44	0.75	1.18	0.61	1.29	0.73
0.77	1.41	1.27	1.38	1.14	1.46	1.03	1.04	0.96	1.04	1.03	1.46	1.14	1.38	1.27	1.41	0.77
0.74	1.37	0.69	1.15	1.50	1.13	0.73	0.96	0.85	0.96	0.73	1.13	1.50	1.15	0.69	1.37	0.74
0.77	1.41	1.27	1.38	1.14	1.46	1.03	1.04	0.96	1.04	1.03	1.46	1.14	1.38	1.27	1.41	0.77
0.73	1.29	0.61	1.18	0.75	1.44	1.53	1.03	0.73	1.03	1.53	1.44	0.75	1.18	0.61	1.29	0.73
0.68	1.30	0.92	1.16	1.09	1.02	1.44	1.46	1.13	1.46	1.44	1.02	1.09	1.16	0.92	1.30	0.68
0.52	1.21	1.22	0.74	0.59	1.09	0.75	1.14	1.50	1.14	0.75	1.09	0.59	0.74	1.22	1.21	0.52
	0.64	0.97	0.44	0.74	1.16	1.18	1.38	1.15	1.38	1.17	1.16	0.74	0.44	0.97	0.64	
	0.35	0.65	0.97	1.22	0.92	0.61	1.27	0.69	1.27	0.61	0.92	1.22	0.97	0.65	0.35	
		0.35	0.64	1.21	1.30	1.29	1.41	1.37	1.41	1.29	1.30	1.21	0.64	0.35		
				0.52	0.68	0.73	0.77	0.74	0.77	0.73	0.68	0.52				

ANC

				0.52	0.68	0.74	0.77	0.75	0.77	0.74	0.68	0.52				
		0.35	0.63	1.16	1.27	1.27	1.38	1.35	1.38	1.27	1.27	1.16	0.63	0.35		
	0.35	0.65	0.97	1.21	0.94	0.65	1.27	0.73	1.27	0.65	0.94	1.21	0.97	0.65	0.35	
	0.63	0.97	0.46	0.76	1.15	1.18	1.36	1.16	1.36	1.18	1.15	0.76	0.46	0.97	0.63	
0.52	1.16	1.21	0.76	0.62	1.10	0.80	1.16	1.50	1.15	0.80	1.10	0.62	0.76	1.21	1.16	0.52
0.68	1.27	0.94	1.15	1.10	1.03	1.44	1.44	1.13	1.44	1.44	1.03	1.10	1.15	0.94	1.27	0.68
0.74	1.27	0.65	1.18	0.80	1.44	1.52	1.04	0.76	1.04	1.52	1.44	0.80	1.18	0.65	1.27	0.74
0.77	1.38	1.27	1.36	1.15	1.44	1.04	1.03	0.95	1.03	1.04	1.44	1.16	1.36	1.27	1.38	0.77
0.75	1.35	0.73	1.16	1.50	1.13	0.76	0.95	0.84	0.95	0.76	1.13	1.50	1.16	0.73	1.35	0.75
0.77	1.38	1.27	1.36	1.16	1.44	1.04	1.03	0.95	1.03	1.04	1.44	1.16	1.36	1.27	1.38	0.77
0.74	1.27	0.65	1.18	0.80	1.44	1.52	1.04	0.76	1.04	1.52	1.44	0.80	1.18	0.65	1.27	0.74
0.68	1.27	0.94	1.15	1.10	1.03	1.44	1.44	1.13	1.44	1.44	1.03	1.10	1.15	0.94	1.27	0.68
0.52	1.16	1.21	0.76	0.62	1.10	0.80	1.16	1.50	1.16	0.80	1.10	0.62	0.76	1.21	1.17	0.52
	0.63	0.97	0.46	0.76	1.15	1.18	1.36	1.16	1.36	1.18	1.15	0.76	0.46	0.97	0.63	
	0.35	0.65	0.97	1.21	0.94	0.65	1.27	0.73	1.27	0.65	0.94	1.21	0.97	0.65	0.35	
		0.351	0.632	1.17	1.27	1.27	1.38	1.35	1.38	1.27	1.27	1.16	0.63	0.35		
				0.52	0.68	0.74	0.77	0.75	0.77	0.74	0.68	0.52				

TWINKLE-M / ANC

				1.01	1.00	0.99	1.00	0.98	0.99	0.99	1.00	1.02				
		1.01	1.01	1.04	1.03	1.02	1.02	1.01	1.02	1.02	1.03	1.04	1.01	1.01		
	1.01	1.00	1.00	1.01	0.98	0.94	1.00	0.94	1.00	0.94	0.98	1.01	1.00	1.00	1.01	
	1.01	1.00	0.94	0.98	1.01	1.00	1.01	0.99	1.01	1.00	1.01	0.98	0.94	1.00	1.01	
1.02	1.04	1.01	0.98	0.95	0.99	0.94	0.99	1.00	0.99	0.94	0.99	0.95	0.98	1.01	1.04	1.01
1.00	1.03	0.98	1.01	0.99	0.99	1.01	1.01	1.00	1.01	1.00	0.99	0.99	1.01	0.98	1.03	1.00
0.99	1.02	0.94	1.00	0.94	1.01	1.01	1.00	0.97	0.99	1.01	1.00	0.94	1.00	0.94	1.02	0.99
0.99	1.02	1.00	1.01	0.99	1.01	1.00	1.01	1.01	1.02	1.00	1.01	0.99	1.01	1.00	1.02	0.99
0.98	1.01	0.94	0.99	1.00	1.00	0.97	1.01	1.01	1.01	0.97	1.00	1.00	0.99	0.94	1.01	0.98
1.00	1.02	1.00	1.01	0.99	1.01	1.00	1.02	1.01	1.02	0.99	1.01	0.99	1.01	1.00	1.02	0.99
0.99	1.02	0.94	1.00	0.94	1.00	1.01	0.99	0.97	1.00	1.01	1.00	0.94	1.00	0.94	1.02	0.99
0.99	1.02	0.98	1.01	0.99	0.99	1.00	1.01	1.00	1.01	1.00	0.99	0.99	1.01	0.98	1.02	1.00
1.01	1.04	1.01	0.98	0.95	0.99	0.94	0.99	1.00	0.99	0.94	0.99	0.95	0.98	1.01	1.04	1.02
	1.01	1.00	0.94	0.98	1.01	1.00	1.01	0.99	1.01	1.00	1.01	0.98	0.94	1.00	1.01	
	1.01	1.00	1.00	1.01	0.98	0.94	1.00	0.94	1.00	0.94	0.98	1.01	1.00	1.00	1.01	
		1.01	1.01	1.04	1.02	1.02	1.02	1.01	1.02	1.02	1.02	1.04	1.00	1.01		
				1.02	1.00	0.99	0.99	0.98	0.99	0.99	0.99	1.01				

Legend			
(white)		~	±3%
(yellow)	±3%	~	±5%
(orange)	±5%	~	±8%

Figure 3.2.1-3 Radial Power Distribution Comparison with ANC and TWINKLE-M Case 2, EOC HZP RCCA at Insertion Limit

TWINKLE-M

				0.11	0.14	0.15	0.16	0.16	0.16	0.16	0.14	0.11				
		0.08	0.15	0.27	0.29	0.28	0.31	0.30	0.31	0.28	0.29	0.27	0.15	0.08		
	0.10	0.16	0.23	0.29	0.22	0.14	0.30	0.16	0.30	0.14	0.22	0.29	0.23	0.16	0.10	
	0.19	0.28	0.12	0.20	0.31	0.32	0.37	0.31	0.37	0.32	0.31	0.20	0.12	0.28	0.19	
0.17	0.38	0.38	0.24	0.19	0.35	0.24	0.36	0.47	0.36	0.24	0.35	0.19	0.24	0.38	0.38	0.17
0.23	0.45	0.32	0.42	0.41	0.39	0.55	0.54	0.41	0.54	0.55	0.39	0.41	0.42	0.32	0.45	0.23
0.28	0.50	0.24	0.49	0.32	0.64	0.68	0.46	0.33	0.46	0.68	0.64	0.32	0.49	0.24	0.50	0.28
0.33	0.61	0.58	0.67	0.59	0.78	0.56	0.60	0.57	0.60	0.56	0.78	0.59	0.67	0.58	0.62	0.33
0.36	0.67	0.36	0.66	0.93	0.76	0.54	0.74	0.67	0.74	0.54	0.76	0.93	0.66	0.36	0.68	0.36
0.42	0.79	0.76	0.91	0.84	1.26	1.01	1.08	0.99	1.08	1.01	1.26	0.84	0.91	0.76	0.79	0.42
0.46	0.82	0.42	0.92	0.68	1.54	1.87	1.40	1.04	1.40	1.87	1.54	0.68	0.92	0.42	0.83	0.46
0.48	0.97	0.76	1.07	1.21	1.34	2.17	2.56	2.15	2.56	2.17	1.34	1.21	1.07	0.76	0.97	0.48
0.41	1.01	1.13	0.80	0.83	1.86	1.46	2.67	3.73	2.67	1.46	1.86	0.83	0.80	1.13	1.01	0.41
	0.62	1.07	0.62	1.36	2.43	3.02	4.38	4.11	4.39	3.02	2.43	1.36	0.62	1.07	0.62	
	0.41	0.87	1.68	2.49	2.19	1.85	5.16	5.17	5.17	1.85	2.19	2.49	1.68	0.87	0.41	
		0.57	1.22	2.65	3.34	4.07	5.53	6.13	5.54	4.08	3.34	2.66	1.22	0.57		
				1.22	1.79	2.31	2.85	2.96	2.85	2.31	1.80	1.22				

Ejected Rod

ANC

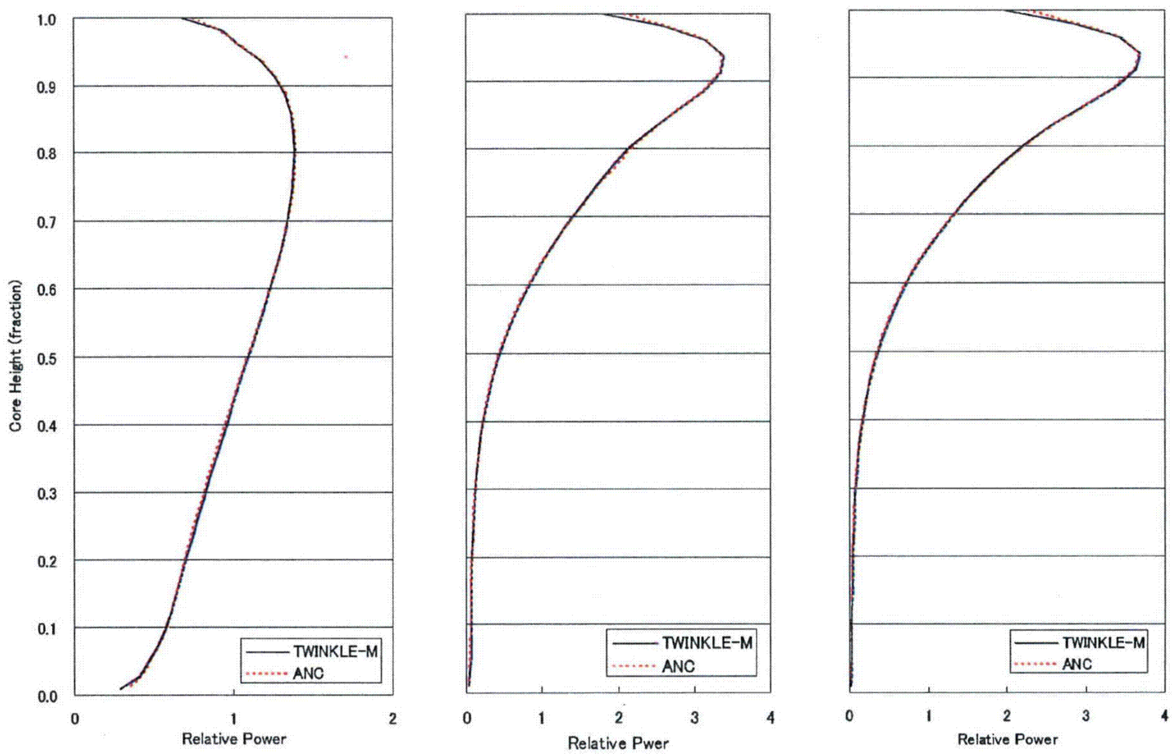
				0.11	0.15	0.16	0.16	0.16	0.16	0.16	0.15	0.11				
		0.09	0.15	0.26	0.28	0.28	0.30	0.30	0.30	0.28	0.28	0.26	0.15	0.09		
	0.10	0.17	0.24	0.29	0.22	0.15	0.30	0.17	0.30	0.15	0.22	0.29	0.24	0.17	0.10	
	0.19	0.28	0.13	0.20	0.31	0.32	0.37	0.32	0.37	0.32	0.31	0.20	0.13	0.28	0.19	
0.17	0.37	0.38	0.25	0.20	0.35	0.25	0.36	0.47	0.36	0.25	0.35	0.20	0.25	0.38	0.37	0.17
0.24	0.44	0.33	0.42	0.41	0.39	0.54	0.53	0.41	0.53	0.54	0.39	0.41	0.42	0.33	0.44	0.24
0.28	0.49	0.26	0.50	0.35	0.64	0.67	0.46	0.34	0.46	0.67	0.64	0.35	0.50	0.26	0.49	0.28
0.33	0.61	0.58	0.67	0.60	0.77	0.56	0.59	0.56	0.59	0.56	0.77	0.60	0.67	0.59	0.61	0.34
0.37	0.68	0.38	0.67	0.93	0.76	0.55	0.73	0.66	0.73	0.55	0.76	0.93	0.67	0.38	0.68	0.37
0.43	0.78	0.77	0.91	0.86	1.25	1.01	1.06	0.98	1.06	1.02	1.25	0.86	0.91	0.77	0.78	0.43
0.47	0.82	0.45	0.93	0.73	1.54	1.86	1.40	1.07	1.40	1.86	1.54	0.73	0.93	0.45	0.82	0.47
0.49	0.96	0.78	1.07	1.23	1.36	2.16	2.52	2.13	2.52	2.17	1.36	1.23	1.07	0.78	0.96	0.49
0.41	0.99	1.13	0.83	0.87	1.87	1.55	2.69	3.70	2.69	1.55	1.87	0.87	0.83	1.13	0.99	0.41
	0.62	1.08	0.66	1.38	2.40	3.00	4.30	4.11	4.31	3.01	2.40	1.38	0.66	1.08	0.62	
	0.41	0.87	1.67	2.45	2.22	1.94	5.11	5.19	5.13	1.95	2.22	2.45	1.67	0.87	0.41	
		0.56	1.20	2.54	3.24	4.00	5.41	6.03	5.42	4.00	3.25	2.55	1.20	0.56		
				1.20	1.80	2.33	2.86	3.00	2.86	2.33	1.80	1.20				

TWINKLE-M / ANC

				1.01	0.99	0.98	0.99	0.98	0.99	0.99	0.99	1.00				
		0.99	0.99	1.02	1.01	1.00	1.01	1.00	1.01	1.00	1.01	1.02	0.99	0.99		
	0.99	0.99	0.99	1.00	0.97	0.94	0.99	0.94	1.00	0.94	0.97	1.00	0.99	0.99	0.99	
	0.99	0.99	0.93	0.97	1.00	0.99	1.01	0.99	1.01	0.99	1.00	0.97	0.93	0.99	0.99	
1.01	1.03	1.00	0.97	0.95	0.99	0.94	0.99	1.00	0.99	0.94	0.99	0.95	0.97	1.00	1.03	1.01
0.99	1.01	0.97	1.00	0.99	0.99	1.01	1.01	1.00	1.02	1.01	0.99	0.99	1.00	0.97	1.01	0.98
0.98	1.00	0.93	0.99	0.94	1.01	1.01	1.00	0.97	1.00	1.01	1.00	0.93	0.99	0.93	1.00	0.98
0.99	1.01	0.99	1.01	0.99	1.01	0.99	1.02	1.02	1.02	0.99	1.01	0.98	1.01	0.99	1.01	0.98
0.97	1.00	0.93	0.98	0.99	1.00	0.97	1.01	1.01	1.02	0.97	1.00	0.99	0.98	0.93	1.00	0.97
0.98	1.01	0.99	1.01	0.98	1.01	0.99	1.01	1.01	1.01	0.99	1.01	0.98	1.01	0.99	1.01	0.98
0.98	1.00	0.93	0.99	0.93	1.00	1.01	1.00	0.97	0.99	1.01	1.00	0.93	0.99	0.93	1.00	0.98
0.98	1.01	0.97	1.00	0.99	0.99	1.00	1.02	1.01	1.02	1.00	0.99	0.98	1.00	0.97	1.01	0.98
1.00	1.02	1.00	0.96	0.95	0.99	0.94	0.99	1.01	0.99	0.94	0.99	0.95	0.97	1.00	1.02	1.00
	0.99	0.99	0.94	0.98	1.01	1.00	1.02	1.00	1.02	1.00	1.01	0.98	0.94	0.99	0.99	
	1.00	0.99	1.01	1.02	0.99	0.95	1.01	1.00	1.01	0.95	0.99	1.02	1.01	0.99	1.00	
		1.02	1.02	1.04	1.03	1.02	1.02	1.02	1.02	1.02	1.03	1.04	1.02	1.02		
				1.02	1.00	0.99	1.00	0.99	1.00	0.99	1.00	1.02				

Legend			
		~	±3%
	±3%	~	±5%
	±5%	~	±8%

Figure 3.2.1-4 Radial Power Distribution Comparison with ANC and TWINKLE-M Case 3, EOC HZP One RCCA Ejected



Case 1	Case 2	Case 3
BOC HFP	EOC HZP	EOC HZP
All RCCAs Out	RCCA at Insertion Limit	One RCCA Ejected

Figure 3.2.1-5 Average Axial Power Distribution Comparison with ANC and TWINKLE-M

3.2.2 Sensitivity Study of Mesh Size

In the ejected rod accident simulation, reactivity insertion and reactivity feedback including Doppler feedback are most important parameters. In the three-dimensional calculation, these effects are dependent on calculational mesh size. The coarseness of the spatial mesh generally influences the accuracy of three-dimensional finite-difference techniques for solving the diffusion equations. A sensitivity study of spatial mesh division utilizing the same cross-section data is performed in order to determine the optimal spatial mesh size.

In this study, sensitivity analysis of the mesh division is performed in the radial direction. Using a fine mesh analysis in the finite-difference technique provides the most accurate results.

(1) Analysis Assumptions

Analysis assumptions and calculation conditions in the typical 4-loop plant with a 17x17-257FA core are as follows. The end-of-cycle hot zero power condition is selected for comparison because it represents the most severe case in the RCCA ejection accident.

(a) Core condition	24 month equilibrium core end-of-cycle
(b) Initial condition	Hot zero power
(c) RCCA position	Insertion limit at initial
(d) Mesh division	Case 1: 2 x 2 meshes per assembly in the radial direction Case 2: 4 x 4 meshes per assembly in the radial direction [ ]in the axial direction in the active core region (both cases)
(e) Ejected rod	One RCCA ejected from fully inserted Bank-D within 0.1 seconds

(2) Results and Conclusions

Results of the RCCA ejection analysis including the main calculation conditions using a 2 x 2 mesh and a 4 x 4 mesh are shown in Table 3.2.2-1. The transient response of the core average power and the hot channel factor is shown in Figures 3.2.2-1 and 3.2.2-2, respectively.

The results indicate that the 2 x 2 mesh calculation is sufficient for use in the accident analysis.

Table 3.2.2-1 Calculation Condition and Results of the RCCA Ejection

	Case 1 2 x 2 mesh	Case 2 4 x 4 mesh
Initial power (fraction of nominal) Average coolant temperature (F) RCS pressure (psia)	10^{-9} 557 2250	Same as 2 x 2
Ejected worth (pcm)	600	595
Delayed neutron fraction (%)	0.44	Same as 2 x 2
Neutron life time (micro seconds)	8.0	Same as 2 x 2
Maximum core power (fraction of nominal)	3.12	2.98
Maximum hot channel factor	27.5	27.3

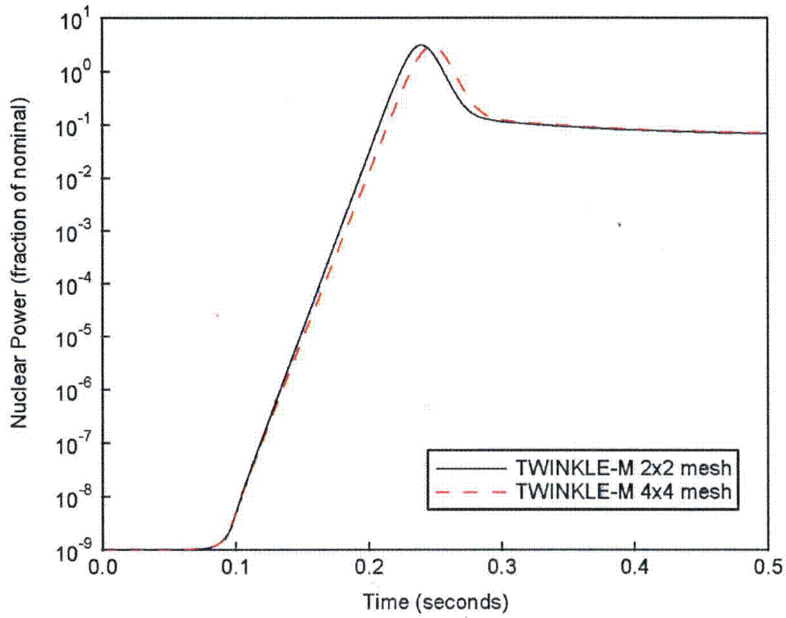


Figure 3.2.2-1 Nuclear power, RCCA Ejection at EOC HZP Comparison with 2 x 2 mesh and 4 x 4 mesh in TWINKLE-M

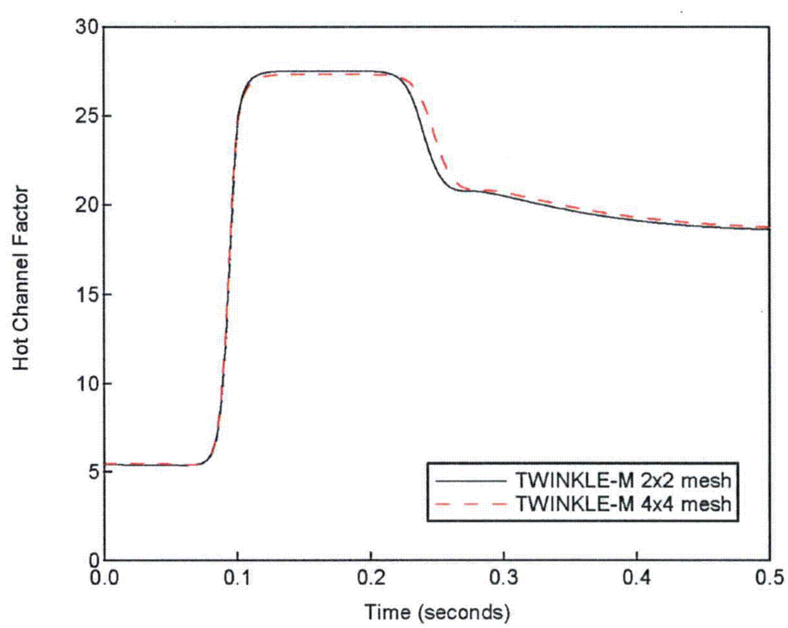


Figure 3.2.2-2 Hot Channel factor, RCCA Ejection at EOC HZP Comparison with 2 x 2 mesh and 4 x 4 mesh in TWINKLE-M

4.0 ACCEPTANCE CRITERIA FOR SRP CHAPTER 15 NON-LOCA EVENTS

The methodology described in this Topical Report is applicable to the US-APWR plant design and modes of plant operation addressed in the non-LOCA accident analysis. In particular, the methodology described is related to the thermal-hydraulic aspects of the SRP Chapter 15 non-LOCA events for US-APWR that challenge the cladding and reactor coolant system fission product barriers; the non-LOCA methodology does not include the dose consequence analysis of radiological releases, accidents that only apply to Boiling Water Reactors, or events that are beyond the design basis.

The accident analysis in the Design Certification and Combined License Applications will be organized consistent with the categories shown below in Table 4-1, as defined in the Standard Review Plan (SRP) NUREG-0800 and the most recent version of Regulatory Guide 1.206. Regulatory Guide 1.206 will serve the same purpose (format and content guide) for new plants licensed under Part 52 as Regulatory Guide 1.70 serves for the current operating US plant Updated Final Safety Analysis Reports.

Table 4-1 Event Classification Categories

Category Number	Event Categorization By Effect on the Plant
15.1	Increase in Heat Removal by the Secondary System
15.2	Decrease in Heat Removal by the Secondary System
15.3	Decrease in Reactor Coolant System Flow Rate
15.4	Reactivity and Power Distribution Anomalies
15.5	Increase in Reactor Coolant Inventory
15.6	Decrease in Reactor Coolant Inventory

Due to the similarity between the MHI US-APWR and the current generation of PWRs in the United States, MHI has determined that no new categories of events (determined by effect on the plant) are required to bound the possible initiating events.

Each of the event categories in Table 4-1 have different potential initiating events that can be further categorized according to their expected frequency of occurrence. Historically, the frequency of each event was categorized as a fault of moderate frequency (ANSI 18.2 Category II), limiting fault (Category III), or design basis fault (Category IV), and frequency-class-based acceptance criteria associated with each category applied to specific accidents. For new plants, the current SRPs no longer use the historical frequency categories by name or number, but instead, re-categorize each event as either an Anticipated Operational Occurrence (AOO) or Postulated Accident (PA). The following definitions of AOO and PA are derived from the SRPs:

- Anticipated operational occurrences (AOOs), as defined in Appendix A to 10 CFR Part 50, are those conditions of normal operation that are expected to occur one or more times during the life of the plant. The SRP reiterates the 10 CFR 50 Appendix A definition of the term AOOs and adds that AOOs are also known as Condition II and Condition III events (referring to events that are categorized in Regulatory Guide 1.70 and Regulatory Guide 1.206 as incidents of moderate frequency and infrequent events). Incidents of moderate frequency and infrequent events have also been previously known as ANSI 18.2 Condition II and Condition III events, respectively.

- Postulated accidents (PAs) are unanticipated occurrences (i.e., they are postulated but are not expected to occur during the life of the plant.) They are analyzed to confirm the adequacy of plant safety systems. These accidents have also been previously known as ANSI 18.2 Condition IV events or "Design Basis Accidents".

It is important to note that AOOs and PAs apply to certain initiating events, but that there are transients and accidents that are more severe and infrequent than AOOs, but not as severe and infrequent as PAs. Examples of these events include AOOs with an assumed coincident single failure or operator error, as well as infrequent events that can only result from coincident component active failures or passive failures. AOOs that occur with such a coincident failure are no longer considered AOOs.

Section 4.1 documents the acceptance criteria MHI plans to use for AOOs and PAs based on the SRPs, modified as needed, to identify the key criteria and additional more restrictive criteria imposed by MHI for each of the non-LOCA accidents to be provided in the Design Certification Application Design Control Document (DCD). The six event categories in Table 4-1 are then expanded in Sections 4.2 through 4.7 to define all of the related initiated events, each of which will be quantitatively analyzed in the US-APWR Design Certification Document (DCD).

4.1 Acceptance Criteria

Licensing analyses are performed to demonstrate that an operating plant can meet the applicable acceptance criteria for a limiting set of AOOs and PAs. This section provides the acceptance criteria used for the accident analyses of the US-APWR.

The General Design Criteria (GDC) are written such that the risk of an event, defined as the product of an event's frequency of occurrence and its consequences, is approximately equal across the spectrum of AOOs and PAs. The first two sub-sections of Section 4.1 provide the general SRP acceptance criteria for the AOO and PA categorization of accidents. Additional event-specific criteria, including event-specific SRP criteria such as PCMI cladding failure limit or internal MHI acceptance criteria, are described in the appropriate event classification discussion.

4.1.1 AOO Acceptance Criteria

The following are the generic criteria necessary to meet the requirements of GDC for AOOs:

i. Pressure in the reactor coolant (P_{RCS}) and main steam (P_{MS}) systems should be maintained below 110 percent of the design values in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code.

ii. Fuel cladding integrity is maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit.

iii. An AOO should not generate a postulated accident without other faults occurring independently or result in a consequential loss of function of the reactor coolant system (RCS) or reactor containment barriers.

General Design Criterion (GDC) 10 within Appendix A to 10 CFR 50, establishes that specified acceptable fuel design limits (SAFDLs) should not be exceeded during any condition of normal operation, including the effects of AOOs. Further guidance for interpreting this regulation is provided in SRP 4.2.

4.1.2 PA Acceptance Criteria

A list of the basic criteria necessary to meet the requirements of GDC for postulated accidents appears below.

i. Pressure in the reactor coolant and main steam systems should be maintained below acceptable design limits.
ii. Fuel cladding integrity is maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit. If the minimum DNBR does not meet this limit, then the fuel is assumed to have failed.
iii. The release of radioactive material shall not result in offsite doses in excess of the guidelines of 10 CFR Part 100. Any event-specific accident limits for allowable radiological releases are described in the appropriate section (i.e., for specific reactivity initiated accidents) below.
iv. The postulated accident shall not, by itself, cause a consequential loss of required functions of systems needed to cope with the fault, including those of the RCS and the reactor containment system.

For the Reactivity Initiated Accidents (RIA), SRP 4.2 Appendix B provides the following additional acceptance criteria regarding core coolability (which are considered an extension of criteria iv above):

- Peak radial average fuel enthalpy must remain below 230 cal/g.
- Peak fuel temperature must remain below incipient fuel melting conditions.
- Mechanical energy generated as a result of (1) non-molten fuel-to-coolant interaction and (2) fuel rod burst must be addressed with respect to pressure boundary, reactor internals, and fuel assembly structural integrity.
- No loss of coolable geometry due to (1) fuel pellet and cladding fragmentation and dispersal and (2) fuel rod ballooning.

4.2 Increase in Heat Removal from the Primary System

This category covers events that lead to heat removal exceeding the heat generation in the core potentially leading to a decrease in moderator temperature resulting in an increased power level and reduced shutdown margin. The following table summarizes the five initiating events considered for the US-APWR, their associated event classification, the computer codes used to analyze the event for compliance with applicable codes and regulations, and a listing of the event-specific acceptance criteria.

Table 4.2-1 Events in Increase in Heat Removal from the Primary System

Event	Class	Code	Acceptance Criteria
1. Decrease in feedwater temperature	AOO	MARVEL-M	• max P_{RCS} & P_{MS} < 110% design • min DNBR > 95/95 DNBR limit *1
2. Increase in feedwater flow	AOO	MARVEL-M	• max P_{RCS} & P_{MS} < 110% design • min DNBR > 95/95 DNBR limit *1 • no SG overfill *1
3. Increase in steam flow	AOO	MARVEL-M	• max P_{RCS} & P_{MS} < 110% design • min DNBR > 95/95 DNBR limit*1
4. Inadvertent opening of a steam generator relief or safety valve	AOO	MARVEL-M, VIPRE-01M*2	• max P_{RCS} & P_{MS} < 110% design • min DNBR > 95/95 DNBR limit*1
5. Steam system piping failure a. Minor break	AOO	MARVEL-M, VIPRE-01M*2	• max P_{RCS} & P_{MS} < 110% design • min DNBR > 95/95 DNBR limit*1
b. Major break (double-ended)	PA	MARVEL-M, VIPRE-01M*2	• max P_{RCS} & P_{MS} < acceptable design limits • min DNBR > 95/95 DNBR limit *1, *3

*1 Indicates the key parameter / acceptance limit of concern.
*2 Steady state analysis
*3 MHI internal design criterion does not allow DNB to occur.

4.3 Decrease in Heat Removal by the Secondary System

This category covers events that lead to unplanned decreases in heat removal by the secondary system. The following table summarizes the six initiating events considered for the US-APWR, their associated event classification, the computer codes used to analyze the event for compliance with applicable codes and regulations, and a listing of the event-specific acceptance criteria.

Table 4.3-1 Events in Decrease in Heat Removal by the Secondary System

Event	Class	Code	Acceptance Criteria
1. Loss of external electrical load and/or turbine trip	AOO	MARVEL-M	• max P_{RCS} & P_{MS} < 110% design*1 • min DNBR > 95/95 DNBR limit
2. Inadvertent closure of main steam isolation valves	AOO	MARVEL-M	• max P_{RCS} & P_{MS} < 110% design*1 • min DNBR > 95/95 DNBR limit
3. Loss of condenser vacuum and other events resulting in turbine trip	AOO	MARVEL-M	• max P_{RCS} & P_{MS} < 110% design*1 • min DNBR > 95/95 DNBR limit
4. Loss of non-emergency ac power to the station auxiliaries	AOO	MARVEL-M	• max P_{RCS} & P_{MS} < 110% design • min DNBR > 95/95 DNBR limit • establish natural circulation flow*1, *3
5. Loss of normal feedwater flow	AOO	MARVEL-M	• max P_{RCS} & P_{MS} < 110% design*1 • min DNBR > 95/95 DNBR limit
6. Feedwater system pipe break a. Minor break	AOO	MARVEL-M	• max P_{RCS} & P_{MS} < 110% design*1 • min DNBR > 95/95 DNBR limit • pressurizer does not fill *3 • evaluate hot leg boiling *3
b. Major break (double-ended)	PA	MARVEL-M	• max P_{RCS} & P_{MS} < 120% design*1 • min DNBR > 95/95 DNBR limit*2 • pressurizer does not fill *3 • evaluate hot leg boiling *3

*1 Indicates the key parameter / acceptance limit of concern.

*2 If the DNBR falls below this value, fuel failure (rod perforation) must be assumed for all rods that do not meet these criteria unless it can be shown, based on an acceptable fuel damage model that includes the potential adverse effects of hydraulic instabilities, that fewer failures occur. If rod internal pressure exceeds system pressure, then fuel rods may balloon shortly after entering DNB. The effect of ballooning fuel rods must be evaluated with respect to flow blockage and DNB propagation. Any fuel damage calculated to occur must be sufficiently limited to the extent that the core will remain in place and intact with no loss of core cooling capability.

*3 MHI internal acceptance criterion

4.4 Decrease in Reactor Coolant System Flow Rate

This category covers events that lead to a decrease in reactor coolant flow that could result in fuel damage if certain specified acceptable fuel design limits (SAFDLs) are exceeded. The following table summarizes the four initiating events considered for the US-APWR, their associated event classification, the computer codes used to analyze the event for compliance with applicable codes and regulations, and a listing of the event-specific acceptance criteria.

Table 4.4-1 Events in Decrease in Reactor Coolant System Flow Rate

Event	Class	Code	Acceptance Criteria
1. Partial loss of forced reactor coolant flow	AOO	MARVEL-M, VIPRE-01M	• max P_{RCS} & P_{MS} < 110% design • min DNBR > 95/95 DNBR limit*1
2. Complete loss of forced reactor coolant flow	AOO	MARVEL-M, VIPRE-01M	• max P_{RCS} & P_{MS} < 110% design • min DNBR > 95/95 DNBR limit*1
3. Reactor coolant pump shaft seizure (locked rotor)	PA	MARVEL-M, VIPRE-01M	• max P_{RCS} & P_{MS} < acceptable design limits • min DNBR > 95/95 DNBR limit *1, *2 • site boundary dose limited to small fraction of 10 CFR 100 values • long-term core coolability maintained
4. Reactor coolant pump shaft break	PA	MARVEL-M, VIPRE-01M	• max P_{RCS} & P_{MS} < acceptable design limits • min DNBR > 95/95 DNBR limit *1, *2 • site boundary dose limited to small fraction of 10 CFR 100 values • long-term core coolability maintained

*1 Indicates the key parameter / acceptance limit of concern.

*2 If the DNBR falls below this value, fuel failure (rod perforation) must be assumed for all rods that do not meet these criteria unless it can be shown, based on an acceptable fuel damage model that includes the potential adverse effects of hydraulic instabilities, that fewer failures occur. If rod internal pressure exceeds system pressure, then fuel rods may balloon shortly after entering DNB. The effect of ballooning fuel rods must be evaluated with respect to flow blockage and DNB propagation. Any fuel damage calculated to occur must be sufficiently limited to the extent that the core will remain in place and intact with no loss of core cooling capability.

4.5 Reactivity and Power Distribution Anomalies

This category covers events associated with unintended fuel rod movement or core flow parameter (temperature, boron concentration, etc.) changes that alter reactivity or power distribution. The following table summarizes the six initiating events considered for the US-APWR, their associated event classification, the computer codes used to analyze the event for compliance with applicable codes and regulations, and a listing of the event-specific acceptance criteria. It should be noted that the event classification of the withdrawal of a single RCCA has been defined as PA. Limited fuel damage has traditionally been allowed for this event when it was classified as a Condition III event per ANSI 18.2. This classification is consistent with its low expected frequency and the multiple failures required to initiate a single rod withdrawal.

Table 4.5-1 Events in Reactivity and Power Distribution Anomalies

Event	Class	Code	Acceptance Criteria
1. Uncontrolled RCCA bank withdrawal from a subcritical or low power startup condition	AOO	TWINKLE-M, VIPRE-01M	• max P_{RCS} & P_{MS} < 110% design • min DNBR > 95/95 DNBR limit*1
2. Uncontrolled RCCA bank withdrawal at power	AOO	MARVEL-M	• max P_{RCS} & P_{MS} < 110% design*1 • min DNBR > 95/95 DNBR limit*1
3. RCCA misalignment a. Dropped RCCA b. Static misalignment	 AOO AOO	MARVEL-M, VIPRE-01M*2	• max P_{RCS} & P_{MS} < 110% design • min DNBR > 95/95 DNBR limit*1,2 • site boundary dose limited to 10% of 10 CFR 100 values
c. Withdrawal of a single RCCA	PA	MARVEL-M, VIPRE-01M*2	• max P_{RCS} & P_{MS} < 110% design • min DNBR > 95/95 DNBR limit*1, *7 • site boundary dose limited to 10% of 10 CFR 100 values
4. Startup of an inactive reactor coolant pump at an incorrect temperature	AOO	N/A	N/A *3
5. CVCS malfunction that results in a decrease in boron concentration in the reactor coolant	AOO	N/A *4	• max P_{RCS} & P_{MS} < 110% design • min DNBR > 95/95 DNBR limit*1 • See Note *5
6. Spectrum of RCCA ejection accidents	PA	TWINKLE-M, VIPRE-01M, MARVEL-M	• max reactor pressure < ASME "Service Limit C" criteria • min DNBR > 95/95 DNBR limit*1, *7 • PCMI fuel failure*1, *6 • site boundary dose limited to 10 CFR 100 values • core coolability maintained *6

*1 Indicates the key parameter / acceptance limit of concern.
*2 Steady state analysis
*3 N-1 loop operation not allowed per plant Tech Specs

*4 This event is evaluated without the use of a computer code.

*5 The following minimum time intervals are available for operator actions between the time an alarm announces an unplanned moderator dilution and the time shutdown margin is lost:

A. During refueling: 30 minutes.

B. During startup, cold shutdown, hot shutdown, hot standby, and power operation: 15 minutes.

*6 The RCCA ejection (RIA) follows SRP 4.2 Appendix B

*7 If the DNBR falls below this value, fuel failure (rod perforation) must be assumed for all rods that do not meet these criteria unless it can be shown, based on an acceptable fuel damage model that includes the potential adverse effects of hydraulic instabilities, that fewer failures occur. If rod internal pressure exceeds system pressure, then fuel rods may balloon shortly after entering DNB.. The effect of ballooning fuel rods must be evaluated with respect to flow blockage and DNB propagation. Any fuel damage calculated to occur must be sufficiently limited to the extent that the core will remain in place and intact with no loss of core cooling capability.

4.6 Increase in Reactor Coolant Inventory

This category covers events that lead to fuel damage or over-pressurization of the RCS due to an unexpected increase in RCS inventory. The following table summarizes the two initiating events considered for the US-APWR, their associated event classification, and identifies the computer codes used to analyze the event for compliance with applicable codes and regulations.

Table 4.6-1 Events in Increase in Reactor Coolant Inventory

Event	Class	Code	Acceptance Criteria
1. Inadvertent operation of the emergency core cooling system during power operation	AOO	N/A	N/A*2
2. CVCS malfunction that increases reactor coolant inventory	AOO	MARVEL-M	• max P_{RCS} & P_{MS} < 110% design • min DNBR > 95/95 DNBR limit*1

*1 Indicates the key parameter / acceptance limit of concern.
*2 Safety injection pump shut off head is below normal operation pressure.

4.7 Decrease in Reactor Coolant Inventory

This category covers events that lead to accidental depressurization of the RCS. The following table summarizes the two initiating events considered for the US-APWR, their associated event classification, and identifies the computer codes used to analyze the event for compliance with applicable codes and regulations.

Table 4.7-1 Events in Decrease in Reactor Coolant Inventory

Event	Class	Code	Acceptance Criteria
1. Inadvertent opening of a pressurizer safety valve[*3]	AOO	MARVEL-M	• max P_{RCS} & P_{MS} < 110% design • min DNBR > 95/95 DNBR limit[*1]
2. Steam generator tube rupture	PA	MARVEL-M	• max P_{RCS} & P_{MS} < 110% design[*4] • min DNBR > 95/95 DNBR limit[*2] • SG does not fill [*1, *2] • site boundary dose limited to 10 CFR 100 values

[*1] Indicates the key parameter / acceptance limit of concern.
[*2] MHI internal acceptance criteria.
[*3] The non-LOCA scope of this accident includes the short-term analysis to evaluate fuel and NSSS response. The LOCA aspects of this accident are outside the scope of this non-LOCA analysis topical report.
[*4] MHI internal criterion to assure that rupture of primary or steam system piping does not occur.

5.0 EVENT-SPECIFIC METHODOLOGY

The objective of this topical report is to provide methods of analysis, details, and examples of the application of the MHI non-LOCA accident analysis to the US-APWR such that questions or issues can be identified as early as possible in the licensing process. As discussed in Section 2, the MARVEL-M, VIPRE-01M, and TWINKLE-M computer codes are the principal computer codes that will be used by MHI for the US-APWR non-LOCA analyses. Depending on the specific nature and computational capabilities needed for specific accidents, these programs are either used alone or in combination with another. Events utilizing computer codes for the non-LOCA accident analysis fall into one of the following three categories based on the combination of codes used:

- Analyzed using MARVEL-M only
- Analyzed using MARVEL-M and VIPRE-01M in sequence
- Analyzed using TWINKLE-M and VIPRE-01M in sequence

The first category that uses MARVEL-M alone includes most of the non-LOCA transients that challenge the design limits for the RCS and main steam system pressure limits, as well as loop-symmetric accidents at full-flow conditions that fall within the capabilities of the simplified MARVEL-M DNBR model. These accidents do not require detailed calculation of localized fuel parameters and do not require spatially dependent transient calculations for accident-specific power levels or power distributions. The RCCA Bank Withdrawal at Power event is such a transient.

The second category that uses MARVEL-M in combination with the VIPRE-01M fuel rod code is used for accidents that challenge the DNB design limits under reduced flow conditions such as the partial loss of flow, complete loss of flow, locked RCP rotor, or RCP sheared shaft conditions. The loop-dependent and core total flow, core inlet conditions, pressure and power are calculated using the MARVEL-M program, and then the VIPRE-01M code is used to determine the hot channel or hot spot fuel response including DNBR, fuel temperatures, and cladding temperature. The Complete Loss of Forced Reactor Coolant Flow is an example of such an event. The flow transient is calculated in the MARVEL-M code using the internal reactor coolant pump model in conjunction with the core and reactor coolant loop characteristics.

The third category that uses TWINKLE-M in combination with the VIPRE-01M fuel rod code is reserved for rapid reactivity transients requiring space- and time-dependent nuclear power and power distribution calculations for input to a detailed fuel response calculation. The Spectrum of RCCA Ejection event is an example of an event requiring these capabilities, including 3-D TWINKLE-M capabilities if needed.

There are several other events that make use of special models or code capabilities of the MARVEL-M code.

- The Steam System Piping Failure is a transient that is characterized by a non-uniform cooldown in combination with the assumption that the most reactive control rod be fully withdrawn. The steam line break flow calculation is unique to this event, and reactor vessel inlet mixing and reactivity weighting models in MARVEL-M are used to conservatively predict core reactivity and nuclear power using point kinetics. In addition, certain ECCS functions such as steamline isolation, EFWS actuation, feedwater isolation, and RCS boration using the safety injection system are modeled in this event. The VIPRE-01M code is used to confirm that the DNB design limit is met for selected steady state points characterized by temperature, pressure, power, power distribution, inlet temperature distribution, and flow conditions unique to the steamline break event.

- The Feedwater System Pipe Break is another non-uniform accident that involves modeling break flow from one of the secondary loops. The feedwater system pipe break models loss of inventory from the saturated liquid water mass in the steam generator and unlike the steamline break cooldown, results in RCS heatup and pressurization. This event also uses the 4-loop capability of the MARVEL-M code to model the failure of EFWS to feed one of the intact steam generators.
- The Steam Generator Tube Rupture event uses the MARVEL-M capability to calculate primary-to-secondary flow based on primary and secondary pressures calculated by the code. Operator actions to establish steam generator cooling using the non-faulted steam generators, manual opening of the steam generator relief valves, and manual opening of a pressurizer depressurization valve are also modeled by MARVEL-M for this accident.

In summary, the following six events have been selected to demonstrate the wide spectrum of key analytical methods (combinations of codes) and specialized models used by MHI in the non-LOCA accident analysis for the US-APWR. These events also represent the SRP accident categories for cooldown events (15.1), heatup events (15.2), flow reduction events (15.3), reactivity events (15.4), and reactor coolant inventory reduction events (15.6).

Analyzed using MARVEL-M only

- Uncontrolled RCCA Bank Withdrawal at Power

Analyzed using MARVEL-M / VIPRE-01M sequence

- Complete Loss of Forced Reactor Coolant Flow

Analyzed using TWINKLE-M / VIPRE-01M sequence

- Spectrum of RCCA Ejection

Requiring special treatment

- Steam System Piping Failure (VIPRE-01M core modeling)
- Feedwater System Pipe Break (4-loop MARVEL-M capability)
- Steam Generator Tube Rupture (primary-to-secondary flow model)

A detailed description of the methodology for each of these events is provided in separate subsections of Section 5, and sample transient results for each of the events are provided in Section 6.

5.1 Uncontrolled RCCA Bank Withdrawal at Power

Event Description

The uncontrolled RCCA bank withdrawal at power results in an increase in both the nuclear power and core heat flux. Because the heat removal from the steam generator lags the core power generation until the steam generator relief or safety valves open, there is an increase in the reactor coolant temperature. Unmitigated, the power increase and concurrent coolant temperature rise could eventually exceed a DNB, overpower, or RCS pressure limit. In order to avoid damage of the fuel cladding, the protection system trips listed below are designed to terminate any such transient before a design limit is exceeded.

This transient is categorized in AOO and the acceptance criteria are shown in Section 4.5.

Reactor Protection

The following automatic reactor trip signals are assumed to be available to provide protection from this transient:

- Neutron flux high trip (high setting)
- Neutron flux rate high trip
- Over power ΔT high trip
- Over temperature ΔT high trip
- Pressurizer pressure high trip
- Pressurizer water level high trip

The reactor protection system overpower and overtemperature ΔT trips are designed to provide margin to the core protection design limits.

Method of Analysis

(1) Analysis Code

The MARVEL-M code is used to determine the plant transient following an uncontrolled RCCA bank withdrawal at power. A reactivity insertion into the core is simulated by external reactivity. Minimum DNBR is calculated by the MARVEL-M code using DNBR data tables with average and hot spot heat flux, RCS pressure, and core inlet temperature. The DNBR data tables are made up of several pre-calculated conditions using the VIPRE-01M code with an assumed constant core flow rate. A suitable rod bundle DNB correlation and the Revised Thermal Design Procedure (RTDP) [Reference 14] are used for a DNBR evaluation.

Comparison of the MARVEL-M DNBR calculation using the DNBR data tables with the VIPRE-01M DNBR calculation for this transient is shown in Appendix A.

(2) Analysis Assumptions

Analysis assumptions and calculation conditions for this event are as follows.

(a)	Initial condition	Nominal power, T_{avg}, RCS pressure for DNB evaluation
(b)	Power distribution	Design power distribution
(c)	Moderator Density Coefficient	Least positive (BOC), Most positive (EOC)
(d)	Doppler Power Coefficient	Least negative (BOC), Most negative (EOC)
(e)	Trip Parameters	Conservative reactivity insertion curve and trip delays

(f) Reactor protection	Neutron flux high trip and Over temperature ΔT high trip are assumed.

(3) Calculation Case

Analyses for DNBR evaluation are performed for a range of reactivity insertion rates ranging from a small reactivity insertion rate through the maximum reactivity insertion rate of 75 pcm/sec using combinations of the following feedback conditions.

- Beginning of cycle (BOC), meaning a minimum feedback condition
- End of cycle (EOC), meaning a maximum feedback condition

Analyses for peak RCS pressure are also performed for this accident. The analysis assumptions such as initial conditions and core parameters are selected to maximize peak RCS pressure.

5.2 Complete Loss of Forced Reactor Coolant Flow

Event Description

A complete loss of forced reactor coolant flow accident results from a simultaneous loss of electrical supplies to the reactor coolant pumps. If the reactor is at power at the time of the transient, the immediate effect of a loss of coolant flow is a rapid increase in the coolant temperature, and the minimum DNBR decreases. The reactor protection trips listed below are available to provide protection for this event. In the analysis, this transient is terminated by the Reactor coolant pump speed low trip to prevent a DNB occurrence.

This transient is categorized in AOO and the acceptance criteria are shown in Section 4.4.

Reactor Protection

The following signals are assumed to be available to automatically trip the reactor and therefore provide protection from this transient.

- Reactor coolant pump speed low trip
- Reactor coolant flow low trip

Method of Analysis

(1) Analysis Code

The MARVEL-M code's built-in reactor coolant pump model is used to determine the plant transient following a complete loss of forced reactor coolant flow. The MARVEL-M code generates an interface file that includes the time-dependent histories of the nuclear power and core inlet flow rate.

The VIPRE-01M code calculates the minimum DNBR during the transient using this interface as a boundary condition assuming a constant design power distribution. A constant RCS pressure and inlet temperature is used in the DNBR calculation for conservatism. A subchannel analysis using VIPRE-01M for the typical 4-loop plant with 17x17-257FA core is performed using a one-eighth core model with a hot assembly located at the center of the core. This model assumes that the radial power distribution and inlet flow distribution are symmetric with respect to the core center [Reference 6]. A suitable rod bundle DNB correlation and Revised Thermal Design Procedure (RTDP) are used.

VIPRE-01M one-eighth core model is shown in Figure 5.2-1 and a calculation flow diagram of the MARVEL-M / VIPRE-01M methodology is shown in Figure 5.2-2.

(2) Analysis Assumptions

Analysis assumptions and calculation conditions for the MARVEL-M analysis are as follows.

(a) Initial condition	Nominal power, T_{avg}, RCS pressure for DNB evaluation
(b) Moderator Density Coefficient	Least positive
(c) Doppler Power Coefficient	Most negative
(d) Trip Parameter	Conservative reactivity insertion curve and trip delays
(e) Others	Conservative inertia momentum of the RCP flywheel

Analysis assumptions and calculation conditions for the VIPRE-01M analysis are as follows:

(f) Power distribution	Design limit of the nuclear enthalpy rise hot channel

	factor and design axial power shape
(g) RCS Pressure	Constant
(h) Core Inlet Temperature	Constant

(3) Calculation Case

Analyses for DNBR evaluation are performed at the beginning-of-cycle (BOC) for the hot full power condition.

Figure 5.2-1 VIPRE-01M 1/8 Core Analysis Modeling (17x17-257FA Core, 4-Loop Plant)

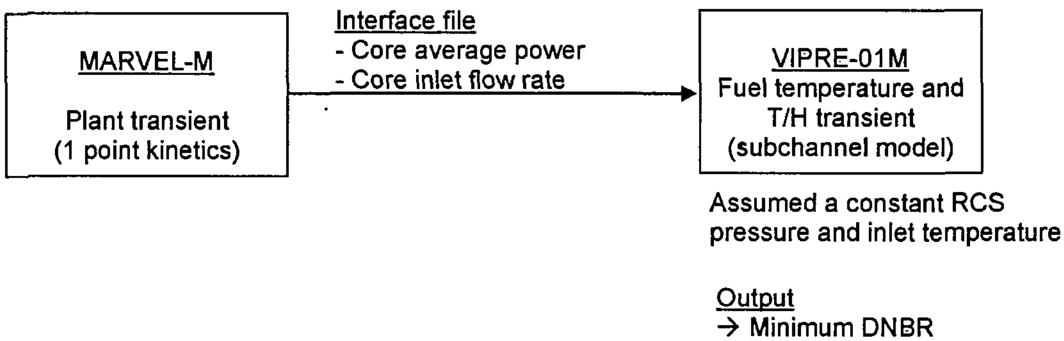


Figure 5.2-2 Calculation Flow Diagram of the MARVEL-M / VIPRE-01M Methodology

5.3 Spectrum of RCCA Ejection

Event Description

This accident is defined as the mechanical failure of a control rod drive mechanism pressure housing, resulting in the ejection of an RCCA and drive shaft. The consequence of this RCCA ejection is a rapid positive reactivity insertion with an increase of core power peaking, possibly leading to localized fuel rod failure. The nuclear excursion is terminated by Doppler reactivity feedback from increased fuel temperature, and the core is shut down by the neutron flux high trip (high and low setting for HFP and HZP, respectively).

This accident is categorized as a PA, and the acceptance criteria are shown in Section 4.5.

Reactor Protection

The following signals are assumed to be available to automatically trip the reactor and therefore provide protection from this transient.

- Neutron flux high trip (high setting)
- Neutron flux high trip (low setting)
- Neutron flux rate high trip

In the safety analysis, the neutron flux rate high trip is ignored.

Method of Analysis

(1) Analysis Code

The TWINKLE-M code is used to determine the core transient including core average and local power behavior following a RCCA ejection. An increase of local power and the Doppler feedback due to an increase of fuel effective temperature are calculated in each spatial mesh.

The three-dimensional method is applied to the hot zero power condition in order to conform to the PCMI (Pellet Cladding Mechanical Interaction) fuel failure criteria, which was lowered and expressed as a function of fuel oxide / wall thickness in SRP Chapter 4.2 Rev.3 Appendix B. Core mesh division is 2 x 2 meshes per assembly in the radial direction and( )in the axial direction for the active core region for the typical 4-loop plant with 17x17-257FA core. For the hot full power case, a one-dimensional method is applied using( )in the axial direction for the active core region.

The VIPRE-01M code calculates the fuel temperature and fuel enthalpy at the hot spot during the transient using two interface files created by the TWINKLE-M code. One of the interface files is a time-dependant history of the core average power and the other is a time-dependant history of the hot channel factor. The hot channel factor time history is used for the three-dimensional calculation only. The VIPRE-01M analysis uses a one-eighth core model shown in Figure 5.2-1.

The MARVEL-M code is used to calculate the RCS pressure transient using the VIPRE-01M results which are core total void and heat flux histories.

(2) 3-D Methodology

A calculation flow diagram of the three-dimensional methodology including the hot spot temperature analysis and the PCMI fuel failure evaluation is shown in Figure 5.3-1.

Hot spot fuel temperature analysis

PCMI fuel failure evaluation

(a) A local adiabatic fuel enthalpy rise (ΔH) is calculated in the TWINKLE-M code by integration of local power and power density (cal/g-s) in each mesh. This ΔH is considered a peak / average ratio in the mesh using the VIPRE-01M hot spot results. In this way, a relation of between the fuel enthalpy rise (ΔH) and the local burnup is obtained.
(b) A relation of between the local oxide / wall thickness and the local fuel burnup is evaluated *in fuel design*.
(c) Then, a relation of between the fuel enthalpy rise (ΔH) and the local oxide / wall thickness can be obtained. The fuel integrity is confirmed by comparing the calculated fuel enthalpy rise and oxide / wall thickness data with the new PCMI fuel failure criteria.

The three-dimensional calculation is generally the most realistic method to predict localized fuel behavior. The MHI three-dimensional methodology used in the hot zero power RCCA ejection safety analysis is established based on the following separate calculational conservatisms:

A sensitivity study of conservatism of the reactivity and the hot channel factor treatment is shown in Appendix B.

(3) 1-D Methodology

A calculation flow diagram of the one-dimensional methodology for the hot spot temperature analysis is shown in Figure 5.3-2. The number of DNB rods methodology is shown in Figure 5.3-3 and the RCS pressure methodology is shown in Figure 5.3-4.

Hot spot fuel temperature analysis

(a) The TWINKLE-M code analyzes the RCCA ejection using a one-dimensional model for the hot full power initial condition. The reactivity insertion to the core is simulated by an external reactivity insertion by changing the eigenvalue of the neutron kinetics. Other conservative assumptions are described in (4).

(b) The TWINKLE-M code outputs the interface file for the VIPRE-01M code including histories of core average power. The hot channel factor used as the VIPRE-01M input data is assumed a constant during the transient after a RCCA ejection.

(c) The VIPRE-01M code reads the interface file and calculates a fuel temperature transient at the hot spot with conservative parameters and assumptions described in (5). Then a maximum fuel center temperature, a peak fuel enthalpy and a peak clad temperature are obtained.

DNB rods analysis

(a) The TWINKLE-M code analyzes the core average power histories using the same manner as the hot spot analysis without initial uncertainty.

(b) A sensitivity study of a hot channel factor is performed by VIPRE-01M to just give DNB occurrence using a suitable rod bundle DNB correlation and RTDP.

(c) A census of power distribution after a RCCA ejection is created by core design.

(d) Number of DNB rods can be obtained from the census and the hot channel factor just giving DNB occurrence.

RCS pressure analysis

(a) The TWINKLE-M code analyzes the core average power histories using the same manner as the hot spot analysis.

(b) The VIPRE-01M code analyzes the fuel temperature and the thermal hydraulics using the same manner as the hot spot analysis.

(c) The VIPRE-01M code generates the interface file including a time-dependant core total void fraction and core heat flux.

(d) The MARVEL-M code analyzes a plant transient for maximum RCS pressure using the interface file generated by VIPRE-01M.

(4) Analysis Assumptions for the Core Kinetics

Analysis assumptions and calculation conditions are as follows. These conditions follow Regulatory Guide 1.77 Appendix A.

(a) Initial Condition

24 month equilibrium core at the beginning-of-cycle (BOC) and end-of-cycle (EOC)

- Hot full power with initial uncertainty for fuel temperature evaluation
- Hot zero power for fuel enthalpy evaluation

(b) Reactivity Insertion

A conservative large reactivity, chosen at the design limit is inserted within 0.1 seconds. In the case of three-dimensional methodology, the most reactive RCCA ejection is selected. The difference in reactivity compared with the design limit is externally added to the core by changing the eigenvalue of the neutron kinetics.

In the case of one-dimensional methodology, the design limit is externally added to the core within 0.1 seconds.

(c) Doppler Feedback

The Doppler feedback is applied as a conservative multiplier to the change in the fast absorption cross-section for the given change in the calculated fuel effective temperature.

(d) Doppler Weighting Factor for 1-D Method

In the MHI one-dimensional methodology, a small Doppler weighting factor is used to compensate for collapsing the 3-D problem into a 1-D axial model. The suitability and conservatism of this approach is confirmed by a comparison between the three-dimensional and one-dimensional kinetic results presented in Appendix C.

(e) Trip Parameters
The reactor trip is simulated by dropping partially and fully withdrawn rod banks into the core. Maximum time delay from reactor trip signal to rod motion and a conservative RCCA insertion curve are simulated. The trip reactivity used is the design limit, which is -4%ΔK/K for the hot full power case and -2%ΔK/K for the hot zero power case, respectively.

(f) Other Parameters
Minimum delayed neutron fraction and minimum neutron lifetime are used.

(5) Analysis Assumptions for the Fuel Temperature Transient
Analysis assumptions and calculation conditions are as follows.

(a) Initial Condition
Same as (4)(a)

(b) Hot channel factor

[ ]

In the case of one-dimensional methodology, the hot channel factor is assumed to instantaneously increase to the design limit and is conservatively assumed to remain constant, ignoring feedback effects during the transient.

(c) Fuel properties
Initial condition of fuel temperature at hot spot is consistent with the results of the fuel design code FINE [Reference 15]. Pellet and cladding gap conductance in the transient are assumed to be conservative values according to the evaluation purpose.

- Remains constant after initial for fuel temperature and enthalpy analysis
- Instantaneously decreases to zero for the adiabatic fuel enthalpy analysis
- Rapidly increases to the maximum value for the clad temperature analysis
- Realistic increases for the DNB rods and RCS pressure analysis

(6) Calculation Cases
Analyses of the spectrum RCCA ejection are performed for the following cases.

- Hot full power initial condition at beginning-of-cycle
- Hot full power initial condition at end-of-cycle
- Hot zero power initial condition at beginning-of-cycle
- Hot zero power initial condition at end-of-cycle

Figure 5.3-1 Calculation Flow Diagram of the Three-Dimensional Methodology

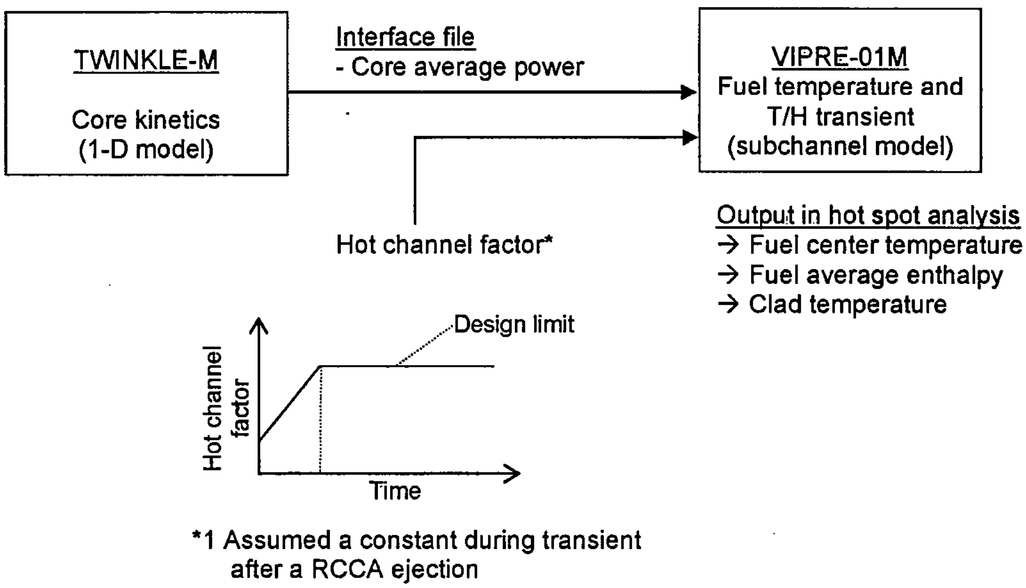


Figure 5.3-2 Calculation Flow Diagram of the One-Dimensional Methodology

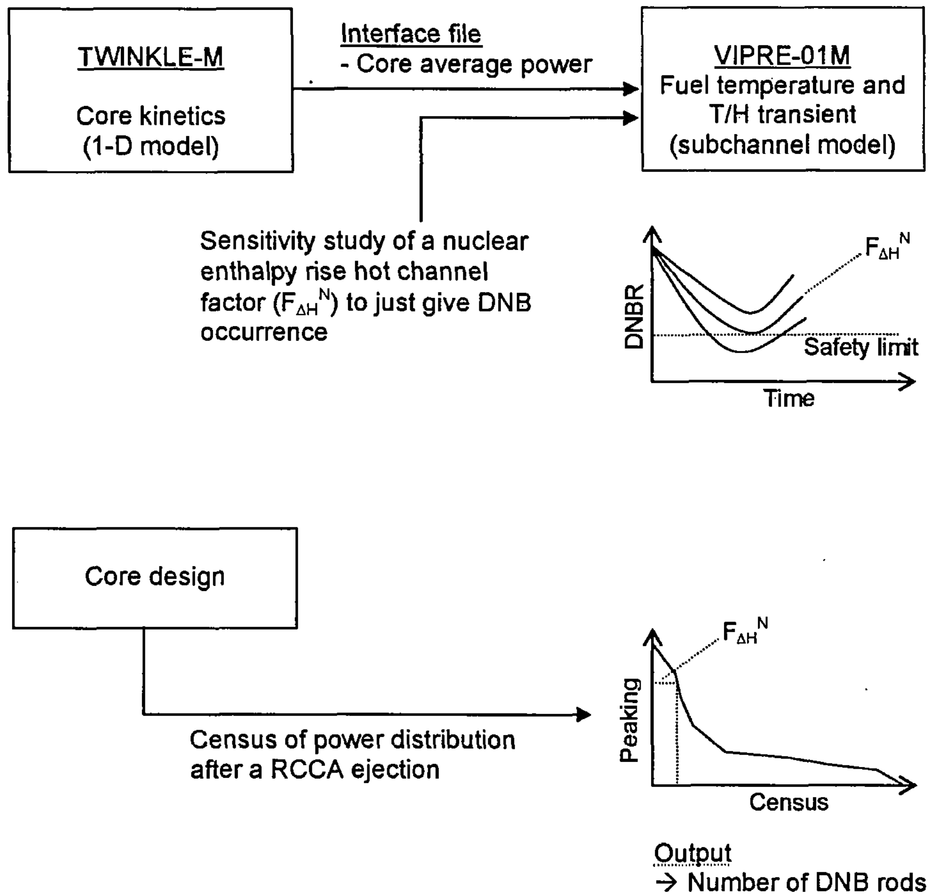


Figure 5.3-3 Calculation Flow Diagram of the DNB Rods Number Methodology

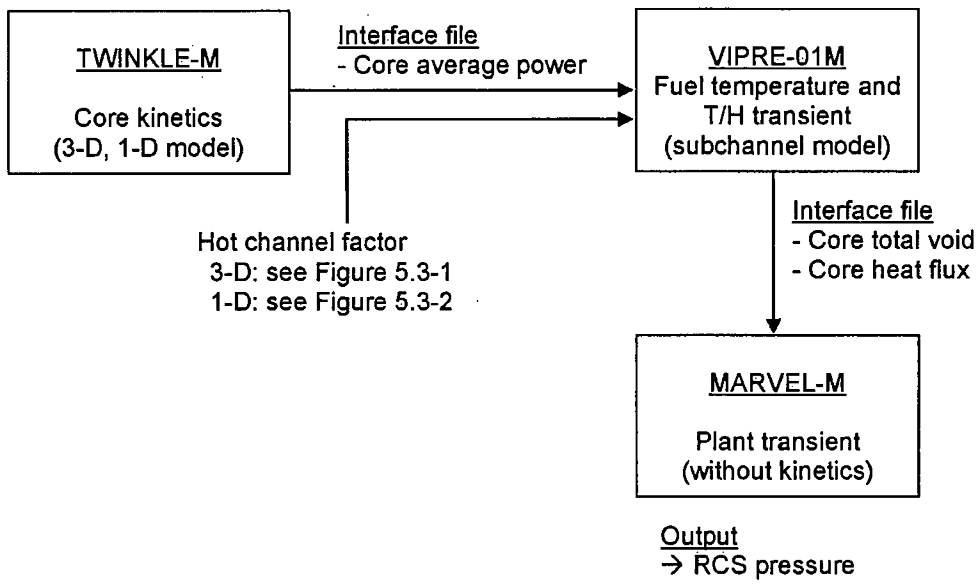


Figure 5.3-4 Calculation Flow Diagram of the RCS Pressure Methodology

5.4 Steam System Piping Failure

The Steam System Piping Failure is a transient that is characterized by asymmetric power generation in the core due to a non-uniform cooldown, which is caused by single steam system piping failure in combination with the assumption that the most reactive control rod be fully withdrawn. The steam line break flow calculation is unique to this event. The specific models for the core inlet mixing and consequent reactivity weighting are used in MARVEL-M analysis to conservatively predict core reactivity and nuclear power using point kinetics. In addition, the reactor coolant flow condition can be natural circulation. Certain Emergency Core Cooling System (ECCS) functions such as steam line isolation, EFWS actuation, feedwater isolation, and RCS boration using the safety injection system are included in the analysis. The VIPRE-01M code is used to calculate minimum DNBR and confirm that the DNB design basis is met in conjunction with other parameters calculated by the MARVEL-M code.

The steam system piping and valve arrangement for US-APWR is shown in Figure 5.4-1.

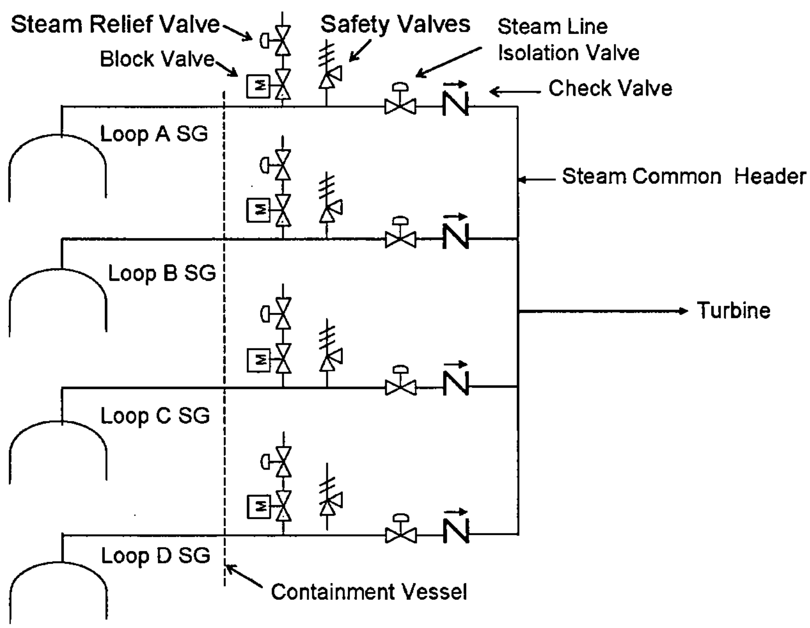


Figure 5.4-1 Steam System Configuration of US-APWR

A steam line break at a steam generator exit nozzle located at inside of the containment and a pipe break at the steam common header are postulated for Chapter 15 accident analysis.

<u>Event Description</u>

1) Steam piping failure inside containment
   A double-ended steam pipe break at a SG exit nozzle is assumed. The break causes

uncontrolled steam release from the faulted SG into the containment until the SG is dried out. Assuming the check valve in the faulted SG steam line does not function as designed, the model accounts for steam release from the other SGs until the steam line isolation valve associated with the faulted SG is completely closed.

2) Steam piping failure outside containment
A double ended pipe break at the steam common header is assumed. The steam pipe break causes steam release from all the SGs until the steam line isolation valves are closed. One of the steam line isolation valves is assumed to fail to close and the steam release continues from that SG until the SG is dried out.

Although the above two steam system piping failure scenarios are different, the effects on the reactor coolant system are very similar.

The major steam pipe rupture results in an initial increase in steam flow, which decreases during the accident as the steam pressure falls. Although an integral flow restrictor is installed in the SG exit nozzle to mitigate the steam flow, a double-ended steam line break causes a large steam flow from the faulted SG to induce a rapid cooldown of the SG secondary side. The energy removal from the reactor coolant system causes a reduction in coolant temperature and pressure. The colder fluid in the loop with the faulted SG is mixed with the flow from the other intact loops. The core inlet temperature distribution and the cooldown of the core water is non-uniform due to the imperfect mixing of the loop flows in the reactor vessel inlet.

In the presence of a negative moderator temperature coefficient, the cooldown results in an insertion of positive reactivity. The effect is the largest at the end of core cycle. If the event occurs at nominal operating conditions, a core power increase results. If the event occurs at hot zero power condition and the most reactive RCCA is assumed stuck in its fully withdrawn position after reactor trip, there is an increased possibility that the core becomes critical and returns to power. A return to power following a steam line rupture is a potential problem mainly because of the existing high radial power peaking factors, assuming the most reactive RCCA to be stuck in its fully withdrawn position.

When the steam pressure in the failed steam generator falls below the Main Steam Line Pressure Low setpoint (in any loop), the ECCS is actuated. The ECCS signal also actuates functions such as EFWS, steam line isolation, and feedwater isolation to isolate the failed SG.

The core is ultimately shut down by a combination of the high concentration boric acid water delivered by the ECCS and the termination of the cooldown when the steam generator inventory is depleted.

The large double-ended steam pipe rupture is categorized as a PA and the acceptance criteria are described in Section 4.2. The MHI analysis conservatively uses a criterion of no DNB for the limiting steam line break, to preclude DNB propagation in the low pressure environment of the fuel.

The steam system piping failure inside the containment causes a containment pressure and temperature increase due to the steam release. The mass and energy release is calculated by the MARVEL-M code as a function of time for analysis of the containment integrity.

Reactor Protection

The following signals are assumed to be available to automatically trip the reactor and therefore provide protection from this transient.

- ECCS Actuation (low steam line pressure signal in any loop or low pressurizer pressure or high-1 containment pressure)
- Over power ΔT high trip
- Over temperature ΔT high trip
- Pressurizer pressure low trip
- Neutron flux high trip

Engineered Safeguards Features

The following features are assumed to be available to mitigate the accident.

- Steam line isolation
- EFWS isolation
- Safety injection
- Main feedwater isolation

Method of Analysis

(1) Analysis code

The MARVEL-M, ANC and VIPRE-01M codes are used for this steam system piping failure analysis. A calculation flow diagram is shown in Figure 5.4-2.

(a) System analysis by the MARVEL-M code

The MARVEL-M code is used to analyze the plant transient following steam piping ruptures. The break flow rate from the SGs is calculated using the Moody correlation. The released steam is conservatively assumed saturated and dry without moisture carry-over, since steam release without carry-over causes the maximum energy release and cooldown.

The overall primary-to-secondary heat transfer coefficient in the steam generators is modeled in the code by the four major thermal resistance components: the primary convection film resistance, the tube metal resistance, the fouling resistance, and the secondary side boiling heat transfer resistance, taking account of the dependency on the relevant operating conditions, such as temperature, pressure and flow. The model is applicable over the wide range of the operating conditions characteristic of the SG during a steam pipe break event.

The RCS model in the code can analyze the non-uniform reactor system transient response to the event. The steam system model in the code can simulate steam flow redistribution from SGs (described in Section 2.1.3.4). The flow mixing in the reactor vessel is modeled in the code. The mixing factors for the reactor vessel inlet and outlet plenums are defined conservatively by the input referring to the mixing test results by the 1/7 scale reactor vessel model (Section 2.1.3.2).

A weighting factor for reactivity feedback can be also input to take account of the azimuthal tilt of the core coolant properties.

The safeguards system and the ECCS sub-system necessary for such non-LOCA analysis are modeled in the code.

(b) DNBR calculation
In the hot zero power condition, the VIPRE-01M code calculates the minimum DNBR. These DNBR calculations are steady state calculations at pre-selected conditions, using the MARVEL-M calculated values of core average heat flux, RCS pressure, core inlet coolant temperatures, core inlet flow rate and boron concentration, for a certain number of state points around the time the highest core average heat flux is reached. Additionally, the core inlet coolant enthalpy distribution coupled with core power distribution, which is calculated by the ANC code considering a steady-state condition assuming a stuck rod, is also input to the VIPRE-01M code. The history files used in the more standard MARVEL-M / VIPRE-01M sequences are not used for the steam piping failure. A suitable bundle DNB correlation is used at the low RCS pressure conditions characteristic of this accident.

A VIPRE-01M subchannel analysis is performed using a one-eighth core model with a hot assembly assumed at the center of the core as shown in Figure 5.2-1 as well as the other DNB concerned transients. However the fuel rods and flow channels are divided into 5 groups shown in Figure 5.4-3 to express the power distribution and inlet enthalpy distribution. Each group is associated with the hot channel, the neighbor channels to hot channel, the remains in the hot assembly, the neighbor assemblies to hot assembly and the remains in the core, respectively. Radial power distributions calculated by the ANC code and inlet enthalpy distribution are averaged for each group. Axial power distributions are represented by a 3 shapes, which are associated with the hot assembly (group 1 through 3), neighbors to the hot assembly (group 4) and the remains (group 5). This symmetric model is validated by the comparison with the asymmetric full core model as described in Appendix D. This comparison demonstrates that the symmetric model can provide minimum DNBR with sufficient accuracy during the steam piping failure transient.

For the hot full power condition, the MARVEL-M code calculates the minimum DNBR using its internal DNBR data tables, with core average heat flux, RCS pressure, and core inlet temperature in the same manner as is used for the RCCA Bank Withdrawal at Power described in Section 5.1. The internal DNBR table used is evaluated by using RTDP and applicable rod bundle DNB correlation. This methodology is acceptable, since the core operating condition is within the range of the pre-evaluated DNBR table in MARVEL-M, because the minimum DNBR occurs within a short time after the reactor trip is initiated. The one rod stuck assumption is considered in defining the shutdown reactivity, but is not meaningful for the period up to reaching the minimum DNBR in the at-power transients of this kind.

(c) Mass and Energy Release Analysis
The MARVEL-M code is used to generate the mass and energy release data as a function of time for the case of Steam Piping Failure inside containment, taking into consideration the following models:

i) The RCS thick metal effect in the MARVEL-M code is used.
ii) When the SG steam pressure decreases below the saturation pressure of the hot feedwater remaining in the feedwater system piping of the faulted SG, it could flash into the shell side of the SG through the feedwater nozzle. This results in an increased mass and energy release to the containment. The effect is calculated by modeling a single mass volume attached to the SG secondary side in the MARVEL-M code for this purpose.

The containment response due to the release of the mass and energy is not addressed in this topical report.

(2) Analysis Assumptions

Analysis assumptions and calculation conditions used in MARVEL-M for the analysis of the core response to the double-ended break from the hot zero power condition are as follows:

(a) Break Size: Double-ended rupture
(b) Moderator Density Coefficient: Maximum positive considering a stuck rod effect
(c) Doppler Power Coefficient: Minimum considering a stuck rod effect
(d) Shutdown Margin: Minimum considering the most reactive rod stuck out of the core
(e) Core inlet mixing modeled to reflect non-uniform effects based on the 1/7 scale reactor vessel model. Sensitivity study of inlet mixing coefficient is shown in Appendix E. The minimum DNBR is not extremely sensitive to small changes of the mixing coefficient near the expected value.
(f) Single Failure: Safety Injection train
(g) Reactivity weighting factor for fluid properties is considered
(h) Steam Quality: Dry steam (100%)

Analysis assumptions and calculation conditions used in VIPRE-01M for the analysis of the core response to the double-ended break from the hot zero power condition are as follows:

(i) Power Distribution: Calculated by detailed core analysis considering a stuck rod in the cold core sector

Comparison of the results from the detailed core analysis with the MARVEL-M predictions verifies the overall conservatism of the methodology. That is, the specific power, temperature, and flow conditions used to perform the DNB analysis are conservative.

Analysis assumptions and calculation conditions used in MARVEL-M for the analysis of the core response to a spectrum of break sizes from the hot full power condition are as follows:

(a) Moderator Density Coefficient: At EOC not considering a stuck rod
(b) Doppler Power Coefficient: Least negative
(c) Minimum Trip Reactivity: Same as RCCA bank withdrawal at power
(d) Power Distribution: DNBR data input to MARVEL-M
(e) Core Inlet Mixing: Same as HZP scenario
(f) Single Failure: RPS train (response unaffected)
(g) Steam Quality: Dry steam (100%)

Analysis assumptions and calculation conditions used in MARVEL-M for the analysis of the mass and energy release evaluation are the same as the assumptions above except thick metal effects and feedwater line water flashing due to SG depressurization are modeled.

(a) Moderator Density Coefficient: At EOC not considering a stuck rod
(b) Doppler Power Coefficient: Least negative
(c) Minimum Trip Reactivity: Same as RCCA bank withdrawal at power
(d) Power Distribution: Not applicable
(e) Core Inlet Mixing: Same as HZP scenario
(f) Single Failure: Safety Injection train
(g) Reactivity Weighting Factor: Same as HZP
(h) Steam Quality: Dry steam (100%)

(3) Calculation Case

Safety analysis is performed for the following cases in the spectrum of steam system piping failures.

Hot full power initial condition at end of cycle

Hot zero power initial condition at end of cycle (with and without offsite power)

DNBR, M&E

Post-scram: double ended rupture, EOC hot zero power

Pre-scram: Spectrum of break sizes and power levels, EOC

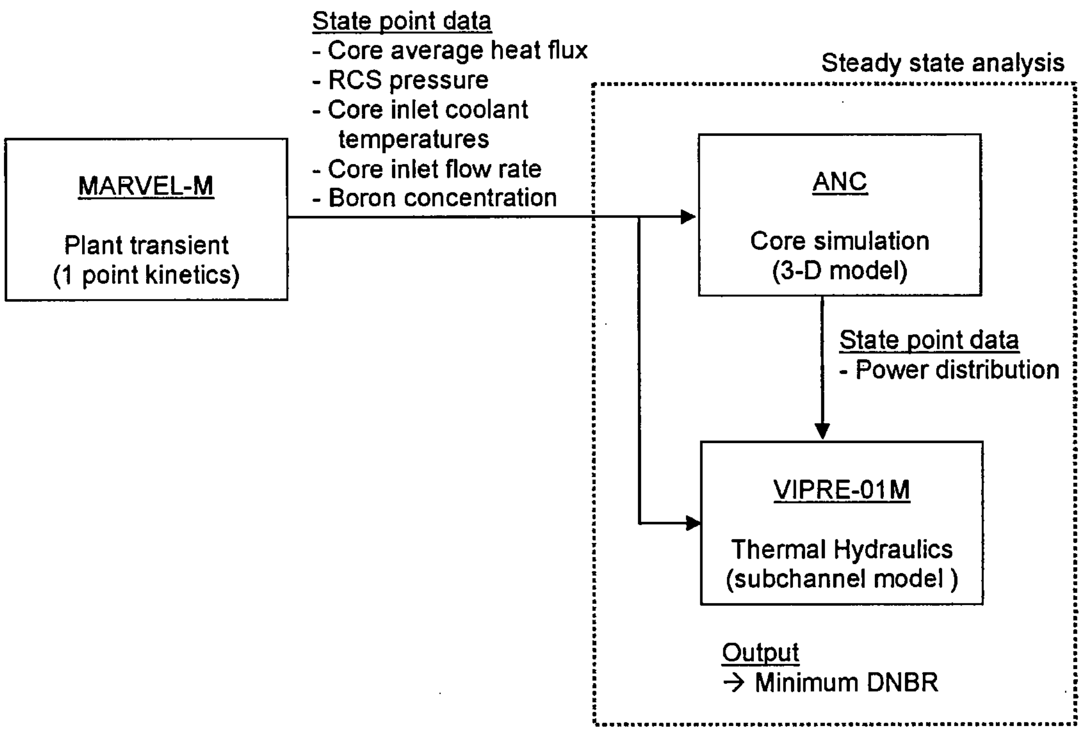


Figure 5.4-2 Calculation Flow Diagram of the Steam System Piping Failure Methodology for the Hot Zero Power Condition

Figure 5.4-3 VIPRE-01M 1/8 Core Analysis Modeling with 5-grouped Power Distributions (17x17-257FA Core, 4-Loop Plant)

5.5 Feedwater System Pipe Break

The Feedwater System Pipe Break is another non-uniform accident that involves modeling break flow from one of the secondary loops. The feedwater system pipe break causes loss of inventory from the saturated liquid water mass in the steam generator and unlike the steamline break cooldown, results in RCS heatup and pressurization. This event also uses the 4-loop capability of the MARVEL-M code to model the failure of EFWS to feed one of the intact steam generators.

Event Description

A major feedwater line rupture is a break in a feedwater line large enough to prevent the addition of sufficient feedwater to the steam generators in order to maintain shell-side fluid inventory in the steam generators. If the break is postulated in a feedwater line between the check valve and the steam generator, fluid from the steam generator may also be discharged through the break.

The feedwater line rupture reduces the ability to remove heat generated by the core from the reactor coolant system for the following reasons:

- Feedwater flow to the steam generators is reduced. Because feedwater is subcooled, its loss may cause reactor coolant temperatures to increase prior to reactor trip.
- Fluid in the steam generator may be discharged through the break and would not be available for decay heat removal after trip.
- The break may be large enough to prevent the addition of main feedwater after trip.

For breaks that are small enough to not be considered a major feedwater line rupture, the plant continues to operate without the need for reactor protection system or engineered safeguards feature actuation. For breaks resulting in continued feedwater addition but at a rate insufficient to maintain steam generator level, the loss of normal feedwater response will bound these breaks. In the limiting case, the double-ended rupture inside the main feedwater check valve will bound the remaining larger breaks.

The double-ended feedwater system pipe break accident is categorized as a PA and the acceptance criteria are described in Section 4.3.

If the postulated double-ended feedwater system pipe break occurs, the RCS heats up and the pressurizer water level and pressure increase. Unless the heatup of the RCS is mitigated, there will be possibility of water relief through the pressurizer safety valve.

The protective actions to mitigate the accident are isolation of the failed SG by closing the feedwater isolation valve and terminating the emergency feedwater supply to the faulted SG, and cooling of the RCS by supplying emergency feedwater to the intact SGs. The emergency feedwater system (EFWS) has two motor-driven and two turbine-driven emergency feedwater pumps. The associated valve arrangement is shown in Figure 5.5-1.

Each emergency feedwater pump supplies emergency feedwater independently to each SG taking water from the emergency feedwater pits. The EFWS is sized to have the capability of supplying sufficient emergency feedwater to preclude the pressurizer filling with water during a postulated feedwater system pipe break, assuming a single failure in one of the sub-systems of the EFWS. The protective actions are automated for the US-APWR.

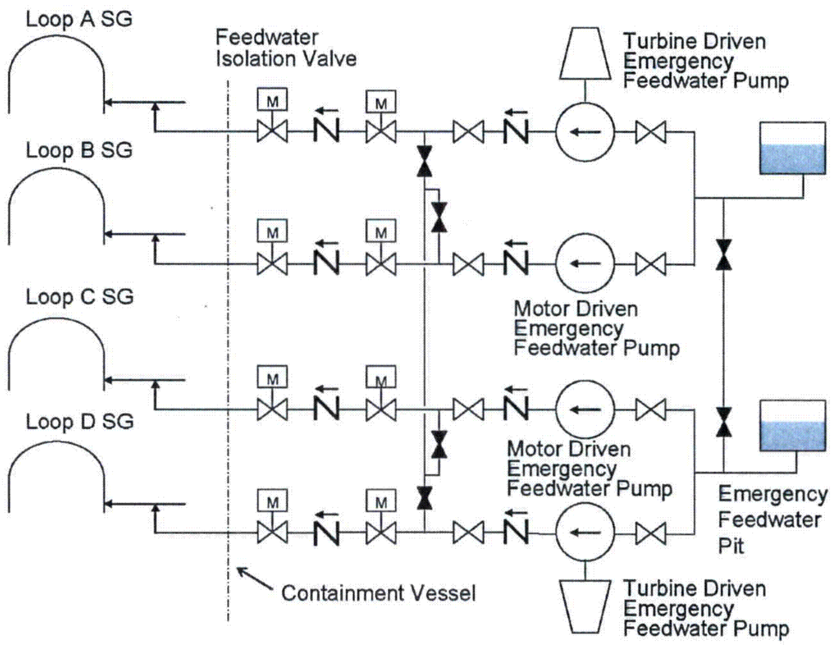


Figure 5.5-1 Emergency Feedwater System of US-APWR

Reactor Protection

The following signals are assumed to be available to automatically trip the reactor and therefore provide protection from this transient.
- Steam generator water level low trip in any loop
- Pressurizer pressure high trip
- Pressurizer water level high trip

Engineered Safeguards Features

The following features are assumed to be available to mitigate the accident.
- EFWS
- EFWS isolation
- Safety Injection

Method of Analysis

(1) Analysis code
The MARVEL-M code is used to determine the plant transient following a feedwater line rupture. The code describes the reactor thermal kinetics, reactor coolant system (including natural

circulation), pressurizer, steam generators, and feedwater system responses. MARVEL-M also computes related variables, including the pressurizer pressure, pressurizer water level, and reactor coolant average temperature.

The feedwater system pipe break causes non-balanced operation, e.g. a faulted SG loop, intact loops with emergency feedwater supply and an intact loop without emergency feedwater supply. The capability to model up to 4 separate loops in the MARVEL-M code is used for the analysis.

The break flow at the feedwater system pipe break is conservatively calculated by the Moody correlation, taking account of the flow restriction at the feedwater inlet nozzle.

In reality, the feedline water discharge could entrain steam when the water level in the faulted SG decreases significantly. In this case, the energy removed from the RCS from the steam release could mitigate the heatup of the RCS. This effect is conservatively neglected in the analysis to present the worst-case RCS heatup results for the feedwater system pipe break event.

(2) Analysis Assumptions

Analysis assumptions and calculation conditions are as follows.

(a)	Break Size and Location:	Double-ended rupture downstream of the feedwater line check valve
(b)	Initial SG Mass:	Loss of normal feedwater is assumed at 0 second until the reactor trip
(c)	Reactor trip:	SG water level low trip
(d)	Break Flow and Timing:	Feedwater pipe break is assumed to occur just after trip
(e)	Break Quality:	0% (saturated liquid)
(f)	Limiting Single Failure:	EFWS train failure is assumed (One SG is ruptured and associated EFWS is not effective. Another SG is intact and without EFWS, due to the single failure. The remaining SGs are intact and are supplied with EFWS.)
(g)	EFWS Isolation to Faulted SG:	Automatic isolation of EFWS to faulted SG by secondary low pressure signal

Analyses for peak RCS pressure are performed for this accident. The analysis assumptions such as initial conditions and core parameters are selected for this analysis to maximize peak RCS pressure.

Due to the steam line check valve feature, the faulted SG pressure will decrease rapidly after the reactor trip is initiated. The pressure in the intact SGs pressure will go up to relief or safety valve setpoints.

(3) Calculation Case

One case is analyzed, the double-ended rupture of the main feedwater pipe at the beginning of cycle (BOC) from hot full power conditions with maximum decay heat assuming loss of offsite power at time of turbine trip.

5.6 Steam Generator Tube Rupture

The Steam Generator Tube Rupture event uses the MARVEL-M capability to calculate primary-to-secondary flow based on the primary and secondary pressures calculated by the code. The SGTR event involves the loss of the reactor coolant due to the leakage to the ruptured SG secondary side, which causes a decrease in the pressurizer water level, eventually emptying the pressurizer. The decrease in the pressurizer pressure may actuate the ECCS. Operator actions to establish steam generator cooling using the intact steam generators, manual opening of the steam generator relief valves, and manual opening of the pressurizer depressurization valve are also modeled by MARVEL-M for this accident.

<u>Event Description</u>

The accident examined is the complete severance of a single steam generator tube. The accident is assumed to take place at power. The SG tube rupture causes the reactor coolant to leak to the SG secondary side through the double-ended breaks from the inlet side and the outlet side. The largest break flow occurs when the rupture location is near the tube sheet in the colder side. The steam output of the ruptured SG may be contaminated with the radioactivity of the leaked primary coolant water. The SG tube break can be detected by N-16 radiation detectors installed on the main steam line that alarms the occurrence of the SG tube rupture. The SG leak causes a reduction in the pressurizer pressure that may trip the reactor by either a Pressurizer Pressure Low signal or a Steam Generator Water Level High-High signal from the ruptured SG. The leak rate is comparatively small for a large 4-loop PWR plant such as the US-APWR. The operator may manually trip the reactor, if the automatic actuation of a reactor trip is delayed. If the pressurizer pressure decreases below the Pressurizer Pressure Low-Low setpoint, the ECCS is actuated. The ECCS signal trips the RCS pumps which then coast down to natural circulation flow. The actuation of the high pressure safety injection sub-system of the ECCS tends to prolong the SG tube leakage, causing a continued increase in the water level and the increase in the steam pressure that may lift the steam relief valve of the ruptured SG.

The operators have to take the following recovery actions:

a) Isolate the ruptured SG
   Operators identify the ruptured SG and isolate the ruptured SG by closing the steam line isolation valve, main feedwater isolation valve and other valves.
b) Terminate the leak flow
   Operators reduce the RCS temperature of the intact loop using the steam relief valves of the intact SGs or turbine bypass system, depressurize the RCS using a pressurizer depressurization valve until primary-to-secondary pressure balance is attained, and terminate the ECCS flow. These actions in turn terminate the leak flow.

For the radiological analysis, the reactor coolant system water is assumed to contain some radioactive fission products corresponding to continuous operation with a limited number of defective fuel rods at the maximum allowance of the Technical Specifications. The accident leads to an increase in contamination of the secondary system due to leakage of radioactive coolant from the reactor coolant system. In the event of a coincident loss of offsite power, or a failure of the condenser steam dump system, discharge of radioactivity to the atmosphere takes place via the steam generator power-operated relief valves or the safety valves. This provides a pathway for the release of radioactivity to the environment.

The acceptance criteria set by MHI for this accident are to preclude additional fuel failures and to not allow the ruptured SG to fill with water. Filling the SG could result in radioactive water relief through the secondary relief valve, further increasing the radioactivity released to the environment. Additionally, the site boundary dose must meet the 10 CFR 100 requirements.

Reactor Protection

The following signals are assumed to be available to automatically trip the reactor and therefore provide protection from this transient.

- Pressurizer pressure low trip
- Over temperature ΔT high trip
- Steam generator water level high-high trip
- ECCS Actuation

Engineered Safeguards Features

The following features are assumed to be available to mitigate the accident.

- EFWS
- EFWS isolation
- Safety Injection

Method of Analysis

(1) Analysis code

The MARVEL-M code is used to calculate the reactor plant transient following a steam generator tube rupture until the primary-to-secondary break flow is terminated. The specific models for the analysis are discussed below:

(a) The SG tube break flow calculation

The initial break flow is conservatively determined assuming critical flow calculated using the primary pressure at the location of the break, accounting for the pressure drop between the tube inlet or outlet and the break location. From that point on, the break flow is calculated by MARVEL-M as a function of the square root of the primary-to-secondary differential pressure, scaled to match the initial flow.

A comparison has been made between the conservative model described above and a more detailed model that checks for critical flow conditions and uses either critical or non-critical flow depending on primary pressure and secondary pressure at the break location. This comparison is provided in Appendix F.

(b) Reactor coolant system response after the pressurizer is emptied

When the RCS pressure deceases significantly, the reactor coolant in the reactor vessel upper head dead volume may flash and form a steam phase at the top, separated from the liquid. It may act as an alternate pressurizer to define the reactor coolant system pressure after the pressurizer is emptied. If ECCS is actuated, the system also functions to maintain the RCS pressure and the ECCS flow and RCS leak flow are balanced. The transient model of the upper head dead volume is discussed in Section 2.1.3.1.

(2) Analysis Assumptions

Analysis assumptions and calculation conditions are as follows.

(a) Double-ended rupture

Conservative leak flow model used

(b) Limiting single failure is assumed
EFWS train failure is typically assumed in the fluid discharge for the dose evaluation.

(c) Conservative assumptions
A secondary relief valve is assumed to stick open after the valve is automatically opened for conservative analysis of radioactivity released for this accident. The steam release through the valve is terminated by the automatic closure of the block valve.
Conservatism in the analysis includes also primary-to-secondary leak rate model, time margins for operator actions, conservative addition of main feedwater and EFWS feedwater to the ruptured SG, and loss of off-site power at the time of reactor trip.

(d) Operator action and the time margin
Operator actions assumed to be taken to mitigate the accident and to recover the reactor safety conditions until the SG leak is terminated are as follows:

(i) Detection of the accident
SGTR causes various relevant indications, including the reduction in pressurizer water level, reduction in pressurizer pressure and the increase in water level in the ruptured SG. The event can be also detected from the SG blowdown radiation monitors, the steam condenser ejector radiation monitors, as well as the main steam line N-16 high-sensitivity radiation monitors installed on each steam line (the high radiation level alarms occur within 2 minutes from the SGTR initiation).

(ii) Identification of the ruptured SG and reactor trip
Operators can identify the ruptured SG from the N-16 radiation monitors and from the increase in the affected SG water level.
A time margin of 10 minutes is assumed for operators to identify the ruptured SG after the audible alarms indicate the event has occurred. Operators are assumed to trip the reactor manually 15 minutes after SGTR initiation.

(iii) Isolation of the ruptured SG
The ruptured SG is isolated by closing the main steam line isolation valve and other isolation valves. The actions to isolate the affected SG are assumed to be completed within 5 minutes after the reactor trip.

(iv) Reduce the RCS temperature
Operators are assumed to start to reduce the RCS temperature by opening the secondary relief valves of the intact SGs 5 minutes after the isolation of the ruptured SG.

(v) Depressurize the RCS and terminate the ECCS
After the RCS hot leg temperatures of the intact loops are reduced sufficiently to assure the subcooling even when the RCS pressure is reduced to the ruptured SG steam pressure, operators reduce the RCS pressure by opening a pressurizer depressurization valve until the primary-to-secondary pressure balance is attained. The ECCS is then terminated manually according to the SI termination criteria specified in the Emergency Operating Instructions.

(3) Calculation Case

Two cases are analyzed, one to determine the maximum integrated atmospheric steam relief and the second to confirm that none of the steam generators overfill.

6.0 SAMPLE TRANSIENT ANALYSIS

As explained in Section 5, six specific events were selected for inclusion in this Topical Report so as to demonstrate the application of each of the key computer codes (or groups of codes), as well as to include certain accidents where special methods or code capabilities are used. The methodology associated with the analysis of each of these six events is described in detail in Section 5. Section 6 provides sample transient analysis results for each of these six events. The results consist of a brief summary description of the event, a sequence of events table with explanation, and figures showing the time-dependent response of key parameters. The DCD analysis for these six events will consist of a combination of event description, descriptions of applicable computer code and models, accident classification, acceptance criteria, event-specific assumptions (initial conditions, time of cycle, core parameters, etc.), results, and conclusions included in Sections 2, 4, 5, and 6 of this report. For certain events, the results presented in Section 6 are of a representative or limiting case for the purpose of illustrating the format, content, and level of detail that are presented in the DCD.

Section 6.1 provides sample results for the Uncontrolled Rod Cluster Control Assembly (RCCA) Bank Withdrawal at Power. This analysis is performed using only the MARVEL-M code and demonstrates that the DNBR acceptance criterion is met. Section 6.2 provides sample results for the Complete Loss of Forced Reactor Coolant Flow. This analysis uses the MARVEL-M code (and associated internal reactor coolant pump model) to calculate the NSSS response, and uses the VIPRE-01M code for the fuel rod and DNBR calculations. Section 6.3 provides sample results for the Spectrum of RCCA Ejection Accidents. This rapid core reactivity transient uses the multidimensional TWINKLE-M transient code to calculate the core power distribution and the VIPRE-01M code for the fuel rod response used to evaluate fuel damage due to Pellet Clad Mechanical Interaction (PCMI) or other failure mechanisms. Results from the 3-D HZP case and the 1-D HFP case are presented. Section 6.4 provides sample results for the limiting Steam System Piping Failure (Main Steamline Break). The MARVEL-M code utilizes steamline break flow models and accident-specific inlet mixing and reactivity weighting factors to calculate a conservative NSSS response, and the VIPRE-01 code is used to evaluate the DNBR for an accident-specific power distribution assuming the most reactive rod stuck out of the core. Section 6.5 provides sample results for the limiting Feedwater System Pipe Failure (Main Feedline Break). The MARVEL-M code (and its associated secondary break model and steam generator heat transfer model) is used to calculate the NSSS response. Section 6.6 provides sample results for the Steam Generator Tube Rupture event. The MARVEL-M code is used to model the primary-to-secondary flow and manual actions to terminate the accident, as well as calculate parameters used in the radiological response analysis (not included in this Topical Report).

6.1 Uncontrolled RCCA Bank Withdrawal at Power

Event Description

This event is an uncontrolled control rod bank withdrawal at power initiated by either a failure of the rod control system or an operator error. The positive reactivity insertion results in a power transient, which increases the core heat flux creating a potential challenge to the DNB limits.

Events Analyzed

A range of cases utilizing different reactivity insertion rates at both BOC and EOC will be evaluated and presented in the DCD. The sample results presented in this section include a plot of minimum DNBR as a function of reactivity insertion rate initiated from Hot Full Power (HFP) assuming minimum reactivity feedback (BOC). In addition, plots of key parameters versus time are provided for the most limiting HFP DNBR cases at BOC conditions, which per Figure 6.1-1 occur at withdrawal rates of 2.5 pcm/sec and 75 pcm/sec.

Minimum DNBR is calculated by using the Revised Thermal Design Procedure (RTDP) and the WRB-2 DNB correlation [Reference 16].

The analysis uses conservative assumptions for moderator density coefficient, Doppler power coefficient, and trip simulation (trip setpoint, trip reactivity curve, rod drop time, RPS signal processing delays) as described in Section 5.1.

Analysis Results

The overall response of the primary and secondary systems and DNBR are evaluated by the MARVEL-M code. DNBRs are calculated internal to MARVEL-M using DNBR input data separately calculated by VIPRE-01M and an algorithm that adjusts DNBR for changes in RCS parameters. Although the MARVEL-M DNBR model has the capability to model flow variations, constant RCS flow is assumed for this event. The focus of this reactivity insertion event is the challenge to the DNB design limit resulting from the power increase.

Figure 6.1-1 shows the minimum DNBR versus reactivity insertion rate (pcm/sec) for bank withdrawals initiating from HFP conditions assuming minimum feedback core physics parameters and the availability of pressure control systems (pressurizer spray). Reactor trips credited for the accident from HFP conditions include the over temperature ΔT high and power range neutron flux high trips. For slower reactivity addition rates from HFP conditions, protection is provided by the over temperature ΔT high trip. For higher insertion rates, the power range neutron flux high trip provides protection. The minimum DNBR for the HFP cases represented by Figure 6.1-1 occurs at 2.5 pcm/sec reactivity insertion rate protected by the over temperature ΔT high trip. Transient parameter plots are provided as the sample results for the uncontrolled bank withdrawal at power parameters for 75 pcm/sec reactivity insertion rate scenario in Figures 6.1-2 through 6.1-7. The same parameters are provided for the 2.5 pcm/sec insertion rate scenario in Figures 6.1-8 through 6.1-13. The sequence of events for these specific cases (HFP, minimum feedback, pressure control available) are provided in Table 6.1-1 (75 pcm/sec) and Table 6.1-2 (2.5 pcm/sec).

For the limiting 2.5 pcm/sec reactivity insertion rate, the initiation of the bank withdrawal occurs at time = 0 seconds. Power and ΔT increase until the over temperature ΔT high trip is initiated at time = 51.9 seconds, and the minimum DNBR occurs immediately following the trip. The minimum DNBR is greater than the 95/95 DNBR Design Limit for transients using the RTDP, the peak RCS pressure remains below 2750 psia (110% of RCS design pressure), and the steam pressure remains below 1320 psia (110% of the main steam system design pressure).

Table 6.1-1 Sequence of Events for the Uncontrolled RCCA Bank Withdrawal at Power (75 pcm/sec)

Event	Time (sec)
RCCA Bank Withdrawal Begins	0.0
Neutron Flux High Analysis Limit Reached	1.4
Reactor Trip Initiated (Rod Motion Begins)	2.3
Minimum DNBR Occurs	3.4
Peak Hot Spot Heat Flux Occurs	3.5

Table 6.1-2 Sequence of Events for the Uncontrolled RCCA Bank Withdrawal at Power (2.5 pcm/sec)

Event	Time (sec)
RCCA Bank Withdrawal Begins	0.0
Over Temperature ΔT High Analysis Limit Reached	45.1
Reactor Trip Initiated (Rod Motion Begins)	51.9
Peak Hot Spot Heat Flux Occurs	52.2
Minimum DNBR Occurs	52.2

Conclusions

The analysis of this event demonstrates that the resulting power increase does not result in a DNB-related fuel failure. The separate RCS pressure transient is not included in this topical report, but will be described in the DCD. The DNB acceptance criteria for this AOO event as described in Section 4.5 are met.

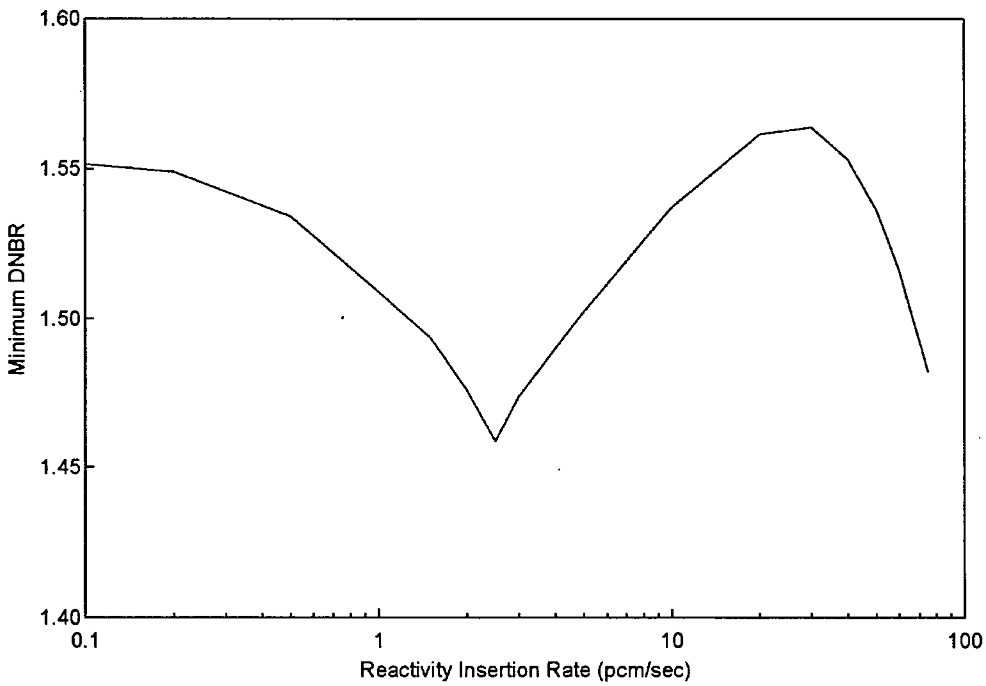


Figure 6.1-1 Minimum DNBR versus Reactivity Insertion Rate (HFP, BOC)

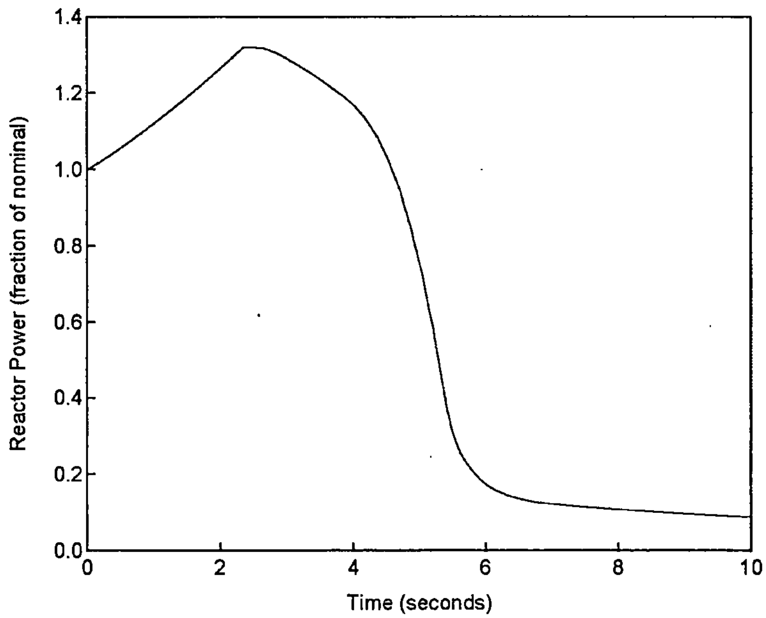


Figure 6.1-2 Reactor Power versus Time
Uncontrolled RCCA Bank Withdrawal at Power (HFP, BOC, 75 pcm/sec)

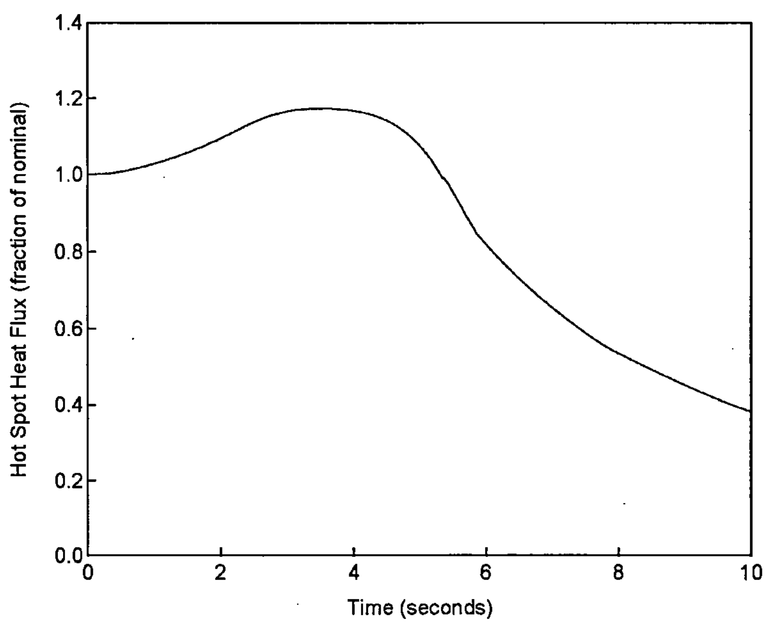


Figure 6.1-3 Hot Spot Heat Flux versus Time
Uncontrolled RCCA Bank Withdrawal at Power (HFP, BOC, 75 pcm/sec)

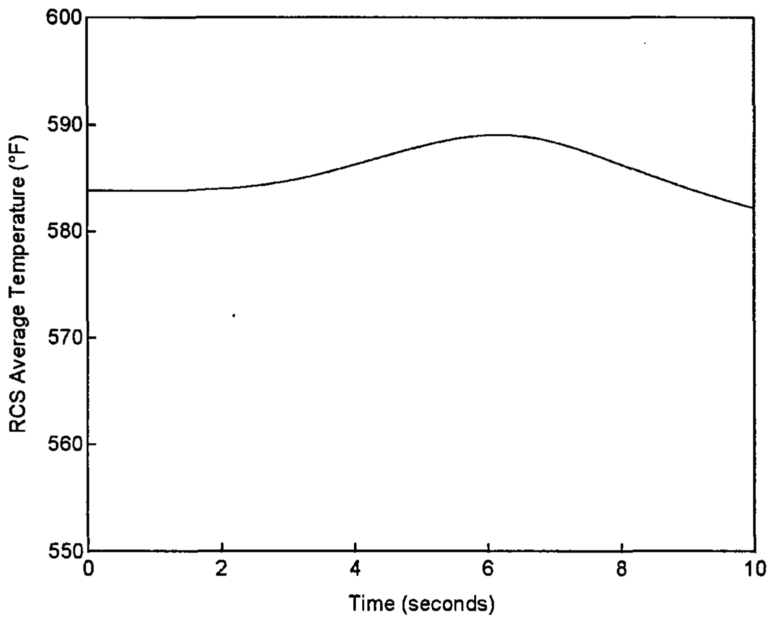


Figure 6.1-4 RCS Average Temperature versus Time
Uncontrolled RCCA Bank Withdrawal at Power (HFP, BOC, 75 pcm/sec)

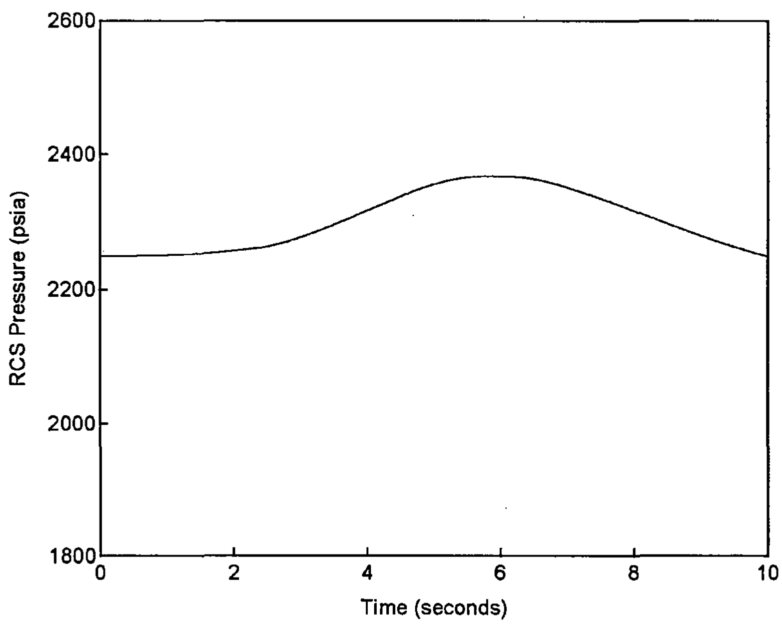


Figure 6.1-5 RCS Pressure versus Time
Uncontrolled RCCA Bank Withdrawal at Power (HFP, BOC, 75 pcm/sec)

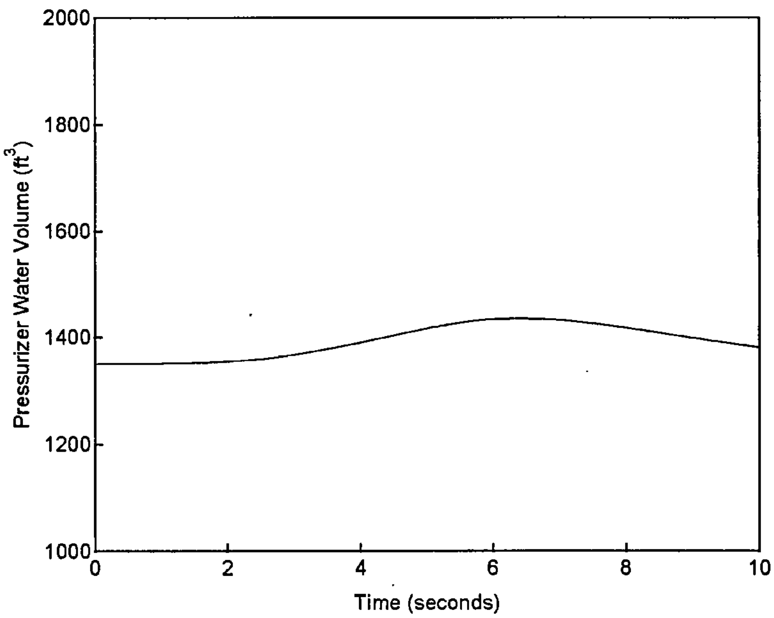


Figure 6.1-6 Pressurizer Water Volume versus Time
Uncontrolled RCCA Bank Withdrawal at Power (HFP, BOC, 75 pcm/sec)

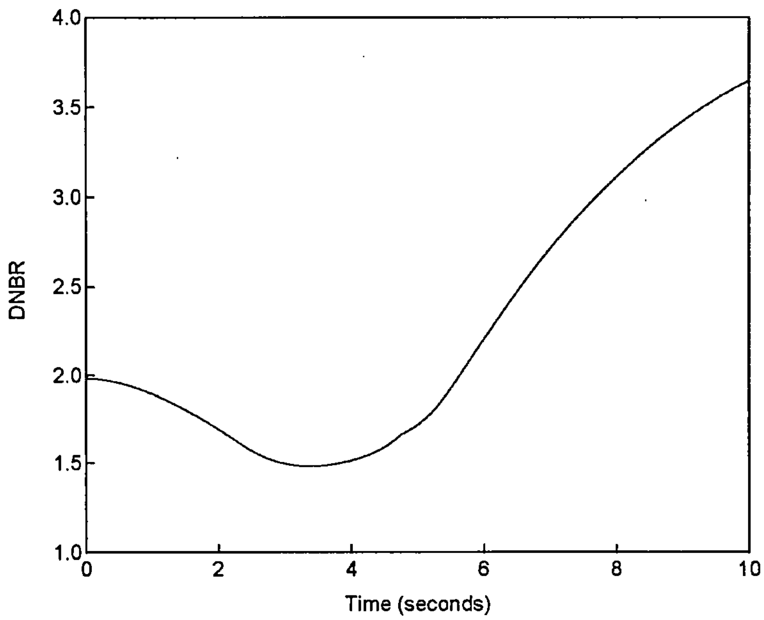


Figure 6.1-7 DNBR versus Time
Uncontrolled RCCA Bank Withdrawal at Power (HFP, BOC, 75 pcm/sec)

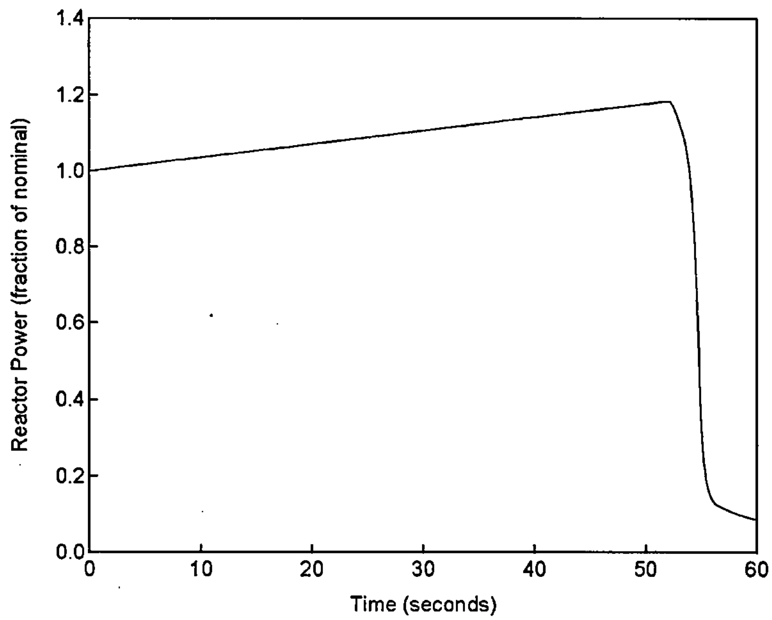


Figure 6.1-8 Reactor Power versus Time
Uncontrolled RCCA Bank Withdrawal at Power (HFP, BOC, 2.5 pcm/sec)

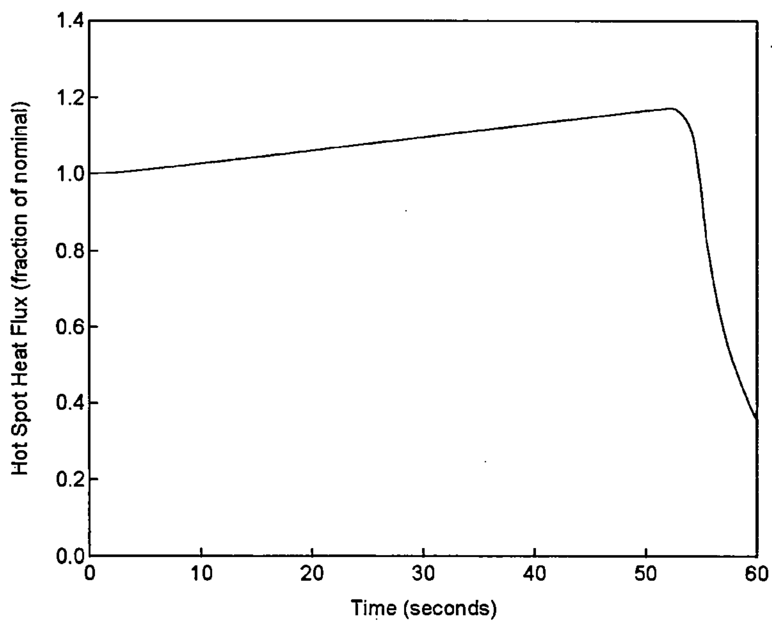


Figure 6.1-9 Hot Spot Heat Flux versus Time
Uncontrolled RCCA Bank Withdrawal at Power (HFP, BOC, 2.5 pcm/sec)

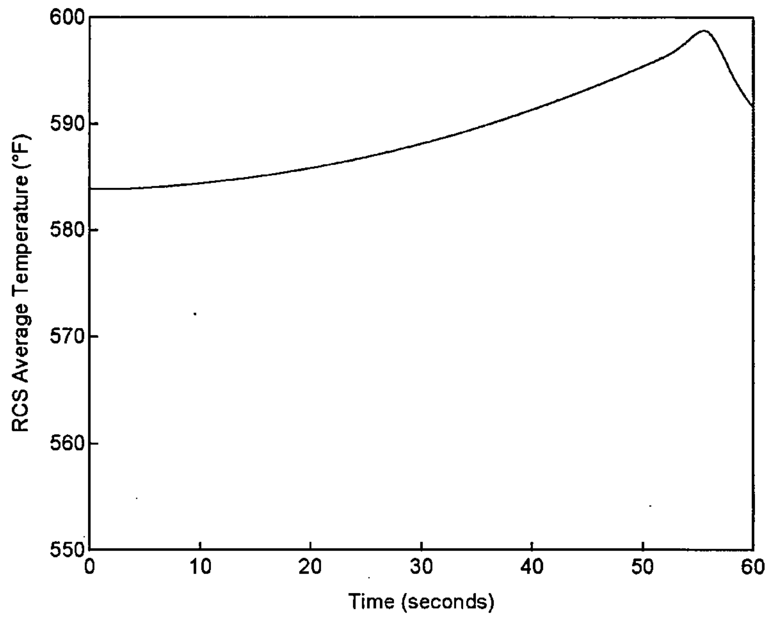


Figure 6.1-10 RCS Average Temperature versus Time
Uncontrolled RCCA Bank Withdrawal at Power (HFP, BOC, 2.5 pcm/sec)

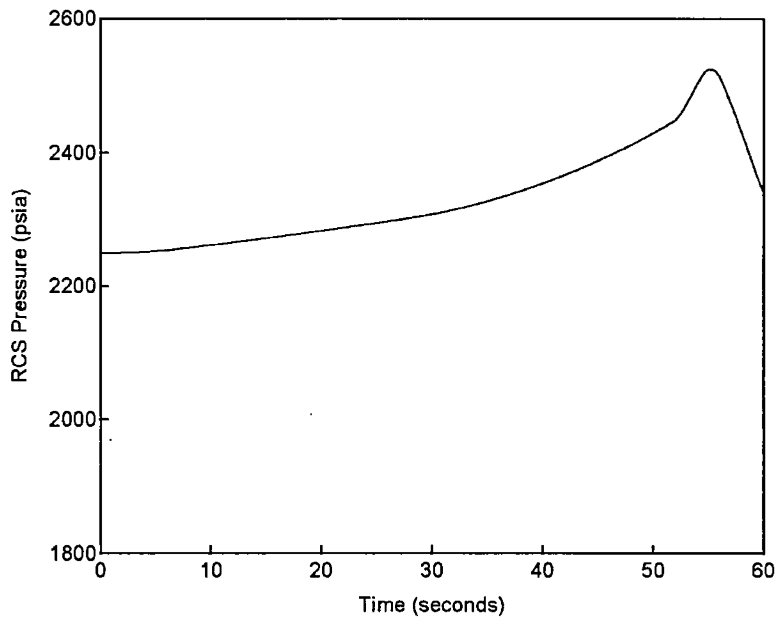


Figure 6.1-11 RCS Pressure versus Time
Uncontrolled RCCA Bank Withdrawal at Power (HFP, BOC, 2.5 pcm/sec)

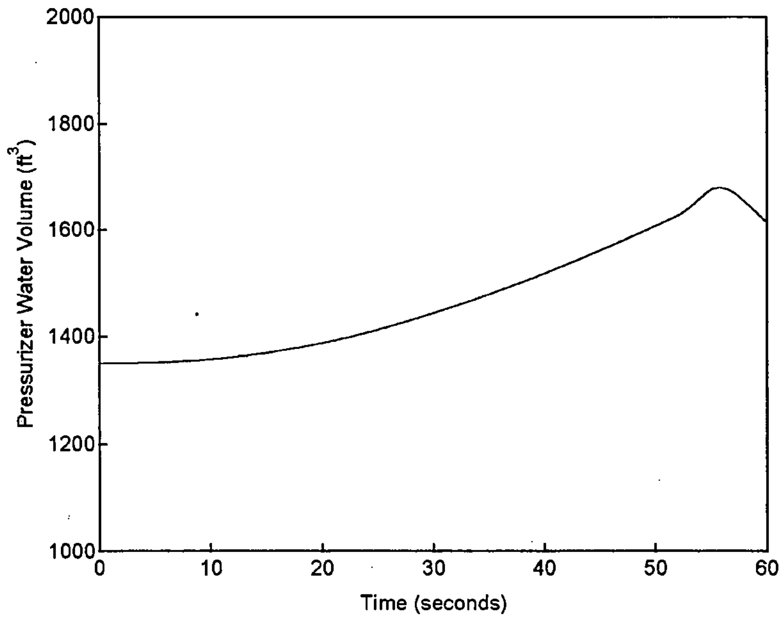


Figure 6.1-12 Pressurizer Water Volume versus Time
Uncontrolled RCCA Bank Withdrawal at Power (HFP, BOC, 2.5 pcm/sec)

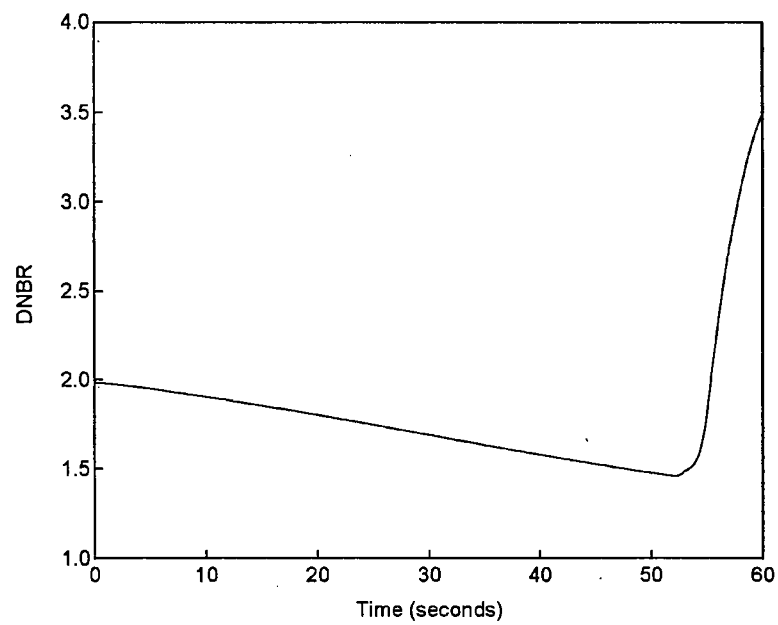


Figure 6.1-13 DNBR versus Time
Uncontrolled RCCA Bank Withdrawal at Power (HFP, BOC, 2.5 pcm/sec)

6.2 Complete Loss of Forced Reactor Coolant Flow

Event Description

Loss of forced reactor coolant flow events can result from a mechanical or electrical failure in one or more reactor coolant pumps (RCPs) or from a fault in the power supply to the pump motor. The complete loss of forced reactor coolant flow is initiated by malfunctions that cause the loss of electrical power to all four reactor coolant pumps during power operation, resulting in a reduction in the core cooling capabilities. Although the reduction in core cooling capability could also cause an increase in the reactor fuel temperature and in the reactor coolant temperature, the DNB limit is the primary design limit of concern due to the combination of core temperature increase and core flow decrease.

Events Analyzed

This section provides a sample transient analysis for the complete loss of forced reactor coolant flow event resulting from a loss of electrical power to all four reactor coolant pumps.

The overall response of the primary and secondary systems is evaluated using MARVEL-M. For the loss of flow transients, the MARVEL-M calculates the time-dependent core flow using the reactor coolant pump model described in Section 2.1.3. Time-dependent normalized values of core flow and core power calculated by MARVEL-M are transferred to the VIPRE-01M code for DNBR calculations using the Revised Thermal Design Procedure (RTDP) and the WRB-2 DNB correlation. Inlet temperature and RCS pressure are held constant at conservative values for the DNBR calculations.

The MARVEL-M analysis uses conservative assumptions for the RCP flywheel inertia, moderator density coefficient (least positive), Doppler power coefficient (most negative), and trip simulation (trip setpoint, trip reactivity curve, rod drop time, RPS signal processing delays), as described in Section 5.2. The initial conditions for power, RCS temperature, and RCS pressure are assumed at their nominal values, consistent with the RTDP methodology.

The RPS trips available to mitigate the complete loss of flow from full power include the low reactor coolant flow and low reactor coolant pump speed.

This event was chosen as one of the six sample analyses because it utilizes MARVEL-M to calculate the NSSS response using its internal reactor coolant pump model and performs the DNBR analysis external to MARVEL-M using the VIPRE-01M code.

Analysis Results

The complete loss of forced reactor coolant flow transient is initiated by a trip of all four RCPs. As the pumps coast down, a reactor trip signal is generated by the RCP speed low trip. Prior to the reactor trip, the power increases and the flow decreases, resulting in a decrease in DNBR. The minimum DNBR occurs shortly after the reactor trip following the sharp decrease in power. The minimum DNBR is greater than the 95/95 DNBR Design Limit for transients using the RTDP, the peak RCS pressure remains below 2750 psia (110% of RCS design pressure), and the steam pressure remains below 1320 psia (110% of the main steam system design pressure).

Table 6.2-1 provides the sequence of events for the complete loss of forced coolant flow event. The transient responses for key parameters are presented in Figures 6.2-1 through 6.2-6.

Table 6.2-1 Sequence of Events for the Loss of Forced Reactor Coolant Flow

Event	Time (sec)
RCPs Trip (Flow Coastdown Begins)	0.0
RCP Speed Low Analysis Limit Reached	0.8
Reactor Trip Initiated (Rod Motion Begins)	1.7
Minimum DNBR Occurs	4.0

Conclusions

The analysis of this event demonstrates that the transient does not result in a DNB-related fuel failure. Additionally, the peak pressures in the reactor coolant and main steam systems remain below 110% of the design pressure. Therefore, the AOO acceptance criteria for this event as discussed in Section 4.4 are met.

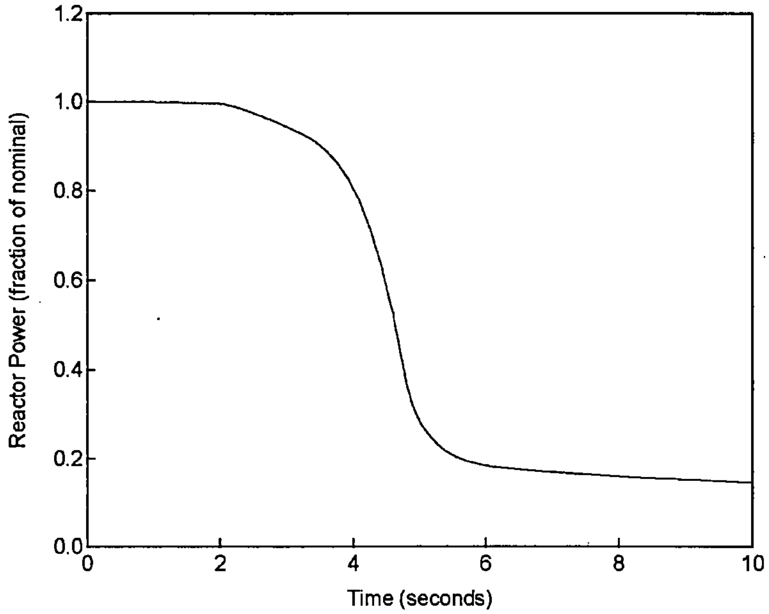


Figure 6.2-1 Reactor Power versus Time
Complete Loss of Forced Reactor Coolant Flow

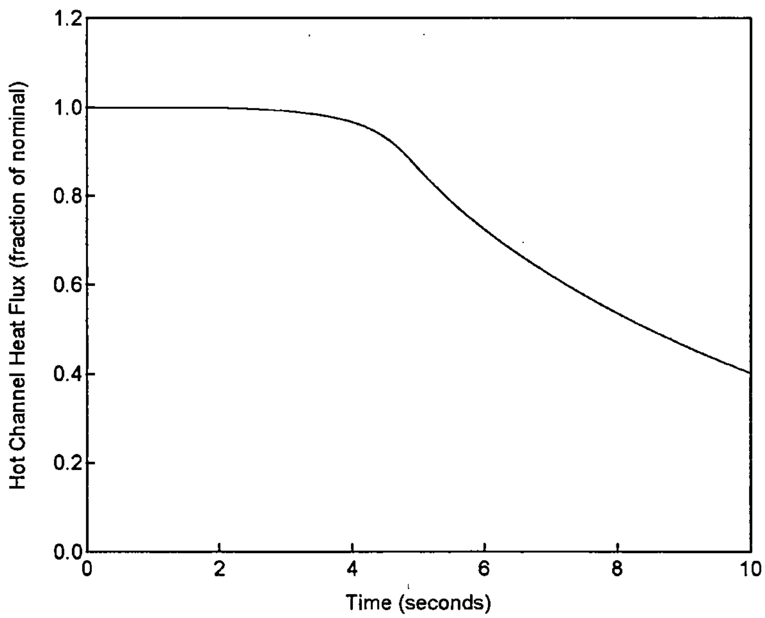


Figure 6.2-2 Hot Channel Heat Flux versus Time
Complete Loss of Forced Reactor Coolant Flow

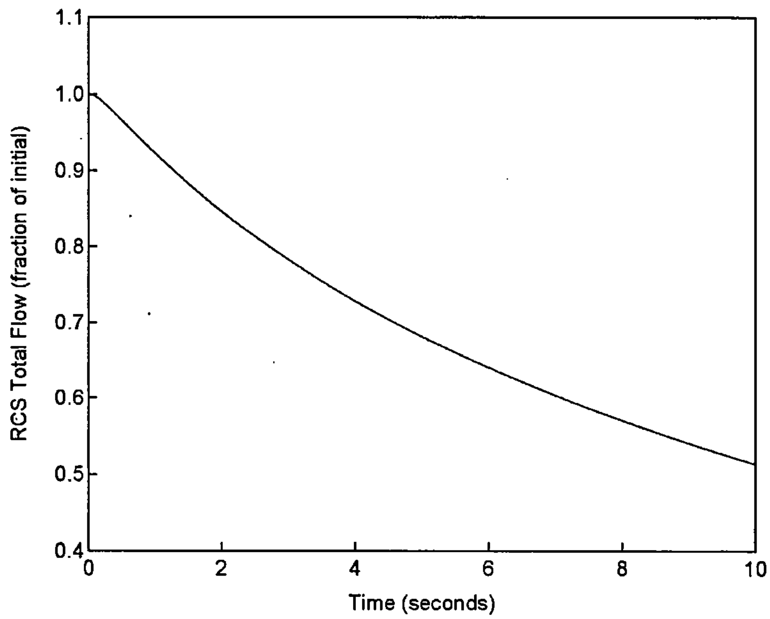


Figure 6.2-3 RCS Total Flow versus Time
Complete Loss of Forced Reactor Coolant Flow

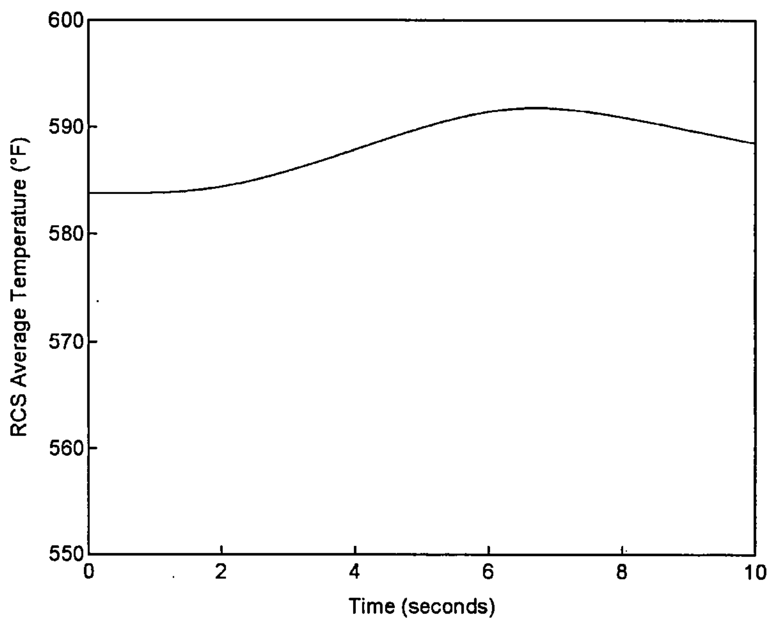


Figure 6.2-4 RCS Average Temperature versus Time
Complete Loss of Forced Reactor Coolant Flow

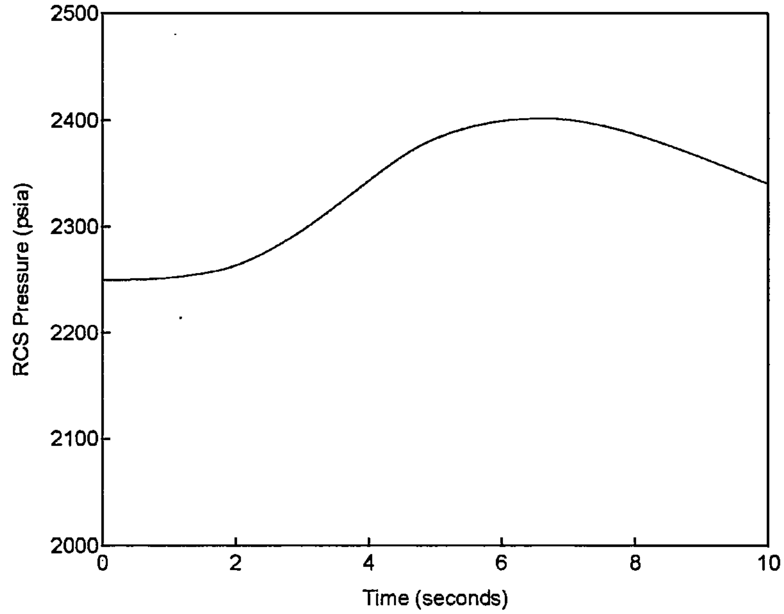


Figure 6.2-5 RCS Pressure versus Time
Complete Loss of Forced Reactor Coolant Flow

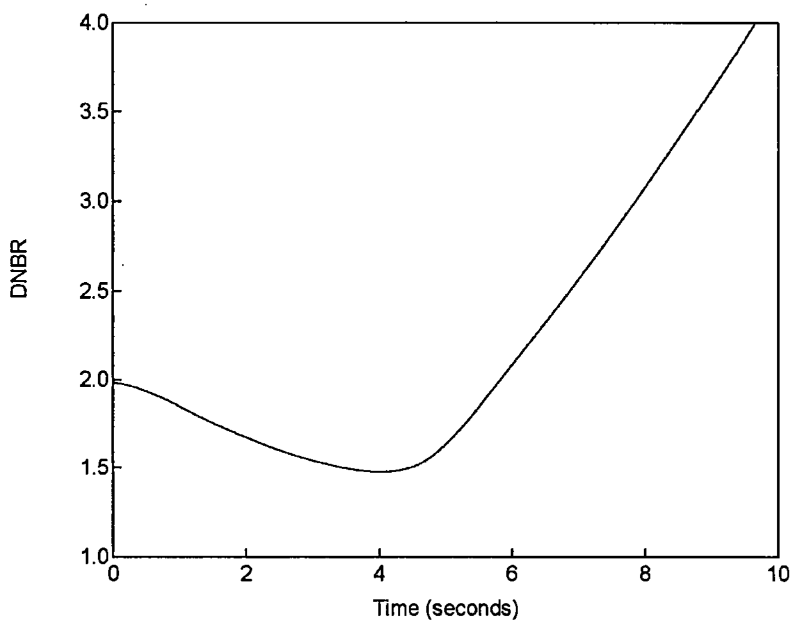


Figure 6.2-6 DNBR versus Time
Complete Loss of Forced Reactor Coolant Flow

6.3 Spectrum of RCCA Ejection

Event Description

This event is defined as a mechanical failure of a control rod drive mechanism pressure housing, resulting in the ejection of an RCCA and drive shaft. This rapid positive reactivity insertion results in a rapid increase in core power and local peaking, challenging the fuel design limits.

Events Analyzed

The limiting RCCA ejection cases, for both the beginning and end of cycle at both zero and full power, are evaluated in the DCD using the methodology described in Section 5.3 with respect to the acceptance criteria described in Section 4.5.

For the hot full power (HFP) cases, the TWINKLE-M spatial neutron kinetics code is used to determine the core average and local power generation with time. Then the VIPRE-01M code is utilized to determine the fuel response at the limiting location using local peaking factors based on design calculations (with safety margin) using the ANC code. For the hot zero power (HZP) cases, the transient is modeled using TWINKLE-M 3-D kinetics and a case-specific local peaking factor (with safety margin) is calculated for use in the VIPRE-01M fuel response analysis.

Sample results are provided for two cases, one HFP case (BOC) using design peaking factors and one HZP case (EOC) using transient-specific 3-D TWINKLE-M peaking factors.

Analysis Results

For the HFP (BOC) case, Control Bank-D is assumed to be inserted to its insertion limit when the rod ejection occurs. A bounding maximum ejected rod worth of 110 pcm and a design hot channel factor of 5.0 are assumed to provide margin for future cores. The reactivity insertion causes a rapid increase in power and the power increase is terminated by Doppler feedback. The reactor trip is initiated by neutron flux high (high setting) and the reactor returns subcritical following the trip. The hot spot peak fuel centerline temperature is 4347°F, which remains below the fuel melting temperature limit.

For the HZP (EOC) case, Control Bank-D is assumed to be fully inserted and the others inserted to their insertion limit when the rod ejection occurs. A bounding maximum ejected rod worth of 800 pcm and a hot channel factor of 35.0 are assumed to provide margin for future cores. The reactivity insertion causes a rapid increase in power and the power excursion is terminated by Doppler feedback. The reactor trip is initiated by neutron flux high (low setting) and the reactor returns subcritical following the trip. The hot spot peak fuel enthalpy is 77.8 cal/g. The number of PCMI failed fuel is zero.

The nuclear power, fuel temperature (centerline and average), and clad temperature transients for the HFP case are presented in Figures 6.3-1 and 6.3-2, and the nuclear power and fuel enthalpy transients for the HZP case are presented in Figures 6.3-3 and 6.3-4, respectively. The relationship between oxide / wall thickness and fuel enthalpy rise is presented in Figure 6.3-5 for the HZP case. The calculated pairs of points are plotted on the same graph as the acceptance criterion. The calculated sequence of events corresponding to these limiting events is provided in Table 6.3-1.

Table 6.3-1 Sequence of Events for the RCCA Ejection

Event	BOC HFP Time (sec)	EOC HZP Time (sec)
Rod Ejection Occurs	0.0	0.0
Neutron Flux High Analysis Limit Reached	0.07 (high setting)	0.15 (low setting)
Peak Nuclear Power Occurs	0.11	0.16
Reactor Trip Initiated (Rod Motion Begins)	0.97	1.05
Maximum Fuel Temperature Occurs	2.8	-
Maximum Fuel Enthalpy Occurs	-	1.60

Conclusions

The analysis of this event demonstrates that the interim core coolability criteria described in SRP 4.2 Appendix B have been met, and therefore, no consequential damage occurs to the RCS or containment as a result of fuel damage. The separate RCS pressure transient using the MARVEL-M code with void data from VIPRE-01M is not included in this topical report. The fuel limit acceptance criteria for this event as described in Section 4.5 are met.

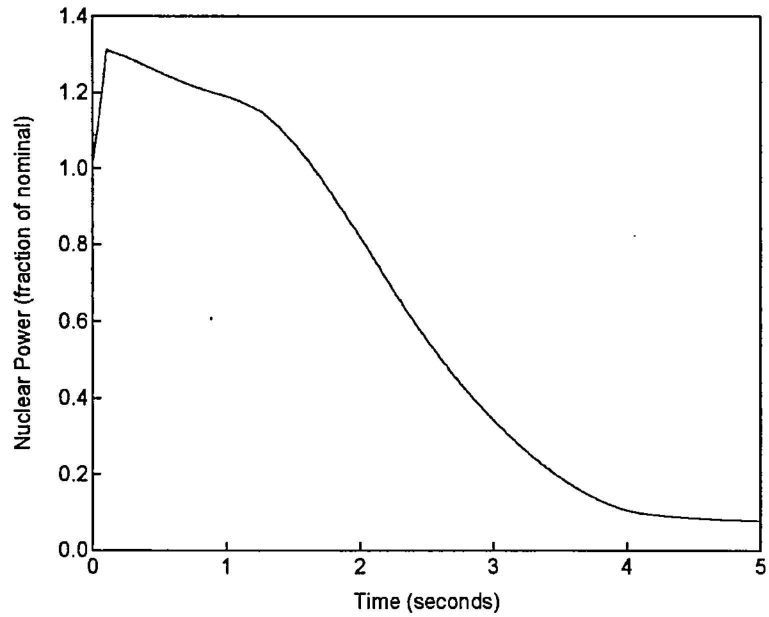


Figure 6.3-1 Nuclear Power versus Time RCCA Ejection (BOC HFP)

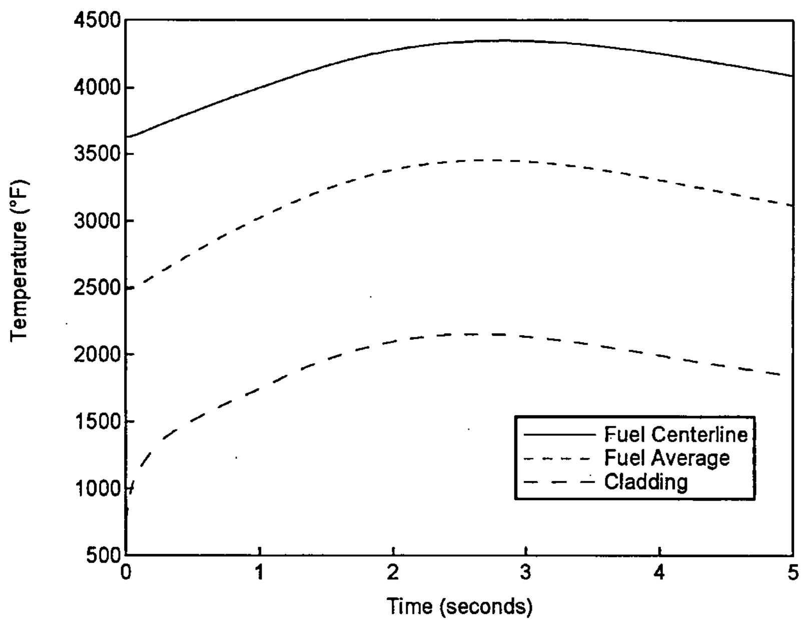


Figure 6.3-2 Fuel and Cladding Temperature versus Time RCCA Ejection (BOC HFP)

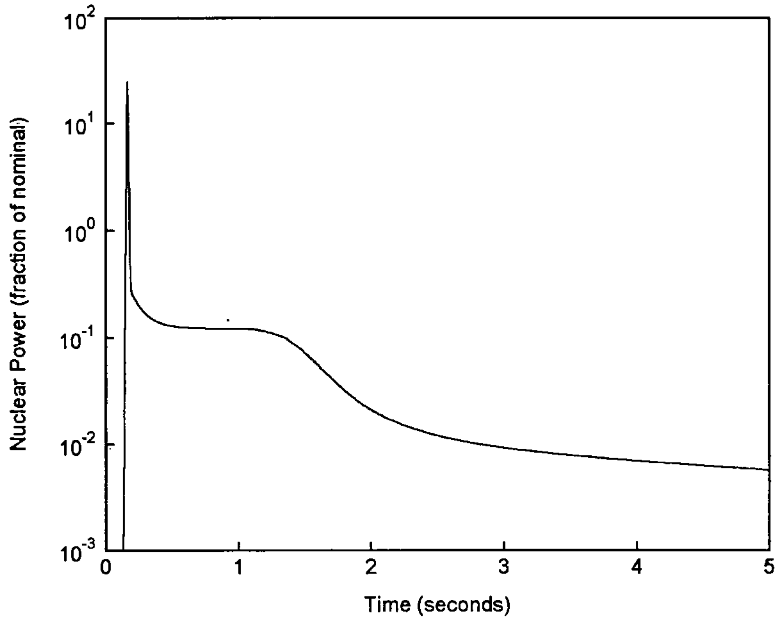


Figure 6.3-3 Nuclear Power versus Time RCCA Ejection (EOC HZP)

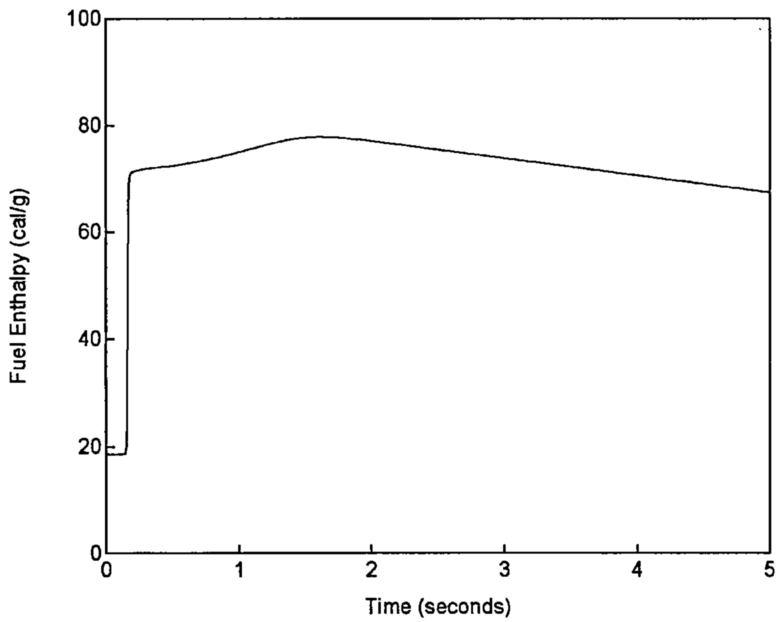


Figure 6.3-4 Fuel Enthalpy versus Time RCCA Ejection (EOC HZP)

Figure 6.3-5 Fuel Enthalpy Rise versus Oxide / Wall Thickness RCCA Ejection (EOC HZP)

6.4 Steam System Piping Failure

Event Description

The rupture of a main steam line results in removal of energy from the reactor coolant system through the steam generator leading to a reduction in coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the event results in a positive reactivity insertion and an increase in power level (at-power) or a return to criticality and power (hot shutdown).

Events Analyzed

The Steam System Piping Failure (Main Steamline Break) accident can be characterized as a spectrum of break sizes and locations that can occur from various operating modes (e.g., hot shutdown, at power) at various times in core life (BOL, EOL), with and without offsite power (forced reactor coolant pump flow). As discussed in Section 5.4, the event is analyzed assuming the most reactive control rod stuck out of the core, which results in both a reactivity penalty as well as a power distribution penalty. The sample results provided in this section are for an instantaneous double-ended guillotine break in a steam pipe, between the steam generator and turbine. Failure of a main steam system pipe (described in this section) is a postulated accident event as defined in Section 4.2. Effects of other minor secondary steam system pipe breaks (classified as an AOO in Section 4.2) are bounded by the analysis of the large double-ended break.

The overall response of the primary and secondary systems is evaluated using the MARVEL-M code. As discussed in Section 2, the MARVEL-M code calculates break flow and its resulting effect on reactivity, power, and RCS parameters. The VIPRE-01M code is then used to determine if DNB occurs for selected steady state core conditions computed by the MARVEL-M code during the transient.

The analysis is performed at the end of the core cycle. The moderator density coefficient has its highest value (moderator temperature coefficient has its highest negative value) at the end of the cycle, causing the cooldown to have the maximum impact on the core transient. As described in Section 5.4, the stuck rod assumption (and its associated reactivity and power distribution effects) has no meaning prior to the minimum DNBR portion of the at-power breaks. Therefore, the at-power analysis to verify that the RPS protects the core limits is done in the same manner as the Bank Withdrawal at Power analysis using only the MARVEL-M code (and its internal DNBR calculation) as described in Sections 5.1 and 6.1. Sample results for these at-power cases are not provided in this section. The large, double-ended break from hot shutdown inside the MSIV with full reactor coolant flow is a representative case, and is the only case presented in this sample transient analysis section.

Analysis Results

Figures 6.4-1 through 6.4-11 are plots of system parameters versus time from the core response analysis for the double-ended steam line failure from hot shutdown with offsite power available. The break is assumed to occur inside the main steam isolation valve on one of the steam lines, resulting in the complete blowdown of one steam generator. If the core is at critical hot zero power conditions when the break occurs, the main steam line pressure low

ECCS actuation signal will trip the reactor, leading to a transient much like the case presented here.

Immediately following the break, a main steam line pressure low (any one steam generator) signal will occur on the affected loop, resulting in steamline isolation and reactor trip signals. The steamline pressures on the other loops will not be affected because check valves in each steam line inside the reactor building upstream of the main steam header prevent steam flow to the break. However, the effect of check valves is conservatively ignored in the analysis prior to steamline isolation. If the break were to occur outside the reactor building (downstream of the main steamline isolation valves), flow to the break would be terminated in all steam lines by the closure of the main steam isolation valves (MSIVs) actuated by the steamline isolation signal. The main steam line pressure low signal also causes an ECCS actuation signal, which in turn, starts the emergency feedwater pumps, isolates main feedwater, and starts the safety injection pumps.

As shown in Figure 6.4-1, the reactor becomes critical with the control rods inserted (assuming the single most reactive rod in the fully withdrawn position), and the reactor returns to power. The cooldown continues at a decreasing rate due to decreasing steam pressure, until the affected steam generator inventory is depleted and emergency feedwater to it is isolated. As can be seen from Figures 6.4-6 and 6.4-8, the intact loop with the pressurizer surge line connection responds differently from the other two intact loops due to the addition of warmer pressurizer outsurge to the hot leg. The steam generator pressure in the pressurizer loop remains higher than the other intact loops after steam line isolation due to this effect and the assumption of no reverse heat transfer from the steam generators to the RCS. When the RCS pressure decreases below the shutoff pressure of the safety injection pumps, borated water begins to flow to the RCS, as indicated by the boron concentration transient shown in Figure 6.4-11. A single failure of one safety injection train is assumed in the analysis. The limiting point in the transient occurs when the nuclear power and core heat flux peak, resulting from the combination of decreasing steam flow and increasing core boron concentration.

Only one steam generator blows down completely following a steam pipe failure transient. Emergency feedwater to the affected steam generator is isolated automatically on an uncompensated steam generator pressure signal as shown in Figure 6.4-10. As shown in Figure 6.4-7, the blowdown is terminated when the affected steam generator mass is depleted, terminating the rapid cooldown. After the faulted steam generator mass is depleted, the pressurizer level recovers and the differences in loop inlet temperature decrease due to mixing in the reactor vessel as shown in Figures 6.4-5 and 6.4-6. The other three steam generators remain available for removal of decay heat from the primary coolant after the initial transient is over.

A steady state analysis is performed at the peak power point in the transient to calculate the minimum DNBR. The ANC code is used to calculate the limiting power distribution assuming the most reactive rod fully withdrawn; the limiting location is in the core quadrant associated with the broken loop. The ANC analysis also confirms the conservatism of the reactivity and nuclear power transients as calculated by the MARVEL-M code. The ANC power distribution and core inlet temperature distribution is used to perform a hot channel DNBR analysis using VIPRE-01M. Because the RCS pressures are below the applicable pressure range for the WRB-2 DNBR correlation, the W-3 correlation [Reference 17] and its associated 95/95 limit are used.

Table 6.4-1 Sequence of Events for the Steam System Piping Failure

Event	Time (sec)
Steam Pipe Rupture (Steamline Break) Occurs	0.0
Main Steam Line Pressure Low Analysis Limit Reached	1.5
MSIVs Closed	10.0
Safety Injection Pumps Start	21.5
Boron Reaches Core	44.7
Automatic Isolation of EFWS to Faulted SG	60.0
Peak Core Heat Flux Occurs	88.2
Faulted SG Water Mass Depleted	333

Conclusions

Although the Steam System Piping Failure is a Postulated Accident (fuel failures are permitted according to the acceptance criteria discussed in Section 4.1.2), the minimum DNBR does not exceed the 95/95 DNBR limit, and therefore no fuel failures are predicted as a result of the accident.

Therefore, the acceptance criteria for core damage as defined in Section 4.2 for this event are met.

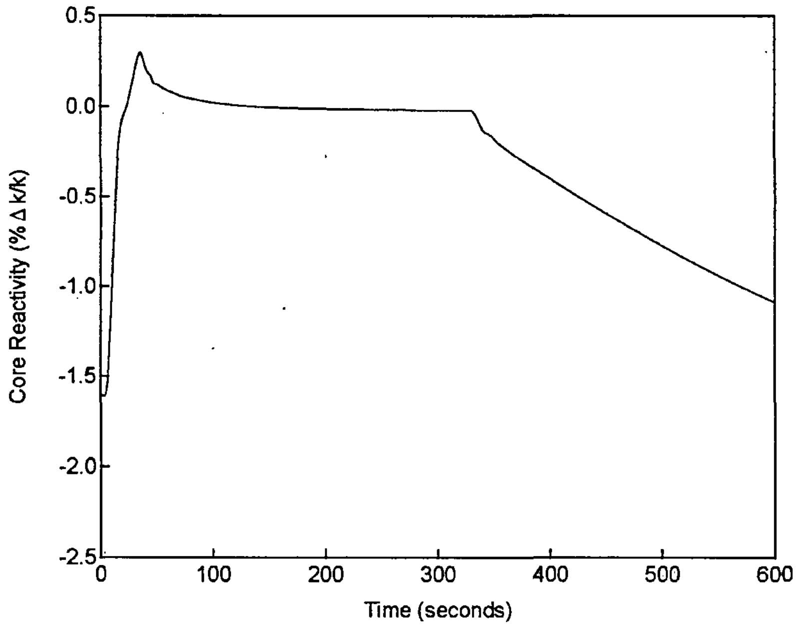


Figure 6.4-1 Core Reactivity versus Time
Steam System Piping Failure – Double-Ended Break from Hot Shutdown

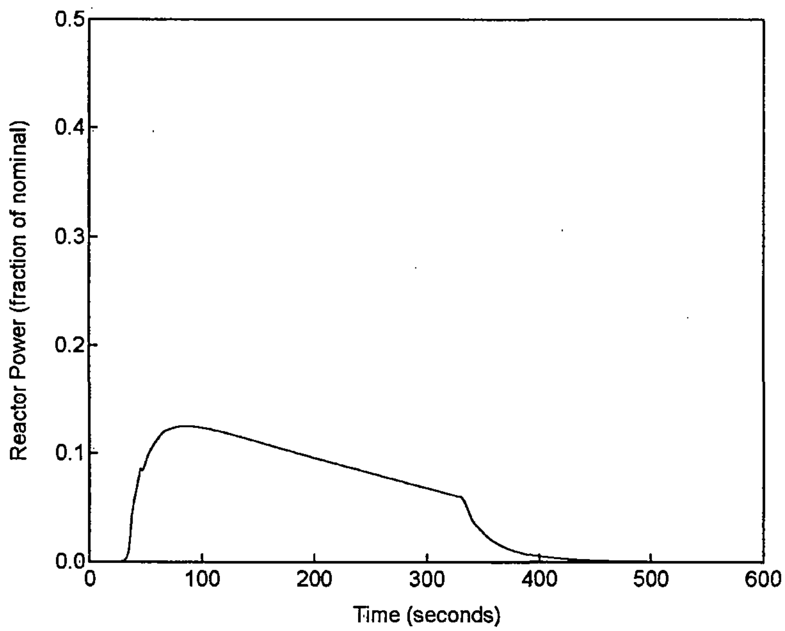


Figure 6.4-2 Reactor Power versus Time
Steam System Piping Failure – Double-Ended Break from Hot Shutdown

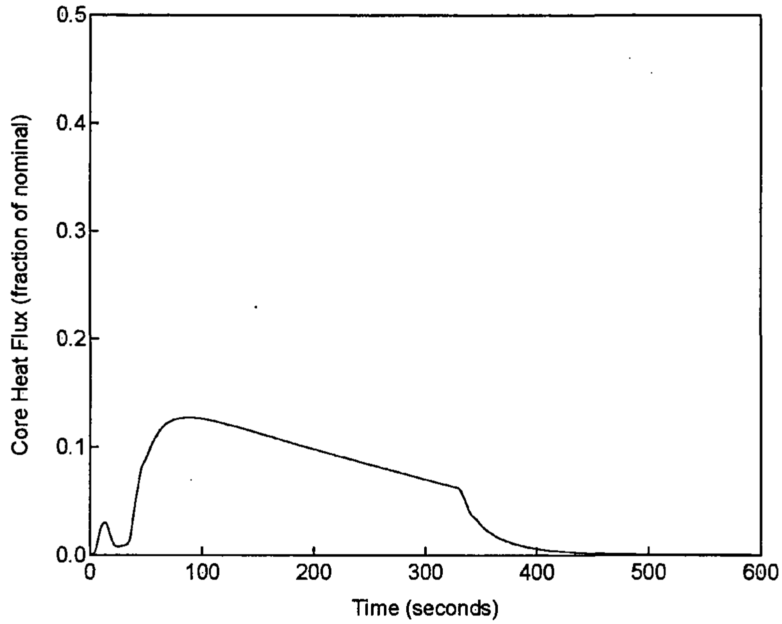


Figure 6.4-3 Core Heat Flux versus Time
Steam System Piping Failure – Double-Ended Break from Hot Shutdown

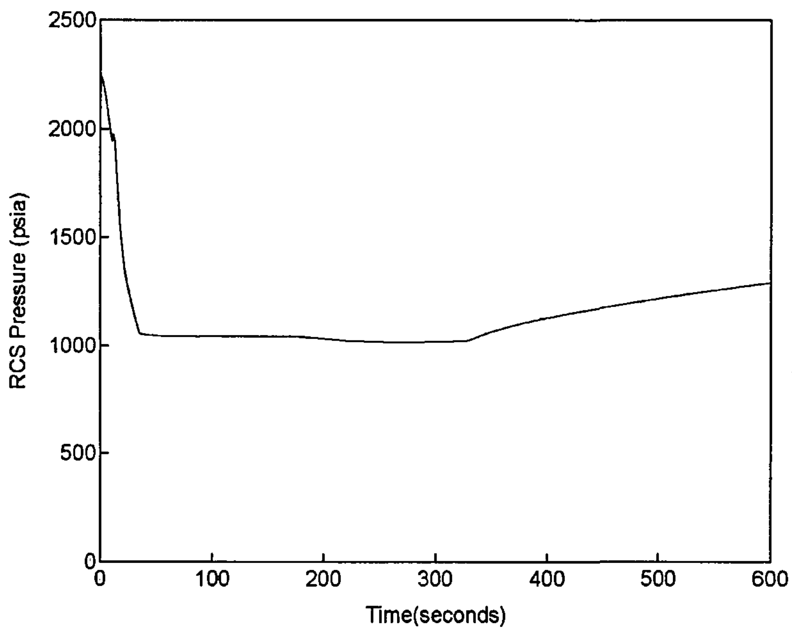


Figure 6.4-4 RCS Pressure versus Time
Steam System Piping Failure – Double-Ended Break from Hot Shutdown

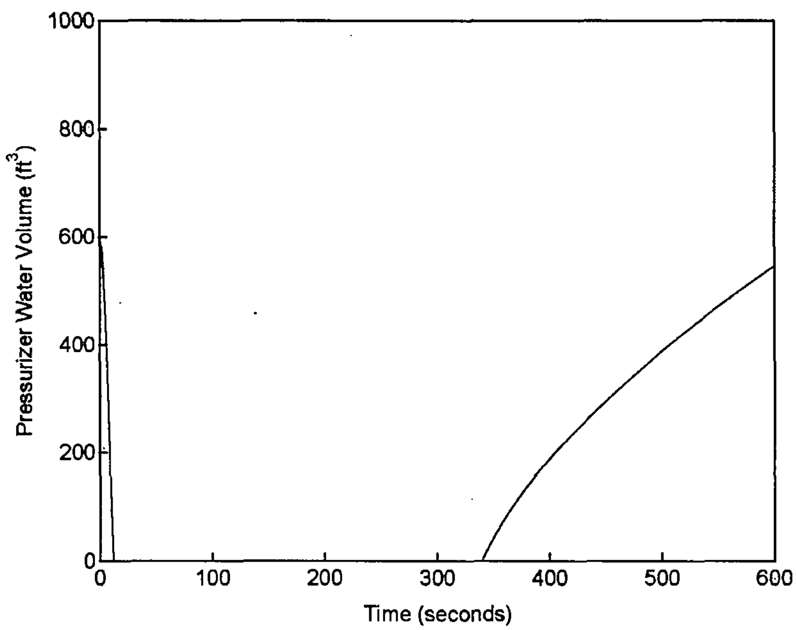


Figure 6.4-5 Pressurizer Water Volume versus Time
Steam System Piping Failure – Double-Ended Break from Hot Shutdown

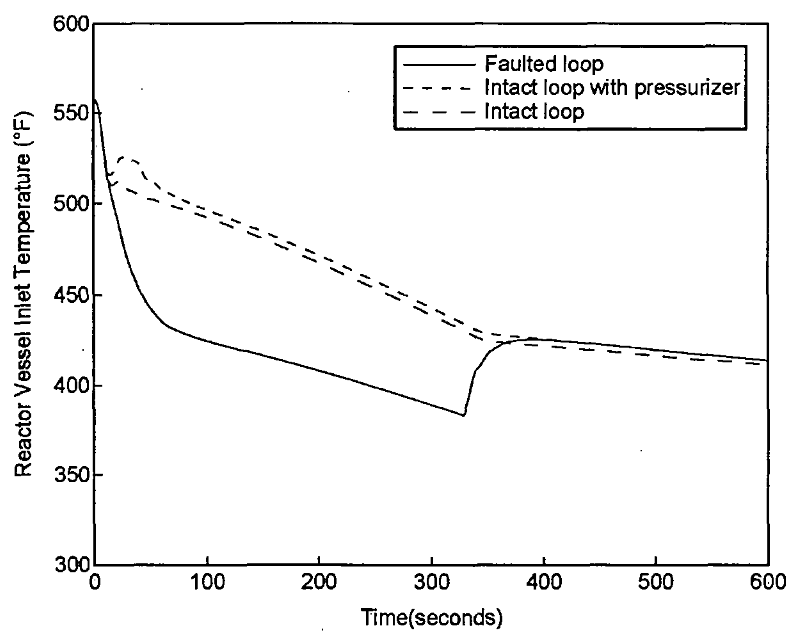


Figure 6.4-6 Reactor Vessel Inlet Temperature versus Time
Steam System Piping Failure – Double-Ended Break from Hot Shutdown

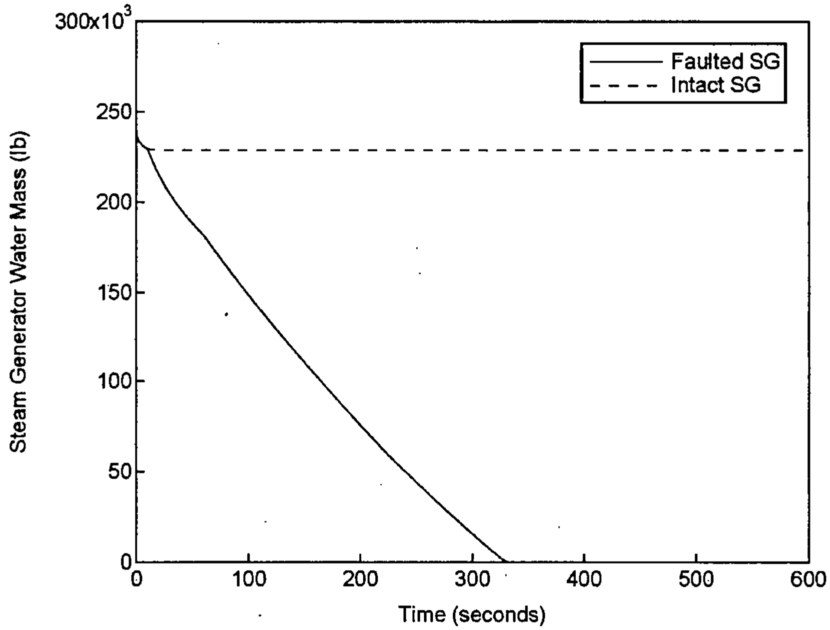


Figure 6.4-7 Steam Generator Water Mass versus Time
Steam System Piping Failure – Double-Ended Break from Hot Shutdown

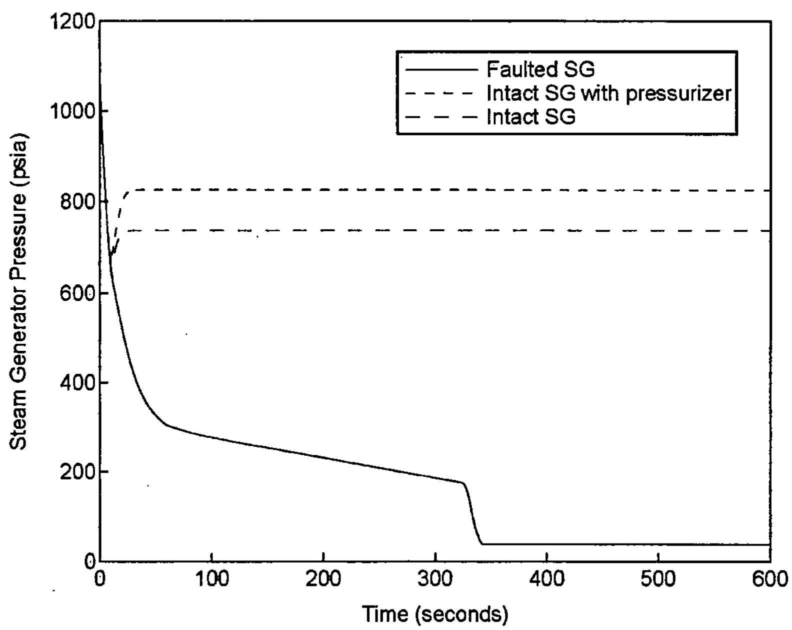


Figure 6.4-8 Steam Generator Pressure versus Time
Steam System Piping Failure – Double-Ended Break from Hot Shutdown

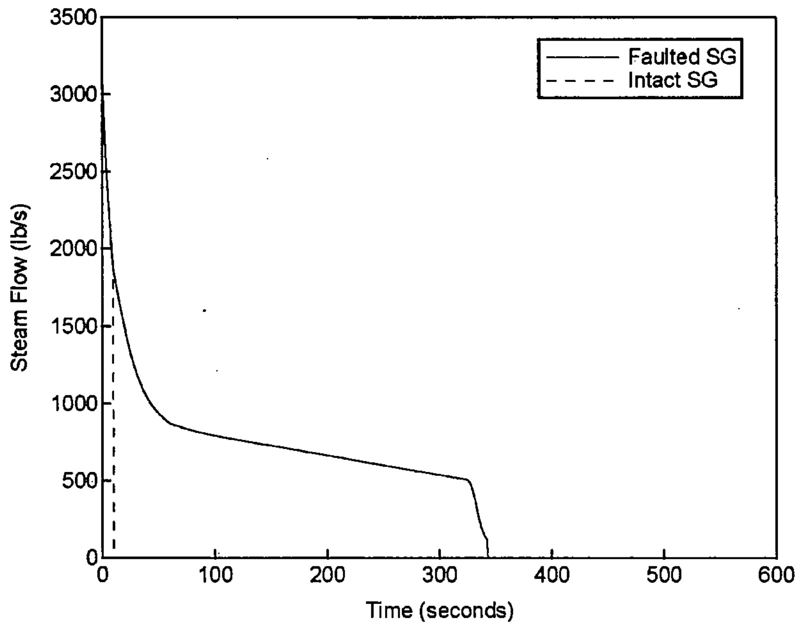


Figure 6.4-9 Steam Flow Rate versus Time
Steam System Piping Failure – Double-Ended Break from Hot Shutdown

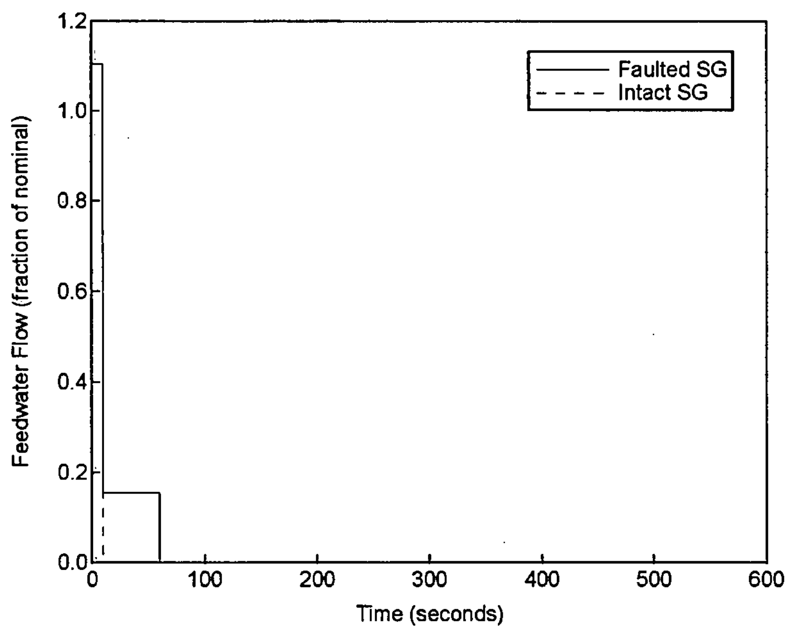


Figure 6.4-10 Feedwater Flow Rate versus Time
Steam System Piping Failure – Double-Ended Break from Hot Shutdown

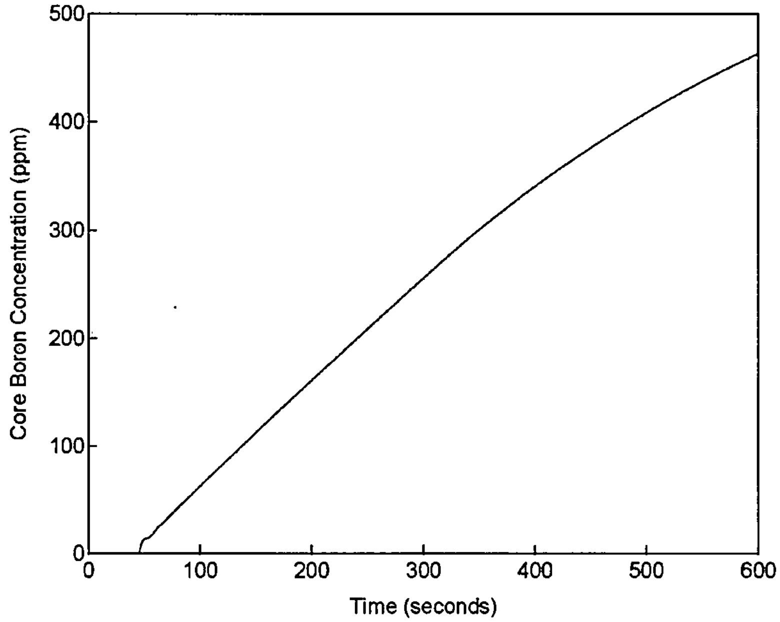


Figure 6.4-11 Core Boron Concentration versus Time
Steam System Piping Failure – Double-Ended Break from Hot Shutdown

6.5 Feedwater System Pipe Break

Event Description

SRP 15.2.8 defines a major feedwater line rupture as a feedwater line break large enough to prevent the addition of sufficient feedwater to the steam generators to maintain shell side fluid inventory in the steam generators. The large, double-ended rupture of one feedwater line is classified as a Postulated Accident that results in a limiting heatup and pressurization of the reactor coolant system and the non-affected portion of the secondary system.

Events Analyzed

This analysis evaluates the effects of the most limiting feedwater line break - a double-ended rupture of one feedwater line between the feedwater line check valve and the steam generator. This results in the rapid blowdown of one steam generator through the break. The emergency feedwater system (EFWS) train that would normally supply the broken loop will spill out of the feedwater line and not contribute to removing heat from the primary system. In addition, a single failure of one of the other four EFWS trains is assumed, resulting in degraded heat removal from one of the remaining intact steam generators. The methodology for this accident is described in detail in Section 5.5.

For conservatism, the feedwater system break is assumed to occur when the plant is in a condition where all of the steam generators are at the steam generator water level low trip setpoint. This conservative precondition minimizes the total steam generator inventory available to remove heat from the RCS and makes the reactor protection system response independent of the steam generator pressure and level dynamics of the feedwater line break prior to the reactor trip. As a convenience to the analyst, this is modeled by assuming a loss of normal feedwater at time = 0 with the feedwater pipe break occurring at the time of the steam generator water level low trip. The MARVEL-M code is used to analyze the overall response of the primary and secondary systems. The MARVEL-M code calculates the break flow using the Moody correlation assuming saturated liquid (quality = 0) flow. This assumption maximizes the rate at which the affected steam generator inventory is depleted and primary-to-secondary heat transfer is decreased, resulting in a conservative RCS heatup. The MARVEL-M code also models reactor thermal kinetics (decay heat), reactor coolant system response including temperatures, pressure, pressurizer level, and flow (natural circulation), as well as the non-uniform primary-to-secondary heat transfer caused by the break and EFWS single failure and heat removal from the steam generator safety relief valves in the intact steam generators.

Analysis Results

Table 6.5-1 provides the sequence of events for the analyzed case. The transient responses for key parameters following a main feedwater line break are presented in Figures 6.5-1 through 6.5-11.

The break is assumed to occur immediately following the reactor trip on low steam generator level resulting from the loss of feedwater flow assumed as a precondition. On the sequence of events and figures, this is at time = 47 seconds. A loss of offsite power is assumed to occur at that time concurrent with the turbine trip. The steam generator mass is depleted very rapidly as shown by the break flow and affected steam generator mass in Figures 6.5-7 and 6.5-8. Primary-to-secondary heat transfer area is reduced when the level is below the top of the tubes in the affected steam generator, resulting in a heatup of the faulted loop as shown in Figure 6.5-3. EFWS is also started on a steam generator water level low signal. EFW flow does not enter the faulted steam generator. EFW is automatically isolated to the affected steam generator by isolation logic developed from main steam line pressure low signals. As a single failure, one of the remaining intact steam generators is assumed to not receive EFW flow. The differences between the response of this steam generator and the others receiving EFW flow can be seen in the RCS loop average temperature and steam generator mass figures. As the transient progresses, the three intact steam generators heat up and pressurize to the steam generator safety valve pressure. The intact steam generator without EFW flow gradually boils off its inventory through the safety valves, and its heat transfer also begins to decrease when level reaches the top of the U-tubes. The two remaining intact steam generators with EFW flow boil off the EFW flow through the safety valves, providing in RCS cooling. The transient "turns around" at the point where decay heat balances the steam generator heat removal capability. This occurs at time = 1585 seconds, at which time the pressurizer water volume peaks and begins to decrease. The pressurizer level peaks before the pressurizer fills. In addition, boiling in the hot leg does not occur in the intact loops receiving EFW flow; subcooling margin is maintained.

It should be noted that fuel rod failure resulting from DNB is of primary concern when the reactor is operating at power, not during a heatup following a reactor trip. As a result of the way the transient is initiated, DNBR is not a parameter calculated during this transient. Subcooling is evaluated to preclude steam binding in the steam generator U-tubes for the steam generators receiving EFW flow during natural circulation flow conditions and to preclude the need to model reflux boiling heat transfer during the transient.

Table 6.5-1 Sequence of Events for the Feedwater System Pipe Failure

Event	Time (sec)
Loss of Feedwater Flow Occurs	0.0
Pressurizer Safety Valves Open	39
SG Water Level Low Analysis Limit Reached	45
Reactor Trip Occurs (Rod Motion Begins)	47
Feedwater Break Initiated	47
Reactor Coolant Pumps Tripped	47
Peak RCS Pressure Occurs	50
SG Safety Valves Open	52
EFWS Isolated to Broken Loop	83
EFWS Pumps Start	187
Peak Pressurizer Water Volume Occurs	1585

Conclusions

The analysis of this event demonstrates that the transient does not result in overpressurization of the RCS, i.e., the peak RCS pressure remains below 120% of RCS design pressure. The peak pressure in the main steam system remains below 120% of the main steam system design pressure. The internal MHI accident-specific acceptance criterion that the pressurizer does not fill has also been met, providing added assurance that the pressurizer safety valves do not exceed their design basis, precluding the occurrence of another accident (Loss of

Coolant). In conclusion, the acceptance criteria for this event described in Section 4.3 have been met.

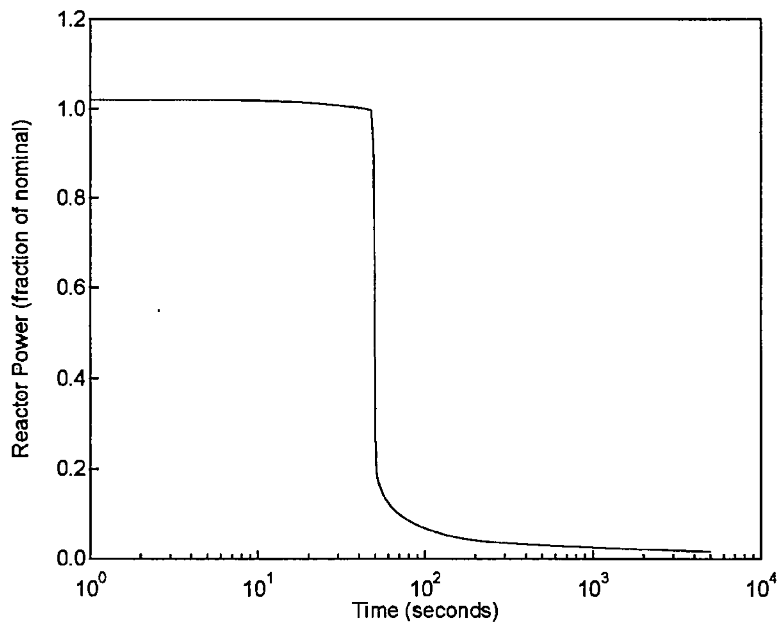


Figure 6.5-1 Reactor Power versus Time
Feedwater System Pipe Break

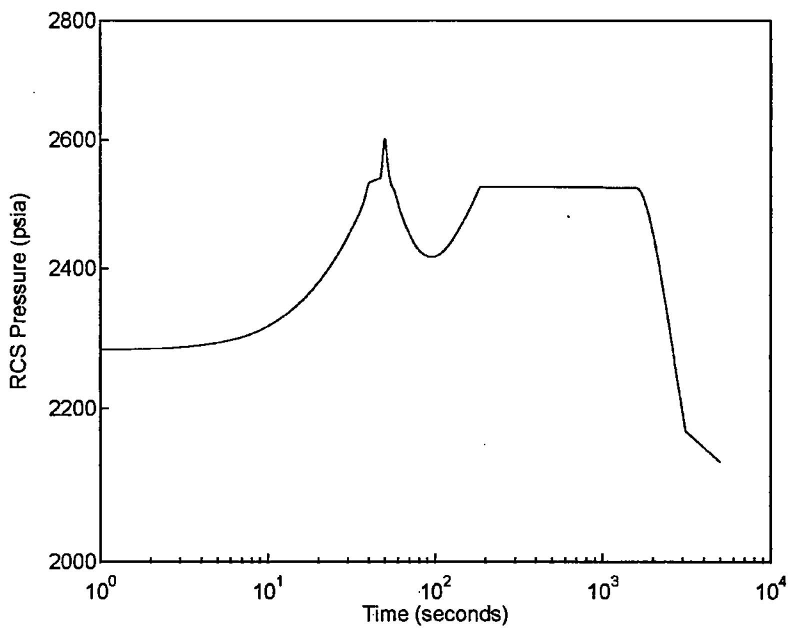


Figure 6.5-2 RCS Pressure versus Time
Feedwater System Pipe Break

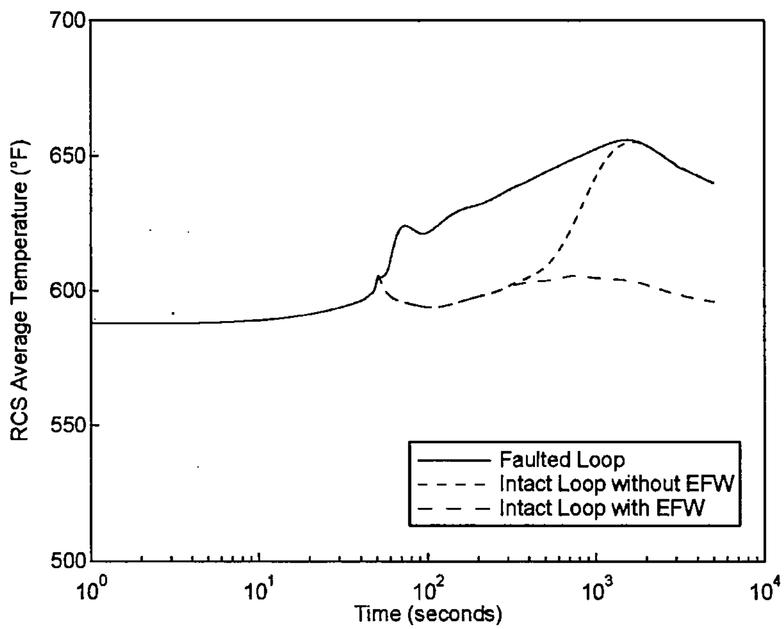


Figure 6.5-3 RCS Average Temperature versus Time Feedwater System Pipe Break

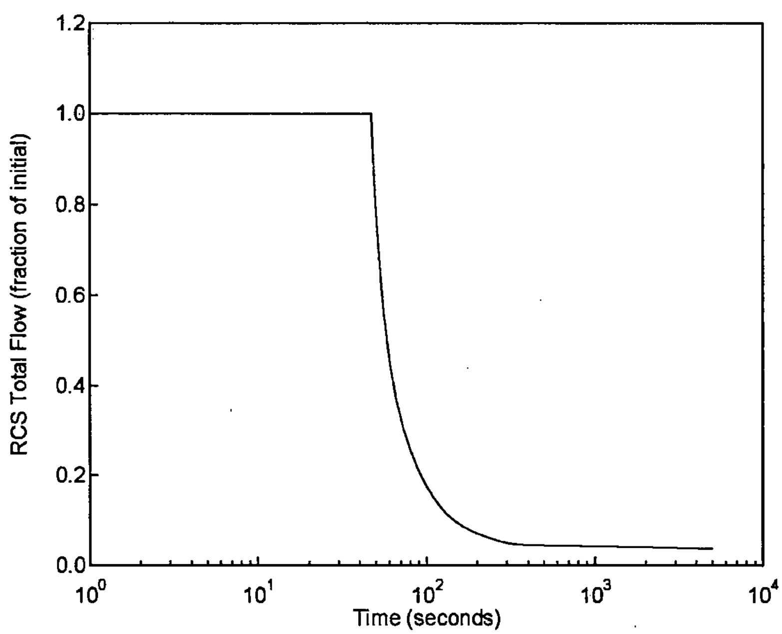


Figure 6.5-4 RCS Total Flow versus Time Feedwater System Pipe Break

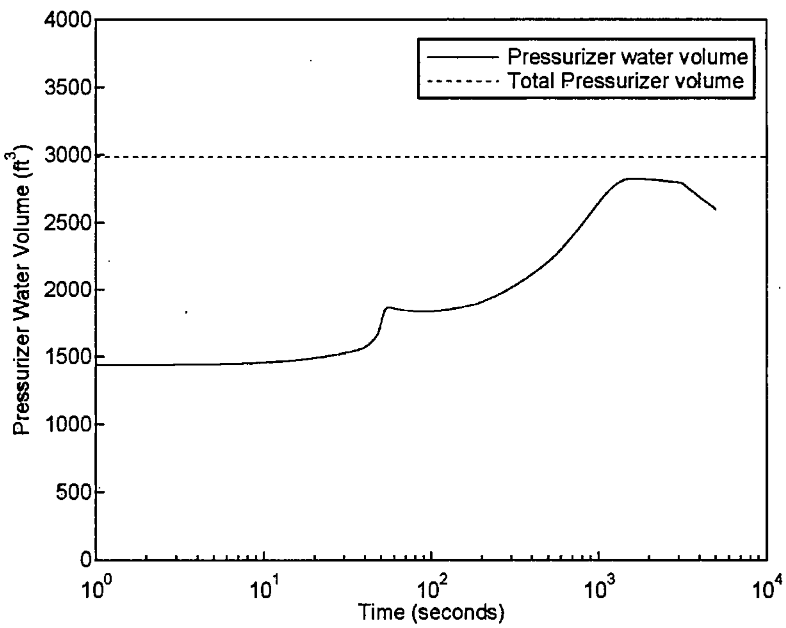


Figure 6.5-5 Pressurizer Water Volume versus Time Feedwater System Pipe Break

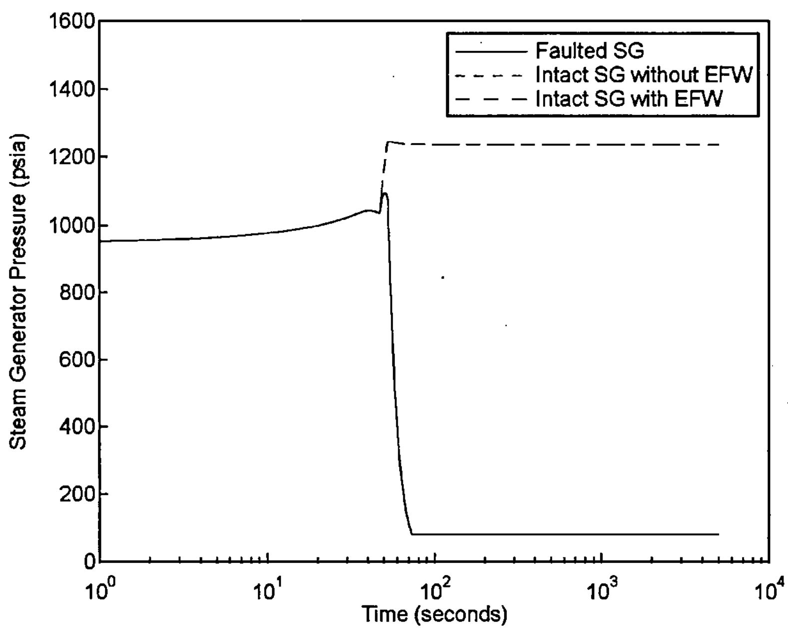


Figure 6.5-6 Steam Generator Pressure versus Time Feedwater System Pipe Break

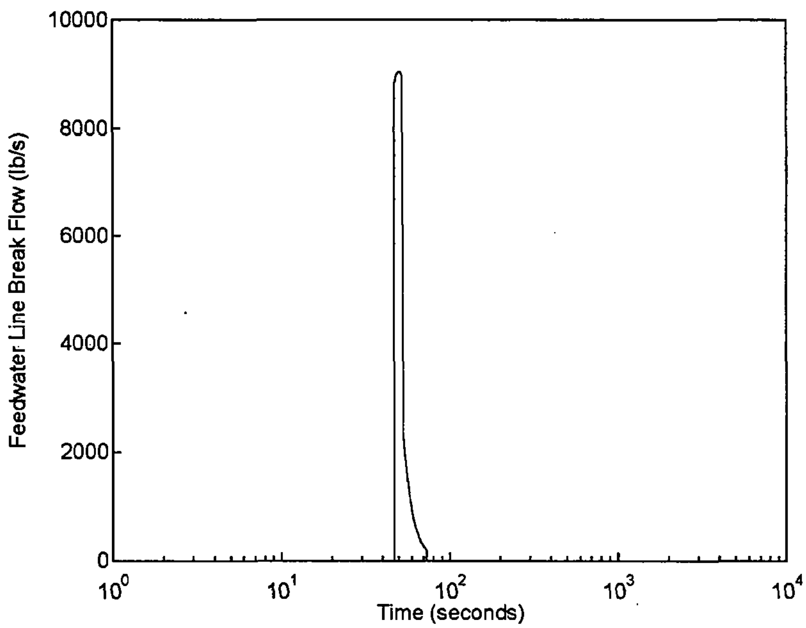


Figure 6.5-7 Feedwater Line Break Flow Rate versus Time Feedwater System Pipe Break

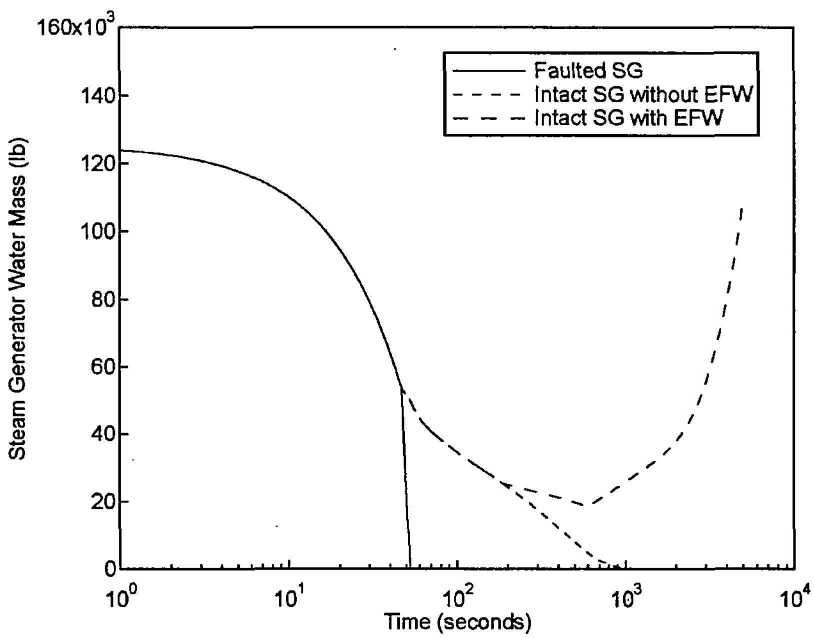


Figure 6.5-8 Steam Generator Water Mass versus Time Feedwater System Pipe Break

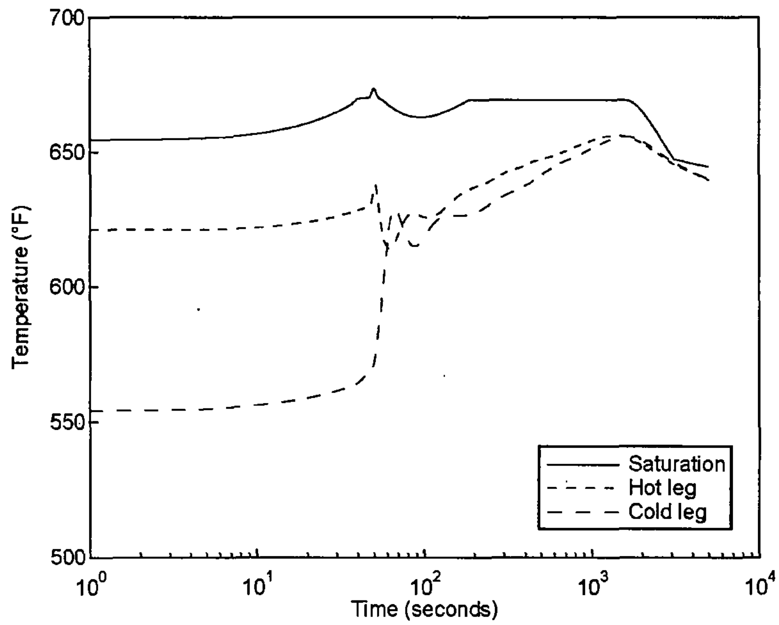


Figure 6.5-9 Temperature versus Time for the Faulted Loop Feedwater System Pipe Break

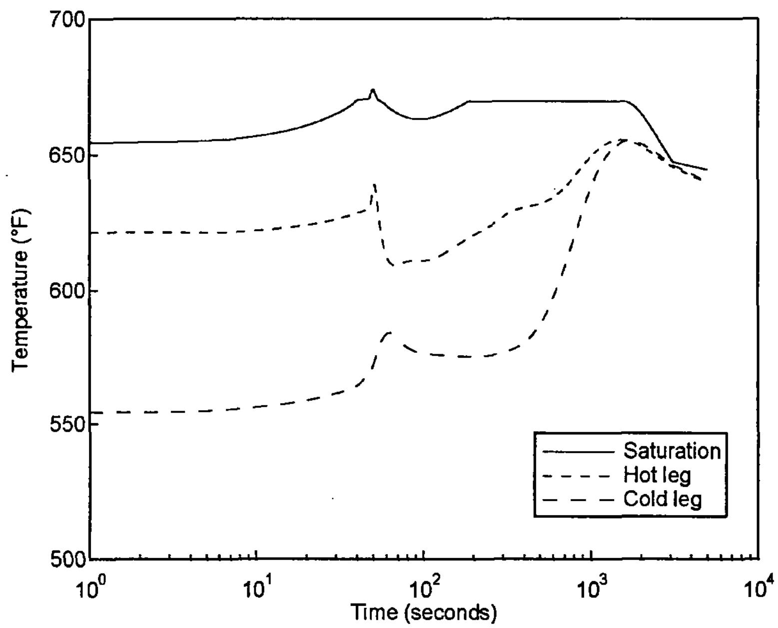


Figure 6.5-10 Temperature versus Time for the Intact Loop without EFW Feedwater System Pipe Break

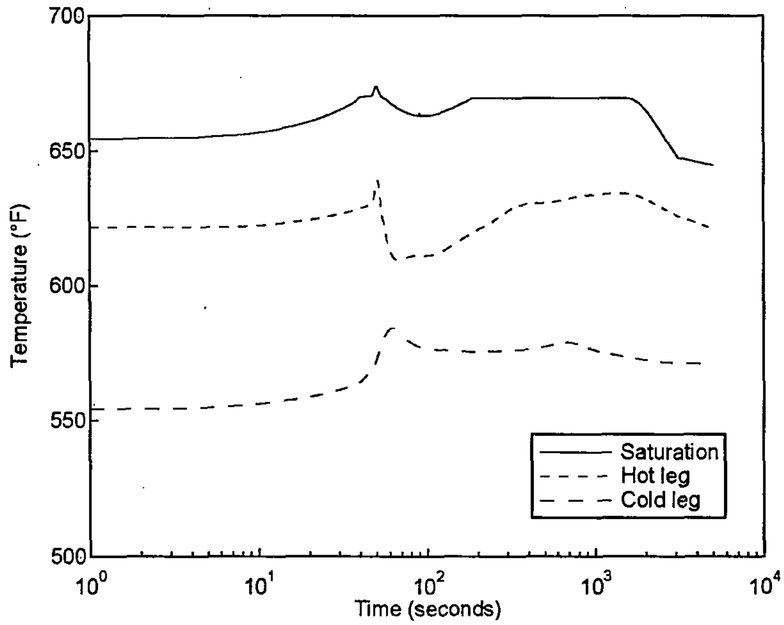


Figure 6.5-11 Temperature versus Time for the Intact Loop with EFW Feedwater System Pipe Break

6.6 Steam Generator Tube Rupture

Event Description

The SGTR event is initiated by a complete severance of a single steam generator U-tube. Leakage of coolant from the primary to secondary side leads to a decrease in the reactor coolant inventory and pressure. The break flow exceeds the makeup capacity of the charging pump causing the pressurizer pressure and level to decrease, leading to a pressurizer pressure low trip.

Events Analyzed

This analysis evaluates the effects of the most limiting SGTR event, which is a double-ended break of a single SG U-tube on the cold leg side just above the tube sheet. The event is terminated when primary-to-secondary leakage stops, which occurs when the RCS pressure is reduced to below the secondary pressure of the ruptured steam generator. Operator action is necessary to recognize the event as a SGTR, terminate ECCS (safety injection), open the pressurizer depressurization valve, isolate the ruptured steam generator, and establish RCS cooling using the main steam relief valves on the intact steam generators to terminate this event. The overall response of the primary and secondary systems is evaluated using the MARVEL-M code.

This event was chosen as one of the six sample analyses because it uses MARVEL-M to model primary-to-secondary coolant flow.

The SGTR is analyzed using different assumptions for the steam generator overfill and the radiological consequence cases. The sample transient analysis presented in this section is for the radiological consequence case.

Analysis Results

Table 6.6-1 presents the sequence of events for the SGTR radiological consequence analysis. Plots of key parameters are presented in Figures 6.6-1 through 6.6-9.

The SGTR is assumed to occur at time = 0. The primary side will exhibit coolant activity during power operation from N-16 and, if present, from fission gap activity as permitted by the Technical Specifications. Following the SGTR, a combination of decreasing pressurizer pressure, decreasing pressurizer level, increasing level in one steam generator, and increased steam line N-16 radioactivity in the same steam generator will alert the operator that a SGTR is in progress as described in Section 5.6. A manual reactor trip is assumed at time = 900 seconds. The pressurizer pressure low reactor trip and over temperature ΔT high trips will protect the DNB limits in the event the manual trip has not occurred.

The primary-to-secondary flow is established by MARVEL-M as described in Section 5.6. When the pressurizer pressure reaches the ECCS setpoint, safety injection will start and deliver flow to the RCS at pressures below the pump shutoff pressure. At time =1500 seconds, the operators establish secondary cooling with opening the steam generator relief valves in the intact steam generators. As described in Section 5.6, the relief valve on the ruptured steam generator is assumed to fail open and is automatically isolated by the in-line block valve on a pressure signal. At time = 2642 seconds, the operator further reduces primary-to-secondary differential pressure by opening the pressurizer depressurization valve based on sub-cooling margin criteria. The pressurizer depressurization valve is closed when the primary and secondary pressure are equal, which occurs at time = 2972 seconds. At time = 2982 seconds the operator is assumed to have terminated safety injection based on RCS pressure termination criteria. At time = 3934 seconds, the primary pressure is reduced below the ruptured steam generator pressure, and the event is terminated.

Table 6.6-1 Sequence of Events for the SGTR (Radiological Consequence Analysis)

Event	Time (sec)
SG Tube Rupture Occurs	0
Manual Reactor Trip (Rod Motion Begins)	900
Ruptured SG Isolated (EFWS, MSIV)	1200
SG Cooling Established (Intact SGs)	1500
Pressurizer Pressure Low-Low ECCS Analysis Limit Reached	1554
EFWS Pumps Start	1694
Open Pressurizer Depressurization Valve	2642
Close Pressurizer Depressurization Valve	2972
Terminate Safety Injection	2982
Primary Leakage Terminated	3934

Conclusions

This sample analysis is for the radiological release analysis and does not address the acceptance criteria described in Section 4.7. A separate transient for fuel failure analysis and steam generator overfill are not included in this topical report.

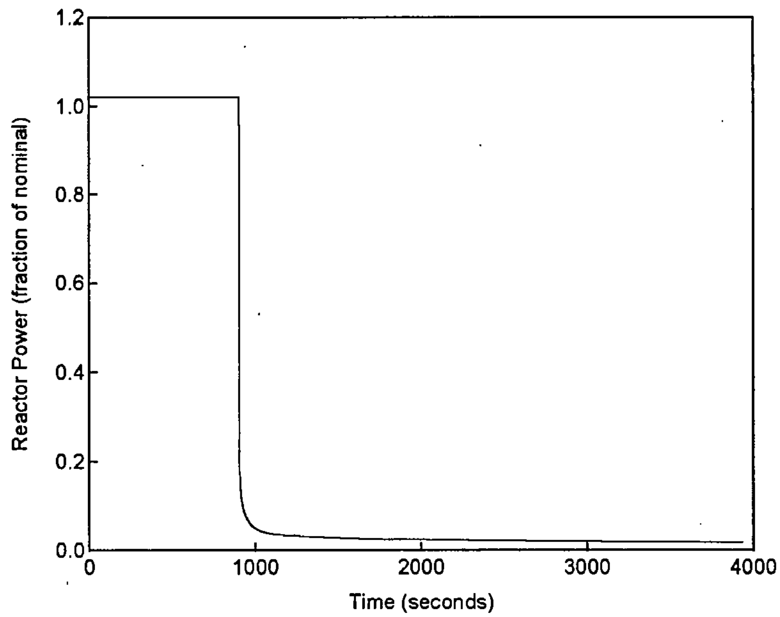


Figure 6.6-1 Reactor Power versus Time SGTR Radiological Consequence Analysis

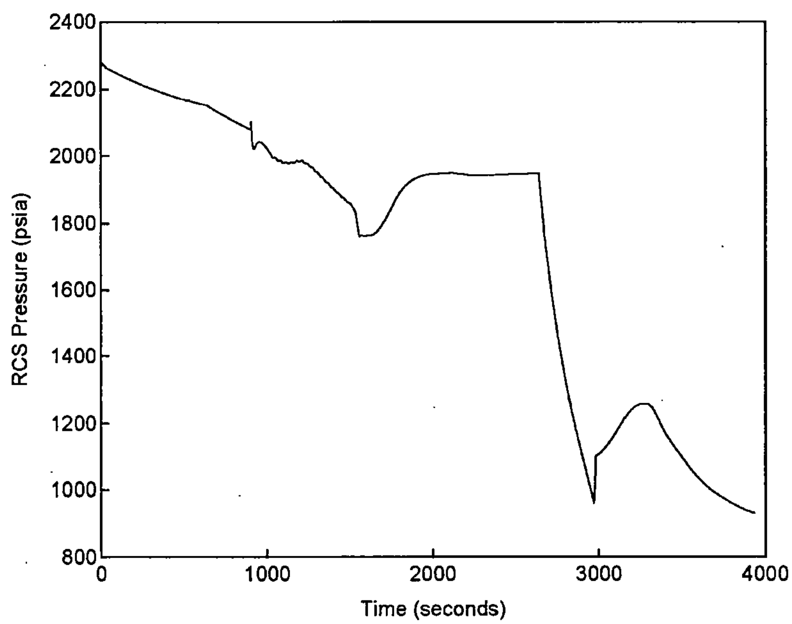


Figure 6.6-2 RCS Pressure versus Time SGTR Radiological Consequence Analysis

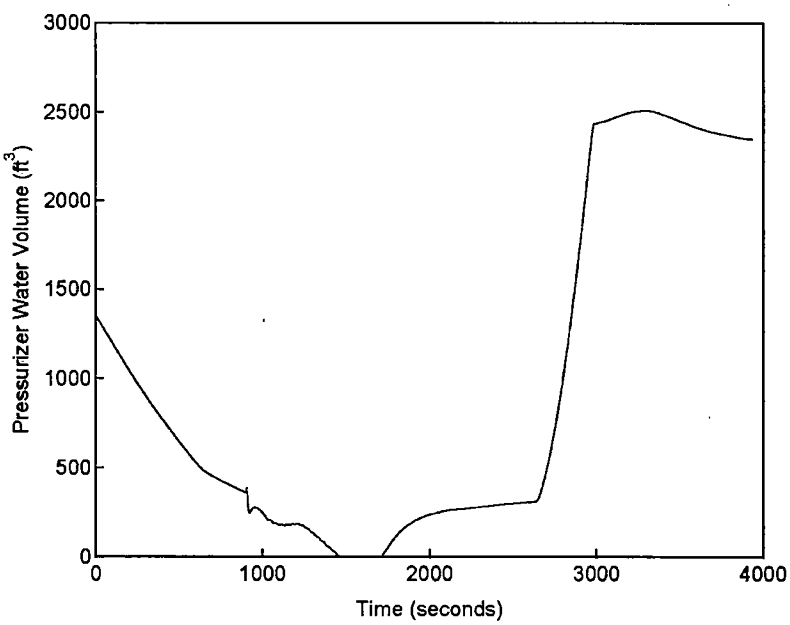


Figure 6.6-3 Pressurizer Water Volume versus Time SGTR Radiological Consequence Analysis

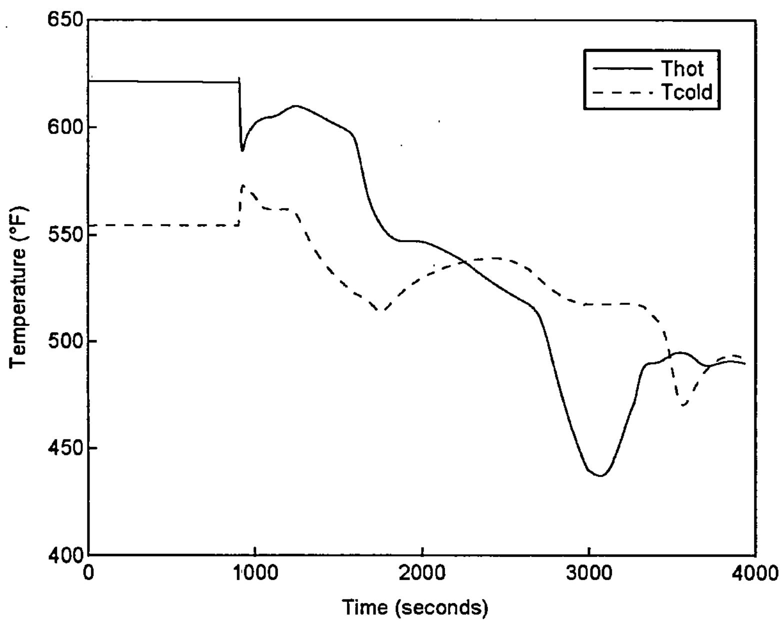


Figure 6.6-4 Ruptured Loop RCS Temperature versus Time SGTR Radiological Consequence Analysis

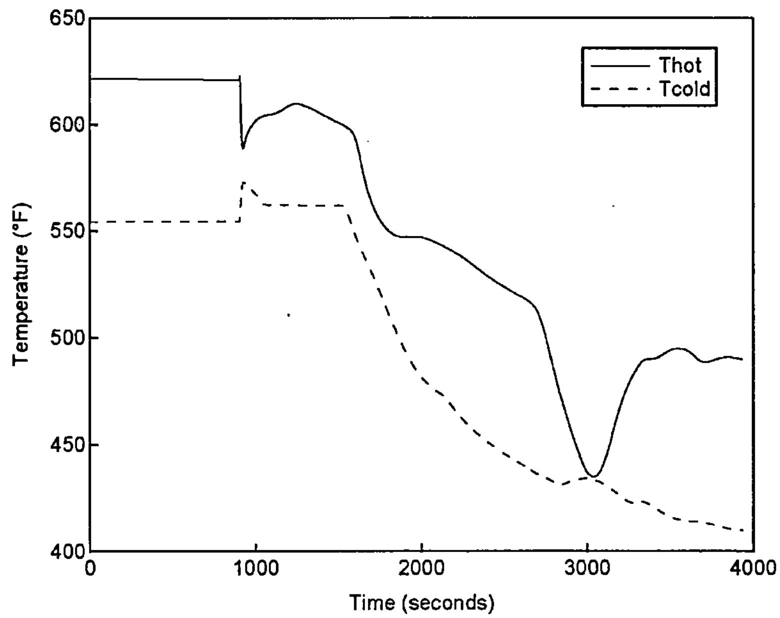


Figure 6.6-5 Intact Loop RCS Temperature versus Time SGTR Radiological Consequence Analysis

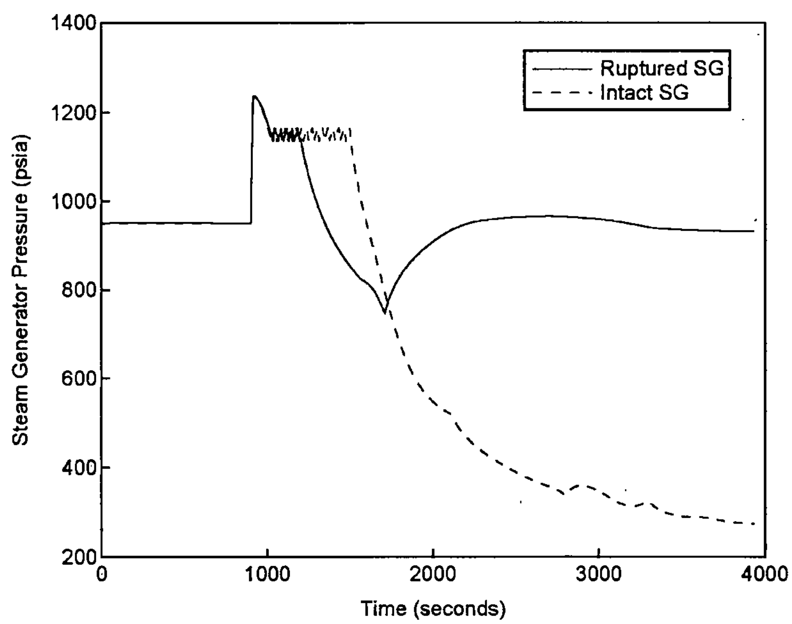


Figure 6.6-6 Steam Generator Pressure versus Time SGTR Radiological Consequence Analysis

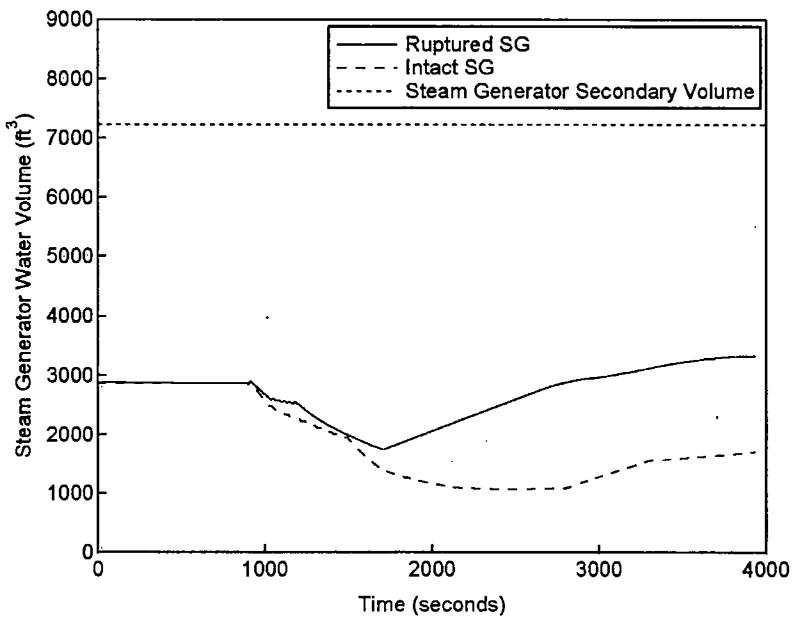


Figure 6.6-7 Steam Generator Water Volume versus Time SGTR Radiological Consequence Analysis

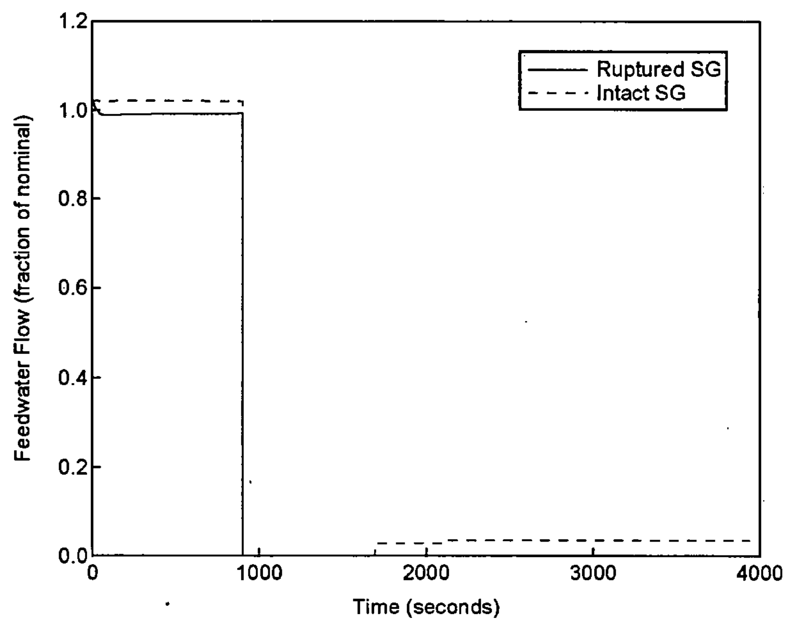


Figure 6.6-8 Feedwater Flow Rate versus Time SGTR Radiological Consequence Analysis

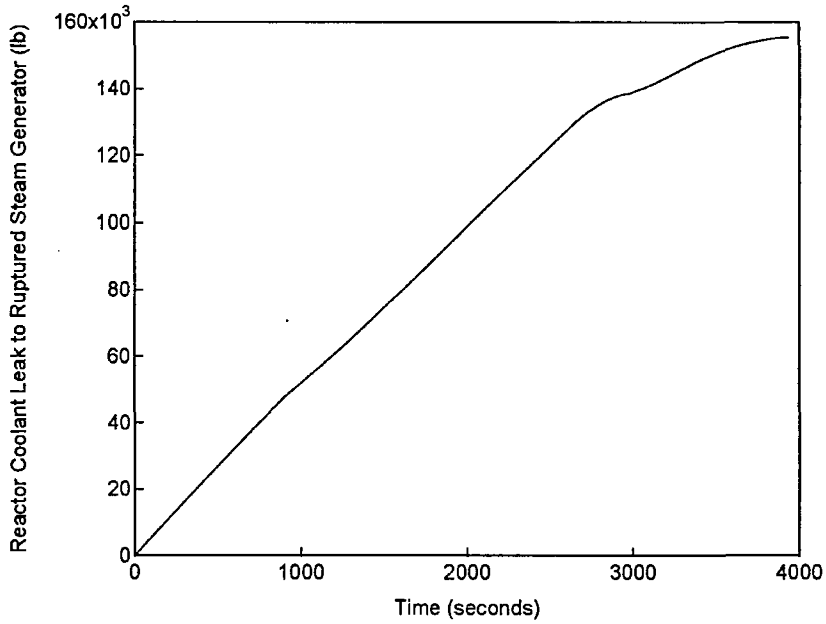


Figure 6.6-9 Reactor Coolant Leakage versus Time SGTR Radiological Consequence Analysis

7.0 CONCLUSIONS

The US-APWR is an advanced PWR design that is functionally similar to existing plants and fuel designs from the perspective of non-LOCA accident analysis. The advanced features of the US-APWR have not created accidents of a different type that are not covered by Chapter 15 of the existing NRC Standard Review Plan, and event classifications and acceptance criteria to be used in the Design Certification License Application have been defined based on existing regulations and regulatory guidance. MHI uses codes and methodologies for non-LOCA analyses of the US-APWR that are similar to NRC-approved codes and methodologies used to evaluate existing plants and fuel. Changes to the codes previously approved by the NRC have been described, justified, and validated by this report.

The codes and methodologies examined were:

- MARVEL-M Plant system transient analysis code
- TWINKLE-M Multi-dimensional neutron kinetics code
- VIPRE-01M Subchannel thermal hydraulics analysis and fuel transient code

Following are confirmed by the analyses in this topical report.

- that the US-APWR responses to various initiating events or conditions are similar to the responses of existing designs and within the range of applicability of MARVEL-M, TWINKLE-M, and VIPRE-01M, and
- that the physical characteristics and phenomena governing the US-APWR responses are similar to those phenomena governing the responses of existing plants.

On the basis of the information in this topical report, it was concluded that the existing codes and methodologies are appropriate for US-APWR analyses. Also, it is concluded that the information provided in this topical report supports its purpose to provide key technical information related to the computer codes, key methods and models and their applicability, event-specific acceptance criteria, and sample results to the NRC during the pre-application phase to facilitate an efficient and timely review of the Design Certification Application.

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APPENDIX A
Evaluation of MARVEL-M DNBR Calculation Method

This appendix evaluates the method of MARVEL-M DNBR calculation. MARVEL-M interpolates DNBR from DNBR data tables. These tables define the relationship between DNBR, the core inlet temperature, the core pressure and the core heat flux made by VIPRE-01M steady state calculations. MARVEL-M DNBR result is compared with the DNBR results obtained by VIPRE-01M steady state calculation for an uncontrolled RCCA bank withdrawal at full power.

Case 1: High reactivity insertion rate at 75 pcm/sec
Case 2: Low reactivity insertion rate at 2.5 pcm/sec

Figure A-1 and A-2 show the MARVEL-M DNBR transient response for Case 1 and Case 2 respectively. MARVEL-M DNBR coincides with the VIPRE-01M DNBR. Therefore, the MARVEL-M is able to calculate DNBR adequately.

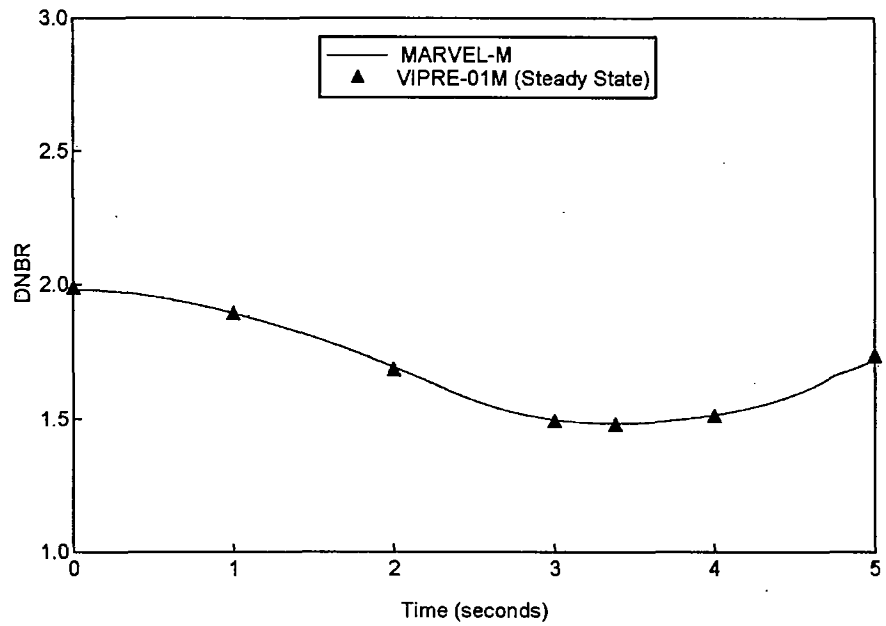


Figure A-1 DNBR Transient, Uncontrolled RCCA Bank Withdrawal at Case 1- Full Power for a High Reactivity Insertion Rate

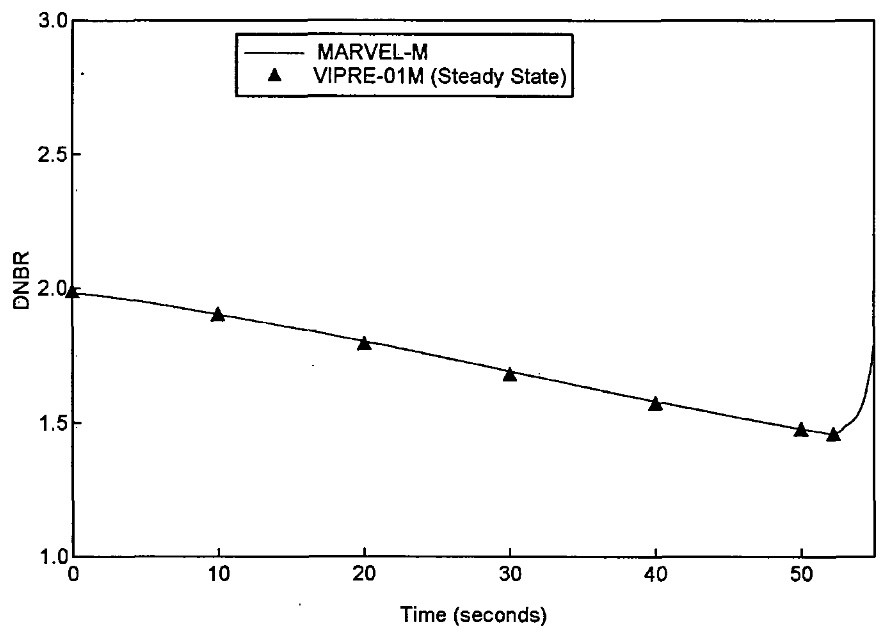


Figure A-2 DNBR Transient, Uncontrolled RCCA Bank Withdrawal at Case 2 - Full Power for a Low Reactivity Insertion Rate

Appendix B
Sensitivity Study of the RCCA Ejection In the 3-D Methodology

Sensitivity Study of the RCCA Ejection in the 3-D Methodology is performed with two key parameters, an ejected reactivity and a hot channel factor, in order to present conservativeness of the MHI RCCA ejection 3-D methodology as described in Section 5.3. The following cases are performed in US-APWR 24 month equilibrium core at the end-of-cycle (EOC).

Case 1: The ejected reactivity and the hot channel factor are the best estimate which does not include uncertainties.
Case 2: The ejected reactivity is adjusted to the design limit from the case 1. The design limit includes safety margins and uncertainties.
Case 3: Conditions of core kinetics are the same as case 2. In the VIPRE-01M calculation, the hot channel factor is adjusted to the design limit.

Note: Other parameters such as delayed neutron fraction, Doppler temperature coefficient and moderator temperature coefficient are the same in three cases.

Table B-1 shows the results of a total maximum fuel enthalpy and a prompt maximum fuel enthalpy rise (adiabatic fuel enthalpy rise) in three cases with these calculation conditions. In the 3-D methodology, a large Doppler feedback effect which consists with a large power peaking factor during transient is expected.

The case 2 and the case 3 have conservative assumptions in the viewpoint of the Doppler feedback effect. In case of the ejected reactivity becomes large, the hot channel factor generally becomes large and the Doppler feedback is expected to be large. However, the case 2 and the case 3 use the same hot channel factor as the case 1 and ignore an increase in effect of the Doppler feedback in the core kinetics. In addition, case 3 has more conservative assumption which is adjusted the maximum hot channel factor to the design limit in the hot spot thermal calculation. This method also ignores an increase in effect of the Doppler feedback.

The case 3 which is licensing case has large conservativeness based on the 3-D methodology and covers uncertainties of parameters and many core variations in the core design.

Table B-1 Calculation Condition and Results of Sensitivity Studies about an Ejected Reactivity and a Hot Channel Factor in the RCCA Ejection

	Case 1	Case 2	Case 3
Reactivity insertion in the TWINKLE-M	600 pcm (Best limit)	800 pcm[*1] (Design limit)	Same as Case 2
Maximum hot channel factor in the TWINKLE-M	27.4 (Best estimate)	Same as Case 1	Same as Case 1
Maximum hot channel factor in the VIPRE-01M	Same as TWINKLE-M[*2]	Same as Case 1	35[*3] (Design limit)
Total maximum fuel enthalpy (cal/g)	45.9	68.4	77.8
Prompt maximum fuel enthalpy rise (cal/g) at T_e[*4]	18.0	43.5	51.6

*4: Corresponding to one pulse width after the peak of the prompt pulse as shown in Figure B-1

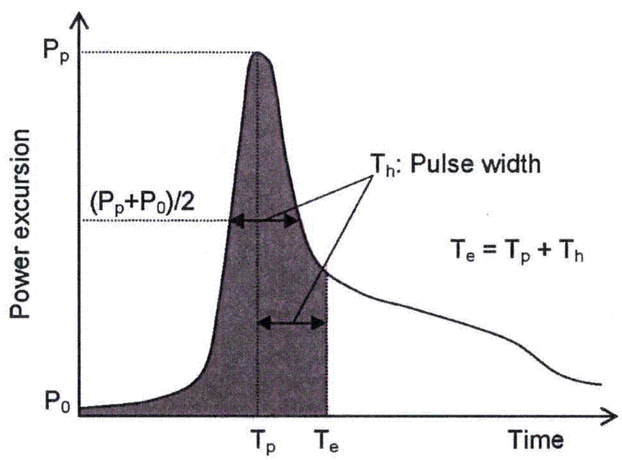


Figure B-1 Definition of the Prompt Maximum Fuel Enthalpy Time

Appendix C
Doppler Weighting Factor of the RCCA Ejection in the 1-D Methodology

A Doppler Weighting Factor (DWF) in the 1-D methodology is used to correct an underestimated effective fuel temperature rise by ignoring an increase in a radial peaking factor. The DWF that is adopted for MHI-designed PWRs is defined as a function of the radial peaking factor (Fxy) as shown in Figure C-1. Conservatism of this DWF is confirmed by the comparison with a 3-D kinetics and a 1-D kinetics results.

Table C-1 shows calculation conditions in US-APWR 24 month equilibrium core at the BOC and EOC. All the key parameters are adjusted to the same condition both in the 3-D and the 1-D analyses. A reactor trip is not simulated to make clear the difference of transient between the 3-D and the 1-D with DWF.

Comparison with the 3-D and the 1-D results of nuclear power transient are shown in Figure C-3 (BOC) and Figure C-4 (EOC). It concludes that the DWF used to the 1-D methodology has a large safety margin during transient.

Table C-1 Calculation Condition of the RCCA Ejection in the Hot Full Power

	BOC (1D / 3D)	EOC (1D / 3D)
Reactivity insertion[*1] (pcm)	112	138
Effective delayed neutron fraction (%)	0.49	0.44
Prompt neutron lifetime (micro sec.)	8.0	8.0
Doppler temperature coefficient (pcm/°F)	-1.43	-1.65
Average axial power distribution	Figure C-2	Figure C-2
DWF[*2]	( )	( )

*1: External reactivity to prevent the power distribution changes by rod motion
*2: Used in the 1-D methodology only

Figure C-1 Radial Doppler Weighting Factor for 1-D Kinetics Analysis

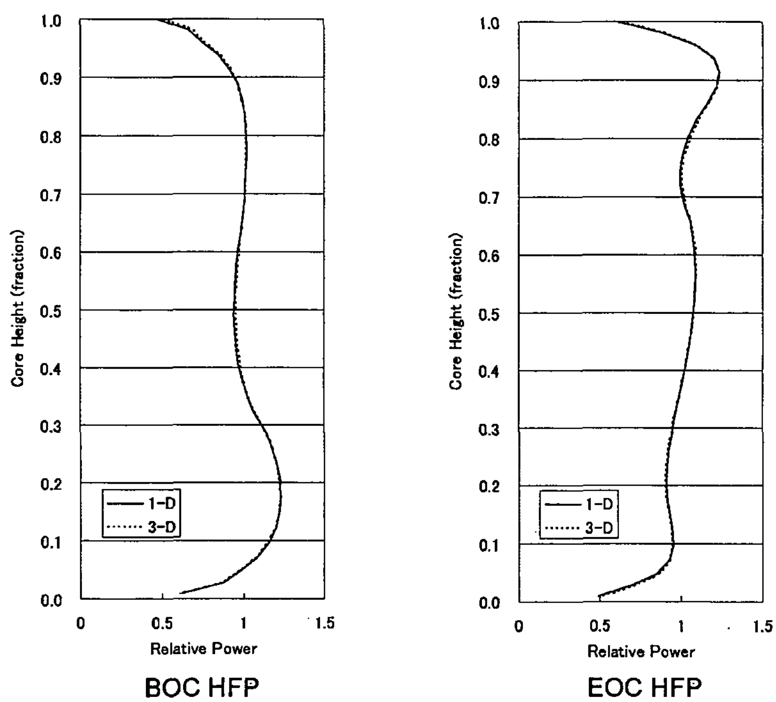


Figure C-2 Average Axial Power Distribution Comparison with 3-D and 1-D (BOC, EOC)

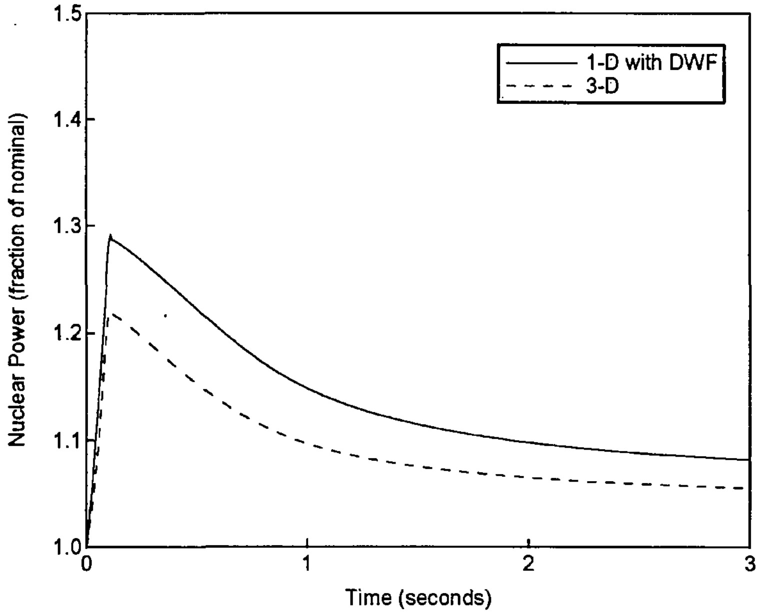


Figure C-3 Nuclear Power versus Time (BOC)
Comparison between 3-D and 1-D with Doppler Weighting Factor

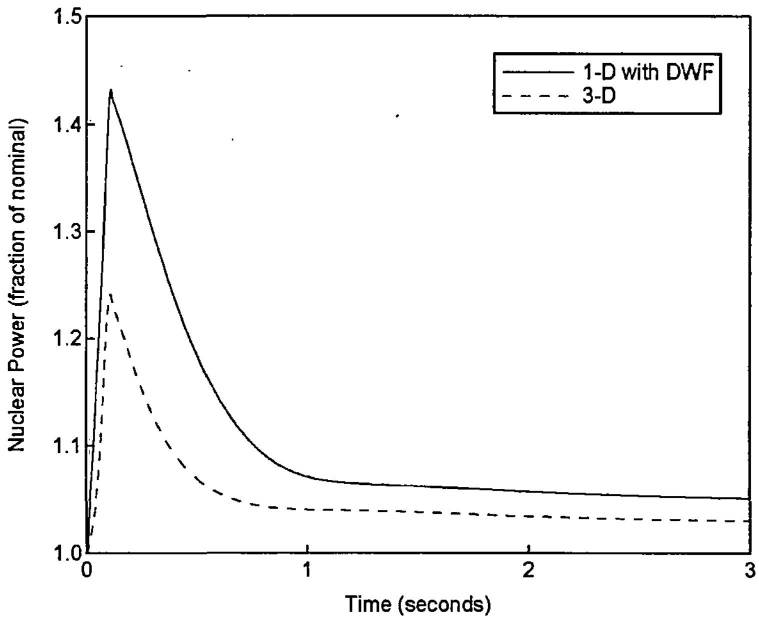


Figure C-4 Nuclear Power versus Time (EOC)
Comparison between 3-D and 1-D with Doppler Weighting Factor

Appendix D
Validation of VIPRE-01M Modeling for Steam System Piping Failure

One-eighth symmetric core model is used in the DNB evaluation for the steam system piping failure. In the model, a hot assembly is assumed to be located in the core center. The core power distribution considering a stuck rod and core inlet temperature distribution are averaged in each of 5 groups, as shown in Figure 5.4-3.
It is confirmed that typical VIPRE-01M analysis model, which has detailed subchannels surrounding the hot channel while the lumped channels represent the remains of the core, can provide sufficiently accurate DNBR [Reference 6]. However the additional study is needed for the applicability of such relatively coarse channel definition and averaged power distribution to the core condition with very steep and asymmetric radial power distribution that may be found in steam system pipe failure event.
The model is validated by comparing with a detailed full core model shown in Figure D-1. Table D-1 shows the calculation model and assumptions. Using a 5-grouped model for a radial power distribution, a detailed power distribution in rod-by-rod is simulated as shown in Figure D-2. Axial distributions of the DNBR and other local fluid parameters at the hot channel in both cases are compared in Figure D-3.
The results show that DNBR evaluated by the both models are in good agreement. The coarse channel definition for the peripherals and the roughly grouped radial power distribution does not affect the prediction of DNBR and other local fluid parameters in the hot channel significantly, in spite of the remarkable radial power distribution. It concludes the one-eighth symmetric core model possesses sufficient calculation accuracy to evaluate the minimum DNBR in the steam system piping failure.

Table D-1 DNBR Calculation Model and Assumptions for the Steam System Piping Failure

Figure D-1 VIPRE-01M Full Core Analysis Modeling (17x17-257FA Core, 4-Loop Plant)

Figure D-2 5-grouped Model for Radial Power Distribution

Figure D-3 Comparison of DNBR and Local Fluid Parameters at Hot Channel between the 1/8 Core Model and the Full Core Model

Appendix E
Sensitivity Study of the Inlet Mixing Coefficient for Steam System Piping Failure

The Steam System Piping Failure at hot zero power condition is a transient that is characterized by non-uniform cooling in combination with the assumption that the most reactive control rod be fully withdrawn. The colder inlet temperature at the stuck rod position results in the increase of radial power peaking factor and the decrease of minimum DNBR.

The effect of changes in the vessel inlet mixing factor is shown in Figure E-1. As seen from Figure E-1, assuming no reactor vessel inlet mixing results in a very small reduction in the minimum DNBR and maintains significant margin to the limit.

Figure E-1 DNBR versus Reactor Inlet Mixing for the Steam System Piping Failure

Appendix F
Detailed Break Flow Model for Steam Generator Tube Rupture

This appendix provides a comparison of the conservative break flow model used in the MARVEL-M SGTR analysis with a realistic and more detailed break flow model.

The break flow models applied to licensing analysis and backup analysis

- A simple but conservative break flow model is applied to licensing analysis.
- To indicate the conservatism of this break flow model, its flow is compared to that of the "realistic steam generator tube leak flow model" (realistic model).
- The description of the each model and result of comparison are outlined below.

Description of the conservative break flow model

- The initial break flow rate is calculated conservatively by the Zaloudek correlation applicable to single-phase flow. The conservatism is maintained by adding margin to the value calculated by the Zaloudek correlation.
- The break flow rate in the transient is calculated based on differential pressure of the primary system and secondary system. It is assumed that break flow rate in the transient is proportional to the square root of the differential pressure as following equation.

$$G = G_0\sqrt{\frac{\Delta P_t}{\Delta P_{nom}}}$$

where

G : Break flow rate

G_0 : initial value of the break flow rate

ΔP_{nom} : Differential pressure between primary and secondary system at initial state

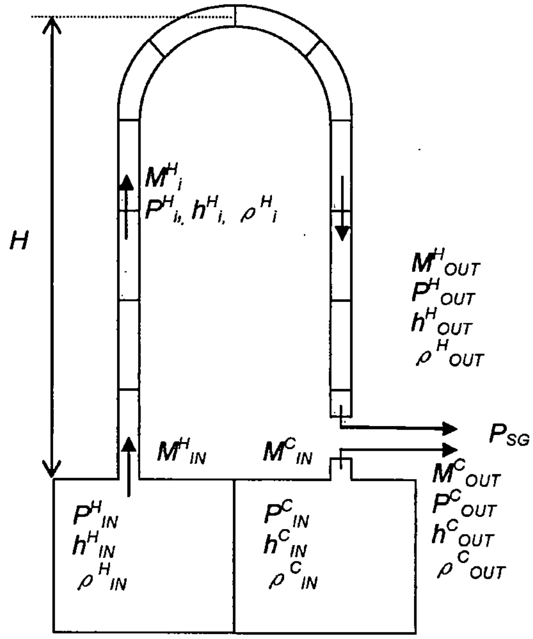
ΔP_t : Differential pressure between primary and secondary system at time = t

Description of the realistic break flow model

- The double ended ruptured SG tube is modeled independently. Also friction loss and resistance due to form losses are considered.
- The conceptual diagram of the realistic model is shown in Figure F-1. For each node, a mass, energy, and momentum balance is performed based on each node's mass flow rate, pressure, enthalpy, and coolant density.

Results of the safety analysis based on each break flow model

- It is assumed that a single tube is ruptured just above the tube sheet of the SG outlet plenum (cold side), because coolant density is the highest at this position in the tube. The type of the rupture is assumed to be a double-ended rupture. For comparison, calculations have also been performed for a rupture just above the tube sheet of the SG inlet plenum (hot side) and a rupture at top of the U-bend. These results are shown in Figure F-2.
- The integrated break flow of the conservative model is larger than that of the realistic model which is evaluated conservatively with discharge coefficient of 1.0 independent of the position of the rupture. Thus, the break flow model applied to licensing analysis is conservative because it bounds the flow predicted by a realistic model using upper-bound assumptions.



Parameter key
- *H*: height
- *M*: mass flow rate
- *P*: pressure
- *h*: enthalpy
- ρ: density

Subscript & Superscript Key
- *C*: Cold leg
- *H*: Hot leg
- *IN*: Entrance
- *OUT*: Outlet (at break)
- *i*: i-th node
- *SG*: Secondary side of the

- This is sample of rupture just above tube sheet of cold leg.
- The broken tube is modeled independently.

Figure F-1 Realistic Model of the Broken SG Tube

Figure F-2 Comparison of the Calculation Result for Each Break Flow Model (Including comparison of the break position)