Mitsubishi Reload Evaluation Methodology

Non-Proprietary Version

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Revision History

Revision	Page	Description
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ABSTRACT

This report describes the Mitsubishi reload evaluation methodology, which is used to realize safe reload cores in compliance with regulatory and customer requirements.

The reload design and evaluation process outlines the steps to:
- Develop the reload loading pattern that complies to operational and licensing requirements
- Compare and evaluate key safety input parameters relative to the reference analyses to determine the extent of reevaluation or reanalysis necessary
- Provide a summary report described reload design results to a licensee

The report includes a summary of the safety analyses that are considered for the reload, as well as the nuclear and thermal-hydraulic inputs to the safety analysis.

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List of Acronyms

ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrence
APWR	Advanced Pressurized Water Reactor
ASME	American Society of Mechanical Engineers
BOC	beginning-of-cycle
COLR	core operating limits report
CVCS	chemical and volume control system
CWO	core wide cladding oxidation
DCD	design control document
DV	depressurization valve
EAB	exclusion area boundary
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
ECCS	emergency core cooling system
EFW	emergency feedwater
EFWS	emergency feedwater system
ESF	engineered safety features
EOC	end-of-cycle
FSAR	final safety analysis report
HFP	hot full power
HZP	hot zero power
ICCC	incore control component
LHR	linear heat rate
LMO	local maximum cladding oxidation
LOCA	loss-of-coolant accident
LOOP	loss of offsite power
LPZ	low-population zone
MHI	Mitsubishi Heavy Industries, Ltd.
MSRV	main steam relief valve
MSSV	main steam safety valve
MOC	middle-of-cycle
NRC	U.S. Nuclear Regulatory Commission
NSSS	nuclear steam supply system
PA	Postulated Accident
PCMI	pellet/cladding mechanical interaction
PCT	peak cladding temperature
PWR	pressurized-water reactor
RCCA	rod cluster control assembly
RCP	reactor coolant pump
RCS	reactor coolant system
RTDP	revised thermal design procedure
RTP	rated thermal power
RTS	reactor trip system
SDV	safety depressurization valves
SG	steam generator
SGTR	steam generator tube rupture

SIS	safety injection system
SRP	Standard Review Plan

1.0 INTRODUCTION

This report describes the Mitsubishi reload evaluation methodology, which is used to confirm the validity of reload core parameters on the reference safety analysis of a plant. The safety analysis for the plant is usually established for the initial core, and revised as necessary for each reload. This existing safety analysis is generally referred to as the reference safety analysis.

The objective of the safety evaluation for each reload is to verify that the reference safety analysis limits are met, or to identify which transients need to be reevaluated based on the changes caused by the reload or other factors (such as new information based on industry experience). The Mitsubishi safety analysis methodology is based upon the 'bounding analysis' concept, in which a set of key safety input parameters are generated to bound all future anticipated operating cycles. For each reload, the key safety analysis inputs are reevaluated, recalculated, and compared to the bounding values used for the reference analyses. If the key safety input parameters are bounded, the reference safety analysis is valid, otherwise, additional evaluation or analysis is required to demonstrate that safety criteria and design bases are met. Although the reference safety analysis is intended to be valid for all plant reloads, changes to the plant, significant changes in anticipated fuel management, or other issues may require additional analyses. Required events are reevaluated and startup tests also confirm that measured parameters are within acceptable bounds as required by technical specifications. If a potential unreviewed safety question is identified, or technical specification changes are required, the license holder is informed to take action as necessary in accordance with 10 CFR 50.59.

Section 2 describes the reload design process which can be mainly divided into design initiation, preliminary design, a safety evaluation, finalization phase and documentation phases.

Section 3 describes the safety evaluation mentioned above in detail. A process for determining key safety input parameters needed to be evaluated in each reload safety evaluation is also described.

Sections 4 and 5 provide calculation process and concept of each key safety input parameter defined at Section 3 in the nuclear and thermal and hydraulic design process.

Based on the reload safety analyses and evaluations described in Section 3, a summary report of reload core design including reload safety evaluation results is provided by Mitsubishi to the licensee, which has the responsibility for plant safety and licensing with regulatory authorities.

2.0 RELOAD DESIGN PROCESS

2.1 Introduction

The Mitsubishi reload design process includes several phases:

- Design initialization, in which a detailed review of previous cycle operations and current reload objectives are evaluated
- Preliminary design phase, in which a reload loading pattern that meets operations goals and safety requirements is established
- Safety evaluation phase, in which the fulfillment with all safety criteria are demonstrated for a selected core design
- Final design phase, in which reload core design is finalized, considering the final changes
- Documentation

Figure 2-1 shows a typical flowchart of the Mitsubishi reload design process.

The safety of the reload core is confirmed through these five phases. In the first and second phases, a preliminary loading pattern is established based on nuclear design process. In the third phase, key safety input parameters are evaluated for the preliminary loading pattern. And it is confirmed whether the reference safety analysis remains valid or a safety re-analysis is necessary. Therefore, the safety evaluation phase is the most important phase in the reload design process. After this safety evaluation, the reload design is finalized and the design results including the safety evaluation results are reported to a licensee.

Each phase is described in detail in the following subsections.

2.2 Design Initialization

The design for a reload cycle is based on the objectives, requirements and constraints for the cycle being designed. As part of the design initialization, information must be collected to ensure that the reload evaluation will be based on the actual fuel and core components that are in the plant, the operating history and the plant system changes projected for the next cycle. Typical examples of information collected prior to the reload design initiation may include:

- Cycle Length and Operational Objectives
  - Power
  - Cycle burnup
  - Outage length
  - Planned capacity factor
  - Planned load follow or extended operation at partial power
  - Coastdown length and type (power vs. temperature)

- Core Design Risk & Margin Management
  - Optimization of operating margins
  - Minimization of operational anomalies risk
  - Minimization of number of fresh assemblies loaded in order to fill the spent fuel pool as slowly as possible

  - Minimization of fast neutron fluence to the reactor pressure vessel to reduce the rate of neutron Irradiation embrittlement
- Core Design Guidelines
  - Peaking factor limits
  - Reactivity temperature coefficients
  - Minimum shutdown margin

- Licensing
  - Reduction of Technical Specifications changes that would be required in order to operate the new core

In addition, cycle specific issues may include concerns stemming from the operating history of previous cycles, industry experience, fuel design features and innovations, and unexpected component failures in similar plants operating with similar fuel designs.

2.3 Preliminary Design Phase

The reload design must meet safety requirements and the operational requirements identified in Section 2.2 to the fullest extent possible. The preliminary core design includes the following activities:

- Determine the enrichment and number of feed assemblies to meet energy requirements
- Develop a preliminary core loading pattern that meets core design guidelines
- Consider fuel design changes in dimensions and/or materials, and of thermal design changes

At the loading pattern design phase, the preliminary loading pattern, including number and enrichment of feed assemblies and the number of burnable absorbers (if any), are determined and meets following core design guidelines:

- $F_{\Delta H}^{N}$ values with all-rods-out, and D-bank-in to the insertion limit are below specified limits
- Moderator temperature coefficient satisfies technical specification requirements
- Sufficient shutdown margin is available to meet the “N-1” shutdown margin criteria for the entire cycle

Previous cycle experiences (such as extended operation at reduced power) are considered in establishment of the preliminary loading pattern.

The following safety evaluation is performed in order to verify that the safety limits are met for the reload core design

2.4 Safety Evaluation

The reload evaluation methodology is based upon the ‘bounding analysis’ concept. The key safety parameters form the basis to determine whether the reference safety analysis remains valid. Section 3 describes these key safety parameters at each event in detail. The values of (or changes to) key safety input parameters are determined for the reload core during the nuclear, thermal and hydraulic, and fuel design process described in Sections 4 and 5. If relevant parameters are bounded, the reference analysis remains valid and no new analysis is needed to verify that the safety limits are still met. If one or more of the key parameters exceed bounding limits, a reevaluation of affected incidents is performed.

Two types of reevaluations are typically performed:

- If the parameter is only slightly out of bounds, or the transient is relatively insensitive to that parameter, a simple quantitative evaluation may be made which conservatively evaluates the magnitude of the effect and explains why the actual analysis of the events does not have to be repeated.

- If the deviation is large and/or expected to have a significant or not easily quantifiable effect on the incident, a re-analysis of the incident is performed. The analysis methods follow FSAR or subsequent submittals to the NRC by the plant. Changes in the procedures are documented as described in Section 2.6. The re-analyzed incident must be shown to meet the appropriate safety limit for the event in order to have acceptable results. A summary of the safety evaluation, including any required technical specification changes if necessary, is provided.

The reload design is considered as ‘final’ if it meets the following safety requirements:

- Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.

- Fuel cladding integrity should be maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR) remains above the acceptable design DNBR on a 95/95 basis.

- The maximum fuel centerline temperature should be less than the fuel melting point so that the fuel cladding will not be mechanically damaged.

- 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.

For postulated accidents,

- Pressure in the RCS and main steam system should be maintained below the acceptable design limits, considering potential brittle as well as ductile failures.

- Fuel cladding integrity will be maintained if the minimum DNBR remains above the design limits. If the minimum DNBR does not meet these limits, the fuel is assumed to have failed.

- The release of radioactive material should not result in offsite doses in excess of the guidelines of 10 CFR Part 100.

- A postulated accident, should not, by itself, cause a consequential loss of required functions of systems needed to cope with the fault, including those of the RCS and reactor containment barriers.

- In addition, for loss-of-coolant accidents (LOCA), the following 10 CFR 50.46 criteria also apply:
  - The calculated maximum fuel element cladding temperature, total oxidation, and hydrogen generation should not exceed the limits in SRP 15.0.
  - Calculated changes in core geometry shall be such that the core remains amenable to cooling.
  - After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

2.5 Final Design Phase

As mentioned in Section 2.4, the reload core loading pattern is finalized when the reload core design meets all safety criteria.
It is recognized, however, that the design process may require several months from the point of design initialization to the completion of the safety evaluation. During this period, requirements for the reload may be influenced by unanticipated plant-specific or non-plant-specific factors. Examples include:

- A change in the anticipated previous cycle burnup that is outside of the anticipated burnup window (extended burnup or early shutdown)
- A change in anticipated operation (extended operation at part power or significant operation with deep rod insertion)
- Indications of fuel failures
- A significant change in energy requirements for the reload cycle
- Fuel performance issues discovered at a similar plant that require action at other plants
- Other plant-specific or generic safety issues discovered during the design process

As result of these unanticipated changes, the reload may require a reevaluation that typically results in two options:

- The design can be evaluated for impacts, and finalized without a loading pattern change
- The design must be revaluated with a loading pattern change

These two options are discussed in more detail below.

(1) Final Design That Does Not Require a Loading Pattern Change

Depending on the nature of the change, a subset of key safety input parameters can be identified as those requiring reevaluation. In most cases, sufficient margin to limits is available to avoid the need for calculations, but will be performed if required. The impact of the change on both safety inputs and anticipated operation is documented.

(2) Final Design That Requires a Loading Pattern Change

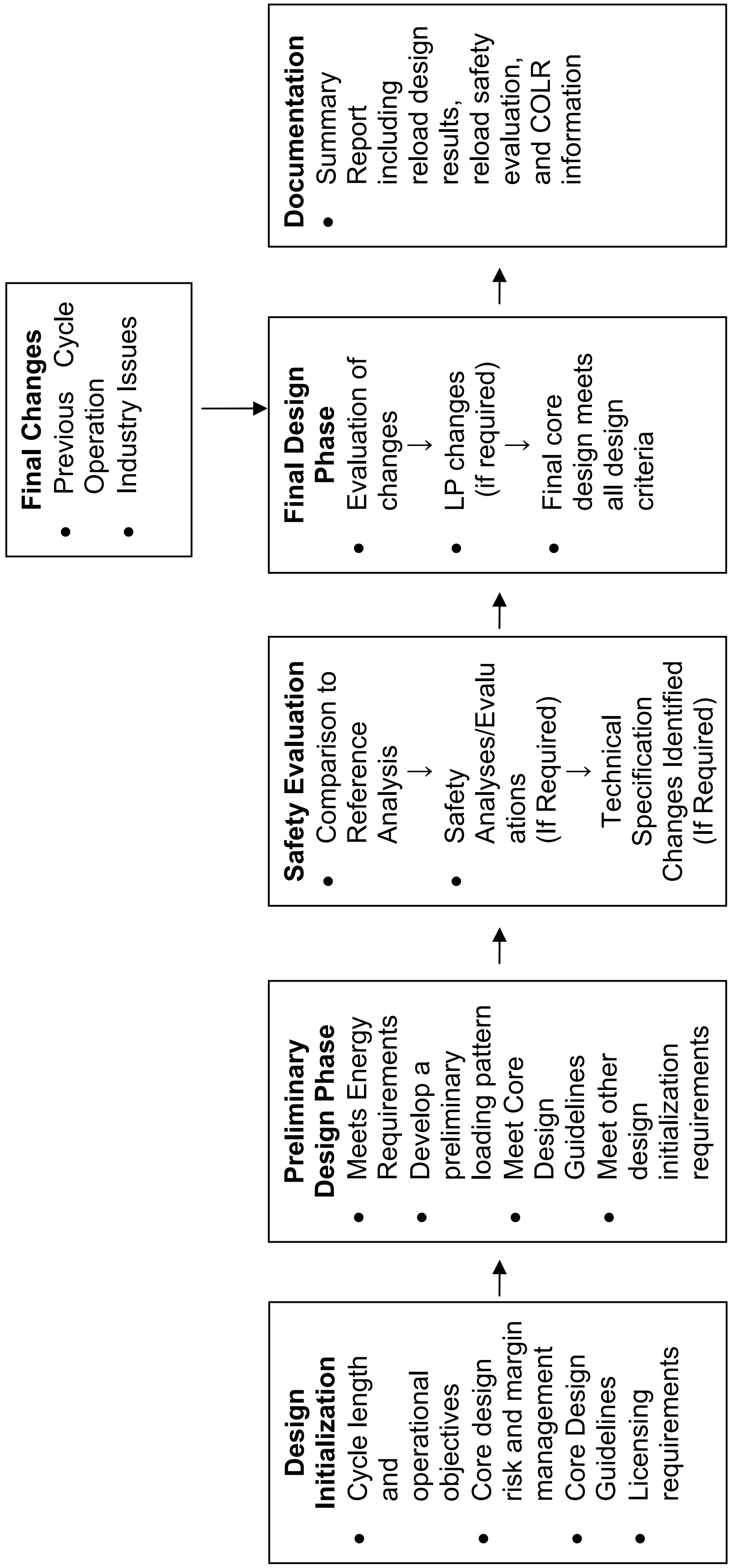
Some of loading pattern changes may require very little reanalysis, while others require a major revision to the safety evaluation. For example, replacing a small number of failed fuel assemblies with similar fuel available from a spent fuel inventory may have a negligible impact on safety input parameters. However, a large change in energy requirements that changes the number and distribution of feed assemblies in the core typically requires a much more extensive reanalysis. In most cases the actual inputs to the safety analyses remain unchanged due to the bounding nature of most inputs.

2.6 Documentation

Based on the safety analyses and evaluations performed for the reload, a summary report is provided by Mitsubishi to the licensee, which has the responsibility for plant safety and licensing with regulatory authorities. The report describes reload design results including reload safety evaluation.

The summary report (referred to as the reload evaluation report) is supplemented by any other technical reports provided to the licensee that are required to support or revise the plant licensing basis. The technical reports provide such materials as COLR information which supports the technical specification requirements, impact on the plant licensing basis due to plant changes planned to be concurrent with the reload, and other issues (such as potential unreviewed safety questions) identified as part of (or concurrent with) the reload design.

In accordance with 10CFR50.59, the licensee may be required to submit documentation to regulatory authorities for review in support of technical specification changes or to address an unreviewed safety question.



LP: Loading Pattern

Figure 2-1 Flow Chart of Typical Design Process

3.0 REFERENCE SAFETY EVALUATION

3.1 Introduction

This section describes key parameters concerning reload safety evaluation. The MHI safety analysis methodology is based upon the ‘bounding analysis’ concept. A set of key safety input parameters are generated to bound all future anticipated operating cycles. For each reload core, the key safety analysis inputs are re-evaluated, re-calculated, and compared to the bounding values used for the reference analyses.

The reference analyses will be provided in the plant-specific FSAR, which should be derived, and therefore, mostly equal to US-APWR DCD (Reference 3-1) Chapter 15 safety analyses. If the parameter is bounded by the DCD, the reference safety analysis is valid; otherwise, additional evaluation or analysis is required to demonstrate that safety criteria and design bases are met.

3.2 Accident Analysis Methods

During the core reload design process, several safety analysis parameters may need to be re-evaluated to ensure the safe operability of the plant. If any changes in parameters are outside the previous bounds, certain analyses will need to be re-evaluated.

Table 3-1 through Table 3-6 provide a useful matrix to quickly determine how changes in different parameters will affect the accident analyses for each event. In certain circumstances, it may be possible to document the sensitivity of the results of a specific change such that a reanalysis is not needed.

3.2.1 Nominal Plant Conditions

Two power ratings are considered:

- The design core thermal power output.
- The design nuclear steam supply system (NSSS) thermal power output, which includes the thermal power generated by the reactor coolant pumps (RCPs).

Most of the event analyses assume the NSSS thermal power output; while the remaining event analyses assume the design core thermal power.

Nominal control system characteristics, developed from operational experience, are modeled in the analyses as they are expected to perform.

Power operation with an inactive loop is not allowed by the technical specifications for the US-APWR; therefore, unless otherwise stated in the individual event analysis section, all RCPs are assumed to be operating at the time of event initiation.

3.2.2 Initial Conditions

Accident analyses are performed with conservative bases. Inputs to the analyses are developed to include allowances for operational uncertainties (such as allowances for instrument, measurement, and setpoint uncertainties) and analysis uncertainties (such as allowances for variations in core parameters and measurement uncertainties for certain basic data, such as delayed neutron fractions). In principle, conservative values are employed for each parameter that is input to calculations performed as part of the accident analyses. The uncertainty allowance for each parameter is applied in the direction that produces the most adverse calculated consequences for the transient.

Nominal conditions (power, temperature, and pressure) are assumed as the initial conditions for departure from nucleate boiling (DNB) limited events that are analyzed using the revised thermal design procedure (RTDP) (Reference 3-2). The RTDP accounts for uncertainty allowances by a statistical combination technique that is part of the procedure.

For certain other events, the initial conditions are obtained by adding the maximum steady-state errors to the rated values in the conservative direction. The conservatively assumed steady-state errors are as follows:

- Core power ±2 percent allowance for calorimetric error
- Average RCS temperature ±4°F allowance for controller deadband and measurement error
- Pressurizer pressure ±30 psi allowance for steady-state fluctuations and measurement errors

3.2.3 Power Distributions

The response of the reactor to a transient condition depends in part on the core power distribution at the beginning of the transient. The reactor core is designed to have a relatively uniform power distribution. This is accomplished by the arrangement of fuel assemblies, the location of rod cluster control assemblies (RCCAs), the grouping of specific RCCAs into banks, the selection of the sequence of RCCA withdrawal steps, and the initial design of the fuel assemblies, including fuel pellet enrichment, arrangement of gadolinia integral fuel rods within fuel assemblies, the type of burnable poison, and the number and location of burnable poison rods within the fuel assemblies.

Power distributions are characterized by the nuclear enthalpy rise hot channel factor ($F_{\Delta H}^{N}$), which is essentially a radial power peaking value; the heat flux hot channel factor (F_Q), which is a local ("point") power peaking value and the axial power distribution.

In the analyses, all transients are assumed to begin with the most severe power distributions that are consistent with operation within the technical specifications. Power peaking factors employed in the analyses are listed in conjunction with specific event analyses.

The value of $F_{\Delta H}^{N}$ increases as the power level decreases due to effects caused by the insertion of RCCAs. An increase in $F_{\Delta H}^{N}$ consistent with technical specification limits is factored into the over temperature ΔT trip limits.

Power increase transients that are relatively slow (e.g., a step increase in steam flow) may cause the reactor to establish a new steady state condition at higher power without causing a reactor trip.

Power increase transients that are relatively fast with respect to the fuel rod thermal time constant (e.g., uncontrolled control rod assembly withdrawal from a subcritical and rod ejection accidents) may cause a large rapid power increase that can challenge one or more fuel design limits. Detailed fuel transient heat transfer calculations are performed for this type of transient using bounding design power distributions, accident-specific power distributions, or in certain cases, time-dependent power distributions calculated during the accident. Power distribution assumptions are described as part of specific event analyses.

3.2.4 Reactivity Coefficients

The transient response of the reactor depends on the reactivity feedback effects, particularly the moderator density or temperature coefficient and the Doppler power or temperature coefficient. A bounding maximum or minimum value of the reactivity coefficients is used in each event analysis. In many cases, the conservative combination of moderator density and Doppler power coefficients represents either beginning or end of core fuel cycle conditions. In certain cases, conservative combinations of parameters that are not representative of realistic situations are used in order to bound their combined effects at all times during the fuel cycle. For most accidents analyzed using the MARVEL-M code (described in Section 3.2.9.1), one of two constant values of the moderator density coefficient is used - a minimum value and a maximum value. The justification for the use of specific values of these coefficients is described on a case-by-case basis in the respective analysis.

3.2.5 RCCA Trip Insertion Characteristics

A reactor trip signal causes all of the RCCAs to be inserted by gravity to the bottom of the active fuel region. In the analyses, the single highest-reactivity-worth RCCA is conservatively assumed to fail to insert (i.e., to remain fully withdrawn).

Figure 15.0-3 in Reference 3-1 is a plot displaying the conservative RCCA displacement as a function of time that is used in the analyses for the RCCA insertion following a reactor trip until RCCAs are fully inserted.

Figure 15.0-4 in Reference 3-1 shows the negative reactivity addition as a function of time that is used in the analysis for the RCCA insertion following a reactor trip. This curve is based on: (1) the same conservatively slow RCCA insertion rate discussed in the preceding paragraph and (2) a conservative bottom-skewed axial power distribution within the core.

This RCCA negative reactivity insertion versus time is input into the computer codes used in the analyses.

3.2.6 Residual Decay Heat

Residual heat in a subcritical core, including decay heat from fission products and actinides, is calculated for the large break LOCA and the non-LOCA transient in accordance with the methodology of ANSI/ANS-5.1-1979 (Reference 3-3).

For the small break LOCA and post-LOCA long-term cooling analysis, the decay heat from fission products is assumed to be equal to 1.2 times the values for infinite operating time in the ANS standard 5.1-1971 (Reference 3-4), conforming to the requirement of 10 CFR 50 Appendix K (Reference 3-5). The heat from the decay of actinides is calculated in accordance with the methodology of ANSI/ANS-5.1-1979.

Input parameters used with ANSI/ANS-5.1-1979 are selected so as to envelope conceivable core conditions for the US-APWR.

3.2.7 Core Protection Limits

The reactor trip system (RTS) initiates signals to open the reactor trip breakers when operating parameters that are monitored by the protection and safety monitoring system (PSMS) approach pre-determined limits. This action removes power to the control rod drive mechanism (CRDM) coils, permitting the rods to fall by gravity into the core.

Instrumentation system time delays are associated with each of the PSMS trip functions. These include delays in signal generation, opening the reactor trip breakers, and the release of the rods by the CRDM. The total response time delay for a reactor trip is the interval from the time the operating parameter reaches the analytical limit to the time the control rods are released and start to drop into the core. The delay for each trip signal is selected to give conservative analysis results.

The difference between the trip analytical limit and the nominal trip setpoint specified in the plant technical specifications includes allowances for instrumentation channel error and setpoint error. The availability and range of each type of instrumentation are consistent with the corresponding predicted parameter values in the specific event analyses. Instrumentation is provided to monitor key plant parameters during events (e.g., EFW flow indication in the control room).

3.2.8 Core Radiological Source Term

Certain accidents can result in the release of primary and secondary coolant to the environment. The technical specifications limit the iodine and noble gas radioactivity in the primary and secondary coolant. Fission products can be present in the primary coolant due to leaking fuel rods (i.e., rods that are leaking prior to the postulated accident). Activity in the primary coolant is assumed to be transferred to the secondary coolant due to leaking steam generator tubes. The quantity of coolant released to the environment during postulated accident sequences is described in the event-specific discussion in the US-APWR DCD (Reference 3-1) for those accidents that have radiological consequences due to coolant releases.

The cladding of previously non-leaking fuel rods can become damaged during certain non-LOCA accidents involving fuel in the reactor core. This breached cladding can release fission products in the gap between the fuel pellet and the cladding of the fuel rod. This fuel rod gap inventory can be transferred to the reactor (primary) coolant, then be transported to the secondary coolant via postulated leaking or failed steam generator tubes, and then be released to the environment when the steam line safety valves or relief valves are assumed to be used to dissipate decay heat by releasing steam to the environment.

For some events, the iodine concentrations in the reactor coolant are calculated using special assumptions that ensure the calculations account for conservatively large quantities of radioactive iodine by assuming: (1) a pre-transient iodine spike and (2) a transient initiated iodine spike.

For the pre-transient iodine spike, a reactor transient is assumed to have occurred prior to the initiating event of the transient and has raised the primary coolant iodine concentration from the maximum value permitted by the technical specifications to a higher specified value.

For the transient initiated iodine spike, the transient itself is assumed to cause an iodine spike in the primary system. The increase in primary coolant iodine concentration is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant (expressed in curies per unit time) increases to a specified value greater than the release rate corresponding to the iodine concentration at the equilibrium value specified in the technical specifications.

The pre-accident noble gas concentrations in the primary coolant are also based on the technical specification limit.

3.2.9 Accident Analysis Codes and Evaluation Models

The fourth row of Table 3-1 through Table 3-6 are includes a summary listing of the computer codes used for analyzing specific events in the US-APWR DCD (Reference 3-1). Additional information about these computer codes is provided in the following sections. Any specialized modeling capabilities that are unique to a specific given event are summarized in the respective event analysis section.

3.2.9.1 MARVEL-M

MARVEL-M (Reference 3-6) is a multi-loop plant system transient analysis code used to calculate detailed transient behavior of pressurized water reactor systems. MARVEL-M has a maximum modeling capability of four coolant loops with four steam generators and associated systems. It simulates reactor kinetics, thermal-hydraulics of the core and reactor coolant system, the pressurizer, main and secondary steam and feedwater systems, and the reactor control and protection system. It also simulates the ESF and other subsystems, which are representative of conventional PWR power plants.

The MARVEL-M program utilizes a space-independent single point reactor kinetics model with six delayed neutron groups. The thermal and hydraulic characteristics of the reactor coolant system are described by time- and space-dependent differential equations. The reactor coolant system is represented by flow nodes, which model transient behaviors of mass and energy for the ranges of sub-cooled and homogenous two phase fluid typically encountered in the analysis of non-LOCA transients. Pressurizer heaters, spray, and safety valves are also considered in the program. Reactivity effects from the moderator, fuel, boron, and rods are also included. MARVEL-M also simulates the protection and monitoring system and control systems.

MARVEL-M has the 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 selected DNBR limits represented by operating parameters of core inlet temperature, pressure and power levels. The simplified DNBR model closely agrees with design calculations when the core operating conditions do not exceed the design power distribution or core protection limits. When conditions exceed these limitations, DNBR analysis is performed by the more detailed external calculation code, VIPRE-01M, which is discussed in the next section.

MARVEL-M outputs the transient response of reactor power, reactor pressure, primary coolant temperature, DNBR, and other parameters. Code inputs include initial conditions such as primary coolant temperature and the reactor power, primary coolant volume and other plant data, nuclear characteristics data, and setpoints for actuation of the reactor trip system and ESF systems. The program is applicable to conventional as well as advanced PWR plants (APWRs).

3.2.9.2 VIPRE-01M

VIPRE-01M (Reference 3-7) is a subchannel thermal hydraulic analysis code with both steady state and transient capabilities, including a fuel thermal transient model. It divides the core into three-dimensional mesh elements and then solves the appropriate equations by applying the mass, momentum, and energy conservation principles to each mesh element.

Inputs into VIPRE-01M include initial conditions such as reactor power, coolant temperature, coolant flow, power distributions, core geometry and fuel properties.

VIPRE-01M calculates time-dependent changes in parameters, such as coolant temperature, coolant density, void fraction, fuel temperature, and minimum DNBR in the core. Boundary conditions are the transient data generated by MARVEL-M or TWINKLE-M.

3.2.9.3 TWINKLE-M

TWINKLE-M (Reference 3-6) is a multidimensional spatial neutron kinetics code which solves the two-group transient diffusion equations using a finite difference technique. This code is used to analyze changes in dynamic behavior of space- and time-dependent neutron flux in response to reactivity accidents.

TWINKLE-M also uses a six-region model for fuel rod heat transfer between the fuel pellet, clad, and primary coolant and a primary coolant thermal hydraulic model, which handles behavior in the vertical-axis using the mesh points used for the dynamic analysis of the neutron flux. This capability enables feedback effects, including Doppler and moderator feedback effects, to be modeled as space-dependent. The feedback effects are taken into account by absorption cross-section compensation at each mesh point.

Inputs into TWINKLE-M include time-dependent changes at each mesh point of the neutron cross sections, core inlet temperature, pressure, flow rate though the core, boron concentration, and control rod motion. Outputs include the neutron flux level, neutron flux distribution, and the thermal response of the core as space- and time-dependent parameters.

3.2.9.4 RADTRAD

The RADionuclide TRansport, Removal, and Dose (RADTRAD) (Reference 3-8) computer code is a computer model for estimating doses at offsite locations such as the EAB and the LPZ, as well as onsite locations (e.g., main control room) due to postulated radioactivity releases from design basis accident conditions. RADTRAD uses a compartment model and simulates radioactive material transport through the containment, and related systems, structures and components. RADTRAD calculates dose consequences for different specified time intervals based on user-input information on the amount, form, and species of the radioactive material released in the reactor plant.

3.2.9.5 ANC

The nuclear analysis code ANC is described in Section 4.2.

3.2.9.6 WCOBRA/TRAC (M1.0)

WCOBRA/TRAC (M1.0), a modified version of the WCOBRA/TRAC, is used for calculation of thermal-hydraulic behavior during a large break LOCA. Its applicability to the US-APWR large break LOCA analysis is discussed in the Topical Report (Reference 3-9).

WCOBRA/TRAC is approved by the U.S. Nuclear Regulatory Commission for use in best estimate large break LOCA calculations for three and four loop conventional PWRs, also the AP600 and AP1000 advanced plant designs. The COBRA portion of the code is based on a two-fluid, three-field, multi-dimensional fluid equations to describe thermal-hydraulic behavior of the vessel component. The TRAC portion of the code is based on one-dimensional, two-phase drift flux model to describe thermal-hydraulic behavior of the major components of PWR, such as steam generators, pipes, pumps, valves and pressurizer.

3.2.9.7 HOTSPOT

HOTSPOT (Reference 3-9) is used for detailed fuel rod model analysis.

HOTSPOT calculates the effect of uncertainties at axial locations of the fuel rod. The code uses a transient conduction model identical to that of the WCOBRA/TRAC computer code. The code also simulates cladding burst, metal-water reaction and fuel relocation following cladding burst phenomena.

3.3 Events Description and Key Parameters

This section provides an event description and key parameters based on the reference analyses in the US-APWR DCD (Reference 3-1). In Table 3-1 through Table 3-6, the column on the left is a list of all key parameters, both generic and event-specific, that are used throughout all the event analyses. A circle is present that indicates that the parameter in the left column is a key parameter for the associated event described in the top rows of the table. Included in these rows of event descriptions are the associated section in the US-APWR DCD, an abbreviation for the event name, the event classification (AOO or PA), the computer code(s) used in the analyses, and an analysis case description. Certain parameters only apply to specific events. Event specific criteria are presented separately at the bottom of the table. Special considerations for calculating these parameters, if any, are described in Section 4, nuclear design, or Section 5, thermal-hydraulic design, of this report.

Thus, if a fuel reload will affect a certain parameter, Table 3-1 through Table 3-6 can be used to determine which analyses need to be re-performed or re-evaluated, and for which events.

3.3.1 Decrease in Feedwater Temperature

3.3.1.1 Event Description

A decrease in feedwater temperature causes a reduction in steam generator secondary temperature, resulting in an increase in primary-to-secondary heat transfer. In the presence of a negative moderator temperature coefficient (positive moderator density coefficient), the decrease in primary temperature (and associated increase in density) results in a positive reactivity insertion and core power increase. Core conditions that could approach the core DNBR or over power limits are protected by the over temperature and over power ΔT and high power range neutron flux (high setpoint) reactor trips. If there is no protection system action, the reactor reaches a new equilibrium condition at a power level corresponding to the steam generator heat removal rate, characterized by the primary-to-secondary ΔT.
A feedwater temperature decrease can be caused by the functional loss of either the high-pressure or low-pressure feedwater heaters.

This event is classified as an AOO. Acceptance criteria, evaluation model(s) and the results of safety analysis based on a bounding method are described in Section 15.1.1 of Reference 3-1.

3.3.1.2 Key Parameters

Generic key parameters for the transient of DNBR, fuel temperature and RCS pressure in this event are as follows.

Detailed information for a set of parameters (maximum and/or minimum values) and other parameters concerning the safety analysis are shown in Table 3-1 with computer codes used and analysis cases.
There are no additional event-specific parameters for this event.

3.3.2 Increase in Feedwater Flow

3.3.2.1 Event Description

An increase in the feedwater flow rate to the secondary side of the steam generator will increase the heat transfer from the primary to the secondary side of the steam generator. This will cause a reduction in the reactor coolant temperature at the reactor vessel inlet. In the presence of a negative moderator temperature coefficient (positive moderator density coefficient), the decrease in primary temperature (and associated increase in density) results in a positive reactivity insertion and core power increase.
This transient is caused by a main feedwater regulation valve that is opened fully due to an operator error or a malfunction of the feedwater control system during rated power or part load operation. The feedwater control system is designed such that a single control system failure will affect only one main feedwater regulation valve, and hence, one steam generator.

This event is classified as an AOO. Acceptance criteria, evaluation model(s) and the results of safety analysis based on a bounding method are described in Section 15.1.2 of Reference 3-1.

3.3.2.2 Key Parameters

Generic key parameters and event-specific parameters are shown in Section 3.3.1.2 plus following parameter.

3.3.3 Increase in Steam Flow

3.3.3.1 Event Description

A rapid increase in steam flow can cause a temporary mismatch between the power produced by the reactor core and the power demanded by the steam generators. This situation can reduce the temperature of the coolant re-entering the reactor vessel, which, in turn, can lead to an increase in reactor power.
The reactor control system is designed to follow a 10% step load change or a 5% per minute ramp load increase. The analysis of the event documented in this section confirms that core, system, and boundary performance is acceptable following the more severe of these two conditions – the 10% step load change – without reactor trip system action.
A rapid increase in steam flow can be caused by (1) an administrative or operator error resulting in an excessive load increase during power operations or (2) an operator error or equipment malfunction that causes a turbine bypass valve, main turbine control valve, or main steam relief valve to inadvertently fully open.

This event is classified as an AOO. Acceptance criteria, evaluation model(s) and the results of safety analysis based on a bounding method are described in Section 15.1.3 of Reference 3-1.

3.3.3.2 Key Parameters

Generic key parameters and event-specific parameters are shown in Section 3.3.1.2 plus following parameter.

3.3.4 Inadvertent Opening of a Steam Generator Relief or Safety Valve

3.3.4.1 Event Description

The inadvertent opening of a main steam relief valve, main steam safety valve, or turbine bypass valve can cause a rapid increase in steam flow and a depressurization of the secondary system. The steam release removes energy from the reactor coolant system (RCS), which causes a reduction in the reactor coolant temperature and pressure. In the presence of a negative moderator temperature coefficient (positive moderator density coefficient), the decrease in primary temperature (and associated increase in density) results in a positive reactivity insertion and core power increase. A return to criticality and power is possible if the event is initiated from hot standby. This event is caused by a malfunction that inadvertently opens a single main steam relief, main steam safety valve or turbine bypass valve, resulting in the uncontrolled release of steam from the main steam system.

This event is classified as an AOO. Acceptance criteria, evaluation model(s) and the results of safety analysis based on a bounding method are described in Section 15.1.4 of Reference 3-1.

3.3.4.2 Key Parameters

Generic key parameters for the transient of DNBR, fuel temperature and RCS pressure in this event are as follows.

Event-specific key parameters in this event are as follows.

The following state point data at the system transient of reference analysis are used.

Detailed information for a set of parameters (maximum and/or minimum values) and other parameters concerning the safety analysis are shown in Table 3-1 with computer codes used and analysis cases.

3.3.5 Steam System Piping Failures

3.3.5.1 Event Description

The increase in steam generation rate caused by the postulated steam system piping failure removes heat from the reactor coolant system (RCS), which, in turn, lowers the temperature and pressure of the reactor coolant. In the presence of a negative moderator temperature coefficient (positive moderator density coefficient), the decrease in primary temperature (and associated increase in density) results in a positive reactivity insertion and core power increase.

Depending on the combination of break location and single failure assumed, the steam flow may be non-uniform (from only one steam generator) or uniform (all steam generators contribute to the break flow), and may either be automatically isolated or result in an uncontrolled blowdown from one steam generator. Because the emergency core cooling system (ECCS) is actuated for this event, the availability of offsite power is also addressed.
The approach used in the analysis is to define a bounding case that envelopes the various assumptions so that each combination does not require a separate analysis.

A spectrum of steam system piping failure sizes and locations from both power operation and hot zero power initial conditions are considered. The full double-ended failure of a main steam system pipe is classified as a PA and failure of a minor steam system pipe is classified as an AOO. Acceptance criteria, evaluation model(s) and the results of safety analysis based on a bounding method are described in Section 15.1.5 of Reference 3-1.

3.3.5.2 Key Parameters

In the power operation case, generic key parameters for the transient of DNBR, fuel temperature and RCS pressure are as follows.

Detailed information for a set of parameters (maximum and/or minimum values) and other parameters concerning the safety analysis are shown in Table 3-1 with computer codes used and analysis cases.
There are no additional event-specific parameters for this case.

In the hot zero power case, generic key parameters and event-specific parameters are the same as Section 3.3.4.2.

3.3.6 Loss of External Load / Turbine Trip

3.3.6.1 Event Description

In a loss of external load event, an electrical disturbance causes loss of a significant portion of the generator load. A loss of external electrical load event may result from an abnormal grid frequency, a trip of the generator and or turbine trip, or spurious closure of the main turbine stop or control valves or main steamline isolation valves. The loss of load event is different from the loss of alternating current (ac) power event discussed in Section 3.3.7 in that offsite ac power remains available for this event to operate the station auxiliaries. Therefore, the onsite safety grade gas turbine generators (GTGs) are not required for this event.

This event is classified as an AOO. Acceptance criteria, evaluation model(s) and the results of safety analysis based on a bounding method are described in Section 15.2.1 and 15.2.2 of Reference 3-1.

3.3.6.2 Key Parameters

Generic key parameters for the transient of DNBR, fuel temperature and RCS pressure in this event are as follows.

Detailed information for a set of parameters (maximum and/or minimum values) and other parameters concerning the safety analysis are shown in Table 3-2 with computer codes used and analysis cases.
There are no additional event-specific parameters for this event.

3.3.7 Loss of Non-Emergency AC Power to the Station Auxiliaries

3.3.7.1 Event Description

The loss of non-emergency alternating current (ac) power is assumed to result in the loss of all power to the station auxiliaries. The causes are a complete loss of the external (offsite) grid accompanied by a turbine-generator trip or loss of the onsite ac distribution system.

This event is classified as an AOO. Acceptance criteria, evaluation model(s) and the results of safety analysis based on a bounding method are described in Section 15.2.6 of Reference 3-1.

3.3.7.2 Key Parameters

Generic key parameters and event-specific parameters are the same as Section 3.3.6.2 except a hot channel factor.

3.3.8 Loss of Normal Feedwater Flow

3.3.8.1 Event Description

A loss of normal feedwater flow could occur from pump failures, valve malfunctions, or a loss of offsite power. The loss of feedwater flow results in a reduction of the secondary system's ability to remove heat generated by the reactor core. As a result, the reactor coolant temperature and pressure increase and will eventually require a reactor trip to protect the fuel and reactor coolant pressure boundary.

This event is classified as an AOO. Acceptance criteria, evaluation model(s) and the results of safety analysis based on a bounding method are described in Section 15.2.7 of Reference 3-1.

3.3.8.2 Key Parameters

Generic key parameters and event-specific parameters are the same as Section 3.3.6.2.

3.3.9 Feedwater System Pipe Break

3.3.9.1 Event Description

The feedwater system pipe break is a non-uniform transient that involves modeling the flow from one of the secondary loops. The feedwater system pipe break causes a loss of inventory from the saturated liquid mass in the steam generator resulting in RCS heat-up and pressurization. Unless the heat-up of the RCS is mitigated, there will be a possibility of water relief through the pressurizer safety valve.

The most limiting feedwater line rupture is a double-ended rupture of the largest feedwater line. A double-ended rupture of the feedwater piping between the main feedwater check valve and steam generator bounds the remaining large break cases. Smaller breaks will result in a slower reduction in steam generator heat removal capability.

This double-ended break of a main feedwater system pipe is classified as a PA. Acceptance criteria, evaluation model(s) and the results of safety analysis based on a bounding method are described in Section 15.2.8 of Reference 3-1.

3.3.9.2 Key Parameters

Generic key parameters and event-specific parameters are the same as Section 3.3.6.2 except a peaking factor.

3.3.10 Partial Loss of Forced Reactor Coolant Flow

3.3.10.1 Event Description

Loss of forced reactor coolant flow events can result from a mechanical or electrical failure in one or more reactor coolant pumps or from a fault in the power supply to the pump motor. A partial loss of forced reactor coolant flow accident results from a simultaneous loss of electrical supply to one or more of the four reactor coolant pump motors. 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 a decrease in minimum DNBR. This transient is terminated by the low reactor coolant flow trip, which prevents DNB occurrence.

This event is classified as an AOO. Acceptance criteria, evaluation model(s) and the results of safety analysis based on a bounding method are described in Section 15.3.1 of Reference 3-1.

3.3.10.2 Key Parameters

Generic key parameters for the transient of DNBR, fuel temperature and RCS pressure in this event are as follows.

Detailed information for a set of parameters (maximum and/or minimum values) and other parameters concerning the safety analysis are shown in Table 3-3 with computer codes used and analysis cases.
There are no additional event-specific parameters for this event.

3.3.11 Complete Loss of Forced Reactor Coolant Flow / Frequency Decay

3.3.11.1 Event Description

The complete loss of forced reactor coolant flow is initiated by malfunctions that cause the loss of electrical power or the decrease of offsite power frequency to all four reactor coolant pumps during power operation, resulting in a reduction in the core cooling capabilities. If the reactor is at power at the time of the transient, the immediate effect of a complete loss of coolant flow is a rapid increase in coolant temperature and decrease in minimum DNBR. This transient is terminated by the low reactor coolant pump speed trip, which prevents DNB occurrence.

This event is classified as an AOO. Acceptance criteria, evaluation model(s) and the results of safety analysis based on a bounding method are described in Section 15.3.1 of Reference 3-1.

3.3.11.2 Key Parameters

Generic key parameters and event-specific parameters are the same as Section 3.3.10.2.

3.3.12 Reactor Coolant Pump Rotor Seizure / Shaft Break

3.3.12.1 Event Description

Reactor coolant pump rotor seizure is initiated by the instantaneous seizure of one reactor coolant pump rotor during power operation. This postulated rotor seizure would cause a rapid reduction in the reactor coolant flow resulting in a decrease in core cooling capacity. This could, in turn, lead to an increase in the reactor fuel temperature, primary coolant temperature, and reactor pressure.

Reactor coolant pump shaft break is initiated by the instantaneous break (failure, or fracture and separation) of one of the reactor coolant pump shafts during power operation. This postulated shaft break would cause a reduction in the reactor coolant flow and decrease the core cooling capacity. This could, in turn, lead to an increase in the reactor fuel temperature, primary coolant temperature, and reactor pressure.

These events are classified as a PA. Acceptance criteria, evaluation model(s) and the results of safety analysis based on a bounding method are described in Section 15.3.3 and 15.3.4 of Reference 3-1.

3.3.12.2 Key Parameters

Generic key parameters are shown in Section 3.3.10.2 plus following parameter.

3.3.13 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical

3.3.13.1 Event Description

An RCCA withdrawal incident is an uncontrolled addition of reactivity to the reactor core caused by the withdrawal of RCCA banks, which results in a power increase. The occurrence of such a transient can be caused by a malfunction of the reactor control system or the control rod drive system. This incident could occur with the reactor subcritical, at hot zero power, or at power. In this section, the transient event analyzed is for the reactor at hot zero power.

This event is classified as an AOO. Acceptance criteria, evaluation model(s) and the results of safety analysis based on a bounding method are described in Section 15.4.1 of Reference 3-1.

3.3.13.2 Key Parameters

Generic key parameters for the transient of DNBR, fuel temperature and RCS pressure in this event are as follows.

Event-specific key parameters in this event are as follows.

Detailed information for a set of parameters (maximum and/or minimum values) and other parameters concerning the safety analysis are shown in Table 3-4 with computer codes used and analysis cases.

3.3.14 Uncontrolled Control Rod Assembly Withdrawal at Power

3.3.14.1 Event Description

The uncontrolled RCCA bank withdrawal at power is caused by a control system or rod control system failure that causes a bank withdrawal to occur. The event may occur from various initial power levels and the positive reactivity insertion can vary from zero to the maximum value depending on RCCA withdrawal rate and assumed maximum bank worth.
An uncontrolled RCCA bank withdrawal at power results in an increase in core heat flux. Since the heat extracted from the steam generator lags behind the core power until the steam generator pressure reaches the main steam safety valve setpoint, the reactor coolant temperature tends to increase. Without a manual or automatic reactor trip, the power mismatch and the rise of reactor coolant temperature could eventually result in DNB.

This event is classified as an AOO. Acceptance criteria, evaluation model(s) and the results of safety analysis based on a bounding method are described in Section 15.4.2 of Reference 3-1.

3.3.14.2 Key Parameters

Generic key parameters for the transient of DNBR, fuel temperature and RCS pressure in this event are as follows.

Event-specific key parameters in this event are as follows.

Detailed information for a set of parameters (maximum and/or minimum values) and other parameters concerning the safety analysis are shown in Table 3-1 with computer codes used and analysis cases.

3.3.15 Control Rod Misoperation (System Malfunction or Operator Error)

3.3.15.1 Event Description

Control rod misoperation includes:

- One or more dropped RCCAs within a group or bank
- One or more misaligned RCCAs (relative to their bank)
- Uncontrolled withdrawal of a single RCCA

Dropped or misaligned RCCAs could be caused by failures or malfunctions of an RCCA drive mechanism or RCCA drive mechanism control equipment. Movement of a single RCCA is never performed during normal operations. However, the capability to move a single RCCA exists in order to restore a dropped RCCA to its correct position under strict administrative procedural control. Each control bank RCCA is assigned to a bank, and all RCCAs in that bank are moved together in a pre-selected sequence such that a single failure cannot cause a single RCCA to withdraw. Given the design of the rod control system, assignment of RCCAs to groups, and strict administrative controls in place for restoration of a single RCCA to its proper insertion step, a single RCCA withdrawal can only be caused by multiple operator errors.

If the dropped RCCAs are not detected and corrective action taken, other RCCAs could be withdrawn to compensate for the reactivity decrease caused by the dropped RCCAs (to restore reactor power and/or average coolant temperature to match the turbine demand). The dropped rod event is therefore analyzed assuming automatic rod control. If RCCAs were withdrawn, the reactor power is restored which results in the increase of the hot channel heat flux relative to the hot channel heat flux prior to the dropped rod. This increase in hot channel heat flux could lead to a reduction in the reactor safety margin.

RCCA misalignment could occur if a fault in the control system causes a single RCCA or the RCCAs in a bank to be moved out of sequence from the pre-programmed sequence. (e.g., not all of the RCCAs in a group move at the same speed or the two groups in a bank do not move at the same speed). If misaligned RCCAs are not detected and corrective action taken, the core power distribution could exceed the design power distribution, resulting in a reduction of margin to the fuel design limits.

A single RCCA withdrawal can result in a core power response similar to a bank withdrawal at power with a concurrent adverse change in power distribution that can exceed the design power distribution shape and peaking factors, resulting in violating fuel design limits before the event is terminated by the reactor trip system.

The dropped and misaligned RCCA misoperation events are classified as AOOs and the uncontrolled withdrawal of a single RCCA is classified by MHI as a PA. Acceptance criteria, evaluation model(s) and the results of safety analysis based on a bounding method are described in Section 15.4.3 of Reference 3-1.

3.3.15.2 Key Parameters

Generic key parameters for the transient of DNBR, fuel temperature and RCS pressure in this event are as follows.

Event-specific key parameters in this event are as follows.

Detailed information for a set of parameters (maximum and/or minimum values) and other parameters concerning the safety analysis are shown in Table 3-4 with computer codes used and analysis cases.

3.3.16 Inadvertent Decrease in Boron Concentration in the RCS

3.3.16.1 Event Description

An inadvertent decrease of the boron concentration in the reactor coolant can occur due to the addition of low-boron-concentration water into the reactor coolant due to a malfunction or improper operation of the chemical and volume control system (CVCS). This transient results in a positive reactivity addition to the core.

This event is classified as an AOO. Acceptance criteria, evaluation model(s) and the results of safety analysis based on a bounding method are described in Section 15.4.6 of Reference 3-1.

3.3.16.2 Key Parameters

Event-specific key parameters in this event are as follows.

3.3.17 Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position

3.3.17.1 Event Description

This section addresses an event in which a fuel assembly or incore control component (ICCC; i.e., burnable absorbers, source assemblies, etc.) is loaded into an incorrect position in the core. This event is highly unlikely to occur because fuel assembly loading is strictly managed by administrative procedures. The fuel assemblies and ICCCs are marked with identification numbers, and are loaded into the core in accordance with a core loading diagram, which is checked by multiple designated operators. Neutron detectors provide continuous indication of the neutron flux as each fuel assembly is placed into the core. After the fuel loading is completed, the identification numbers are checked again to assure that all the fuel assemblies are loaded into the correct position. An in-core power distribution measurement is then performed during Low Power Testing and/or Power Ascension Testing. Therefore, if an incorrect fuel assembly loading were to occur, it would likely be detected before the core reaches high power, thereby avoiding the possibility of exceeding fuel failure limits.

This event is classified by MHI as a PA. Acceptance criteria, evaluation model(s) and the results of safety analysis based on a bounding method are described in Section 15.4.7 of Reference 3-1.

3.3.17.2 Key Parameters

There are no key parameters for this event.

3.3.18 Spectrum of Rod Ejection Accidents

3.3.18.1 Event Description

This accident is defined as the mechanical failure of a control rod drive mechanism (CRDM) housing, which results in the ejection of an RCCA and its 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, and depressurization of the RCS in the long-term period due to the CRDM housing break. The nuclear excursion is terminated by Doppler reactivity feedback from increased fuel temperature, and the core is shut down by the reactor trip.

This event is classified as a PA. Acceptance criteria, evaluation model(s) and the results of safety analysis based on a bounding method are described in Section 15.4.8 of Reference 3-1.

3.3.18.2 Key Parameters

Generic key parameters for the transient of DNBR, fuel temperature and RCS pressure in this event are as follows.

Event-specific key parameters in this event are as follows.

Detailed information for a set of parameters (maximum and/or minimum values) and other parameters concerning the safety analysis are shown in Table 3-5 with computer codes used and analysis cases.

3.3.19 CVCS Malfunction that Increases Reactor Coolant Inventory

3.3.19.1 Event Description

A CVCS malfunction that increases RCS inventory can be caused by an operator error, a test sequence error, or an electrical malfunction. The CVCS normally operates with one charging pump running and a constant letdown flow through the letdown path. The increase of RCS inventory may be caused by an increase in charging flow with letdown operating or by isolation of the letdown path (letdown line and excess letdown line). If the CVCS boron concentration is larger than the RCS boron concentration, the reactor may experience a negative reactivity insertion resulting in a decrease in reactor power and subsequent coolant shrinkage.

This event is classified as an AOO. Acceptance criteria, evaluation model(s) and the results of safety analysis based on a bounding method are described in Section 15.5.2 of Reference 3-1.

3.3.19.2 Key Parameters

Generic key parameters for the transient of DNBR, fuel temperature and RCS pressure in this event are as follows.

Detailed information for a set of parameters (maximum and/or minimum values) and other parameters concerning the safety analysis are shown in Table 3-6 with computer codes used and analysis cases.

There are no additional event-specific parameters for this event.

3.3.20 Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve

3.3.20.1 Event Description

An accidental depressurization of the reactor coolant system (RCS) could occur by the inadvertent opening of a pressurizer pressure relief valve. The causes could be a spurious electrical signal or an operator error. In the US-APWR, there are spring loaded safety relief valves (SRVs), motor operated safety depressurization valves (SDVs) and a motor operated depressurization valve (DV) used for the mitigation of severe accidents. A DV has more relief capacity than a SRV or a SDV, and will result in a more rapid depressurization upon opening. Therefore, the most severe core conditions for this event result from an inadvertent opening of a DV.

This event is classified as an AOO. Acceptance criteria, evaluation model(s) and the results of safety analysis based on a bounding method are described in Section 15.6.1 of Reference 3-1.

3.3.20.2 Key Parameters

Generic key parameters and event-specific parameters are shown in Section 3.3.19.2 plus following parameters.

3.3.21 Steam Generator Tube Rupture

3.3.21.1 Event Description

In the steam generator tube rupture (SGTR) event, the complete severance of a single steam generator tube is assumed. The event is assumed to take place at full power with the reactor coolant contaminated with fission products corresponding to continuous operation with a limited number of defective fuel rods. The event leads to leakage of radioactive coolant from the RCS to the secondary system. In the event of a coincident loss of offsite power or the failure of the steam dump system, atmospheric discharge of radioactivity can take place via the main steam relief valves (MSRVs), or the main steam safety valves (MSSVs), if the setpoint is reached. The assumption of a complete tube severance is considered conservative because the tube material Alloy 690 is a corrosion resistant and ductile material. Since the radioactivity in the secondary system is under continual surveillance, an accumulation of activity as a result of such minor leaks that exceeds the limits established in technical specifications is not permitted during operation.

The operator is expected to recognize the occurrence of a SGTR event, to identify and isolate the ruptured steam generator, and to take appropriate actions to stabilize the plant. These operator actions should be performed in a timely manner to minimize contamination of the secondary system and the release of radioactivity to the atmosphere. In addition, recovery procedures should be carried out on a time scale that ensures that the break flow to the secondary system is terminated before the water level in the ruptured steam generator reaches the steam generator outlet nozzle.

This event is classified as a PA. Acceptance criteria, evaluation model(s) and the results of safety analysis based on a bounding method are described in Section 15.6.3 of Reference 3-1.

3.3.21.2 Key Parameters

Generic key parameters and event-specific parameters are the same as Section 3.3.19.2.

3.3.22 Loss-of-Coolant Accidents

3.3.22.1 Event Description

A loss-of-coolant accident (LOCA) is defined as a break of the reactor coolant system piping or of any line connected to the system. Breaks of small cross section will cause a discharge of the coolant at a rate which can be accommodated by the charging pumps which would maintain an operational water level.

When a larger break occurs, depressurization of the reactor coolant system causes a flow of coolant to the reactor coolant system from the pressurizer resulting in pressure and level decrease in the pressurizer. When the pressurizer low pressure trip set point is reached, reactor trip occurs. The safety injection system is actuated when the appropriate set point is reached. The consequences of the accident are limited in two ways.

1. Reactor trip and borated water injection complement void formation in causing rapid reduction of nuclear power to a residual level corresponding to the delayed fission and fission product decay.
2. Injection of borated water ensures sufficient flooding of the core to prevent excessive clad temperature increase.

Before the break occurs, the plant is in a steady state condition, i.e. the heat generated in the core is being removed through the secondary system. During blowdown, heat from decay, hot internals and the vessel continues to be transferred to the RCS. The heat transfer between the reactor coolant system and the secondary system may be in either direction depending on the relative temperatures.

When the RCS depressurizes to 600 psia, the cold leg accumulators begin to injection water into the reactor coolant loops. The blowdown phase of the transient ends when the RCS pressure decreases to a pressure of the containment atmosphere.

At this time, refill of the reactor vessel begins, and continues until emergency core cooling water has filled the lower plenum of the reactor vessel.

The reflood phase of the transient is defined as the time period lasting from the end of refill until the reactor vessel has been filled with water to the extent that the core temperature rise has been terminated.

This accident is classified as a PA, and the analysis is performed to demonstrate compliance with the criteria of 10CFR50.46.

3.3.22.2 Key Parameters

The statistical methodology (ASTRUM) is applied to the large break LOCA analysis of the US-APWR. The key safety input parameters for the safety analysis of LOCA are as follows.

These parameters are selected for evaluating their ranges and distributions as they are the important safety parameters having significant impacts on PCT, LMO and CWO values.

The analysis is performed with the following additional conservative assumptions:

- As the plant initial condition, the normal power operation is the most limiting.
- The most limiting fuel conditions are used at the time of the accident. Of principal interest is the fuel stored energy.
- The reactor is operating in such a manner that the peak linear heat generation rate in the core is maximized.

3.4 Safety Evaluation Process Summary

When following this reload evaluation process, documentation should be generated that explains the methods used to determine if the reload safety parameters are bounded by existing safety analyses (Table 3-1 through Table 3-6 of this report should be referenced). If certain parameters are not bounded, the results of the new analyses should also be presented. It is possible that each event analysis could be impacted by the new core design, in which case the reload evaluation report would resemble the original DCD. If all parameters are bounded by existing safety analyses, the reload evaluation report would be a short summary of the fact that all reload parameters are more conservative than in the existing analyses.

3.5 References

3-1 US-APWR Design Control Document, Rev. 3, March 2011.

3-2 Friedland, A. J. and Ray, S., Revised Thermal Design Procedure, WCAP-11397-P-A (Proprietary) and WCAP-11397-A (Non-Proprietary), April 1989.

3-3 ANSI/ANS-5.1-1979, American National Standard for Decay Heat Power in Light Water Reactors, Approved August 29, 1979.

3-4 ANS-5.1, AMERICAN NUCLEAR SOCIETY PROPOSED ANS STANDARD Decay Energy Release Rates Following Shutdown of Uranium-Fueled Thermal Reactors, Approved October 1971

3-5 ECCS Evaluation Models, 10CFR 50, Appendix K.

3-6 Non-LOCA Methodology, MUAP-07010-P Rev. 2 (Proprietary) and MUAP-07010-NP Rev. 2 (Non-Proprietary), August 2011.

3-7 Thermal Design Methodology, MUAP-07009-P Rev. 0 (Proprietary) and MUAP-07009-NP Rev. 0 (Non-Proprietary), May 2007.

3-8 S.L. Humphreys et al., RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose estimation, NUREG/CR-6604, U.S. Nuclear Regulatory Commission, April 1998.

3-9 Large Break LOCA Code Applicability Report for US-APWR, MUAP-07011-P Rev. 1 (Proprietary) and MUAP-07011-NP Rev. 1 (Non- Proprietary), March 2011.

Table 3-1 Event Matrix (1) (Sheet 1 of 2)

Section	**15.1.1**	**15.1.2**	**15.1.3**				**15.1.4**	**15.1.5**		
Event*	FTD	EXFW	EXL				CRB	SLB		
Event Classification	AOO	AOO	AOO				AOO	AOO / PA		
Computer Codes Used	MARVEL-M	MARVEL-M	MARVEL-M				MARVEL-M ANC VIPRE-01M	MARVEL-M ANC VIPRE-01M		MARVEL-M
Analysis / Case**	DNBR	DNBR	DNBR Case A	DNBR Case B	DNBR Case C	DNBR Case D	DNBR	DNBR Case A	DNBR Case B	DNBR Case C
Nominal Conditions	**Design Basis Parameters**									
Reactor Power										
Thermal Design Flow										
Min. Measured Flow										
Reactor Avg Temp.										
Reactor Inlet Temp.										
Reactor Coolant Temp.										
RCS Pressure										
Bypass Flow										
Avg Linear Power Density										
	Generic Parameters									

Table 3-1 Event Matrix (1) (Sheet 2 of 2)

Section	**15.1.1**	**15.1.2**	**15.1.3**				**15.1.4**	**15.1.5**		
Event*	FTD	EXFW	EXL				CRB	SLB		
Event Classification	AOO	AOO	AOO				AOO	AOO / PA		
Computer Codes Used	MARVEL-M	MARVEL-M	MARVEL-M				MARVEL-M ANC VIPRE-01M	MARVEL-M ANC VIPRE-01M		MARVEL-M
Analysis / Case**	DNBR	DNBR	DNBR Case A	DNBR Case B	DNBR Case C	DNBR Case D	DNBR	DNBR Case A	DNBR Case B	DNBR Case C
	Event Specific Parameters									

Table 3-2 Event Matrix (2) (Sheet 1 of 2)

Section	**15.2.1**		**15.2.6**	**15.2.7**			**15.2.8**		
Event*	LOL / TT		LOOP	LONF			FLB		
Event Classification	AOO		AOO	AOO			AOO / PA		
Computer Codes Used	MARVEL-M		MARVEL-M	MARVEL-M			MARVEL-M		
Analysis / Case**	DNBR	RCS&MSP	PZR	DNBR	RCS	PZR	RCS	Hot Leg	PZR
Nominal Conditions	**Design Basis Parameters**								
Reactor Power									
Thermal Design Flow									
Min. Measured Flow									
Reactor Avg Temp.									
Reactor Inlet Temp.									
Reactor Coolant Temp.									
RCS Pressure									
Bypass Flow									
Avg Linear Power Density									
	Generic Parameters								

Table 3-2 Event Matrix (2) (Sheet 2 of 2)

Section	**15.2.1**		**15.2.6**	**15.2.7**			**15.2.8**		
Event*	LOL / TT		LOOP	LONF			FLB		
Event Classification	AOO		AOO	AOO			AOO / PA		
Computer Codes Used	MARVEL-M		MARVEL-M	MARVEL-M			MARVEL-M		
Analysis / Case**	DNBR	RCS&MSP	PZR	DNBR	RCS	PZR	RCS	Hot Leg	PZR
	Event Specific Parameters								

Table 3-3 Event Matrix (3) (Sheet 1 of 2)

Section	**15.3.1.1**	**15.3.1.2**		**15.3.3**		
Event*	PLOF	CLOF	UF	LOR & SHB		
Event Classification	AOO	AOO		PA		
Computer Codes Used	MARVEL-M VIPRE-01M	MARVEL-M VIPRE-01M		MARVEL-M VIPRE-01M		
Analysis / Case**	DNBR	DNBR	DNBR	DNBR	PCT	RCS
Nominal Conditions	**Design Basis Parameters**					
Reactor Power						
Thermal Design Flow						
Min. Measured Flow						
Reactor Avg Temp.						
Reactor Inlet Temp.						
Reactor Coolant Temp.						
RCS Pressure						
Bypass Flow						
Avg Linear Power Density						
	Generic Parameters					

Table 3-3 Event Matrix (3) (Sheet 2 of 2)

Section	**15.3.1.1**	**15.3.1.2**		**15.3.3**		
Event*	PLOF	CLOF	UF	LOR & SHB		
Event Classification	AOO	AOO		PA		
Computer Codes Used	MARVEL-M VIPRE-01M	MARVEL-M VIPRE-01M		MARVEL-M VIPRE-01M		
Analysis / Case**	DNBR	DNBR	DNBR	DNBR	PCT	RCS
	Event Specific Parameters					

Table 3-4 Event Matrix (4) (Sheet 1 of 2)

Section	**15.4.1**			**15.4.2**	**15.4.3**			**15.4.6**
Event*	RWS			RWP	DROP	Misaligned	SRWP	DIL
Event Classification	AOO			AOO	AOO	AOO	PA	AOO
Computer Codes Used	TWINKLE-M VIPRE-01M MARVEL-M			MARVEL-M	MARVEL-M	VIPRE-01M	MARVEL-M	N/A
Analysis / Case**	DNBR	TCL	RCS	DNBR	DNBR	DNBR	DNBR	--
Nominal Conditions	**Design Basis Parameters**							
Reactor Power								
Thermal Design Flow								
Min. Measured Flow								
Reactor Avg Temp.								
Reactor Inlet Temp.								
Reactor Coolant Temp.								
RCS Pressure								
Bypass Flow								
Avg Linear Power Density								
	Generic Parameters							

Table 3-4 Event Matrix (4) (Sheet 2 of 2)

Section	**15.4.1**			**15.4.2**	**15.4.3**			**15.4.6**
Event*	RWS			RWP	DROP	Misaligned	SRWP	DIL
Event Classification	AOO			AOO	AOO	AOO	PA	AOO
Computer Codes Used	TWINKLE-M VIPRE-01M MARVEL-M			MARVEL-M	MARVEL-M	VIPRE-01M	MARVEL-M	N/A
Analysis / Case**	DNBR	TCL	RCS	DNBR	DNBR	DNBR	DNBR	--
	Event Specific Parameters							

Table 3-5 Event Matrix (5) (Sheet 1 of 2)

Section	**15.4.7**				**15.4.8**								
Event*	Inadvertent Loading				Rod Ejection								
Event Classification	PA				PA								
Computer Codes Used	ANC				TWINKLE-M, VIPRE-01M, MARVEL-M, ANC								
Analysis / Case**	Case A	Case B	Case C	Case D	HFP,BOC			HFP,EOC		HZP,BOC	HZP,EOC		
					Fuel	RCS	DNBR	Fuel	DNBR	Fuel	Fuel	PCMI	
Nominal Conditions	**Design Basis Parameters**												
Reactor Power													
Thermal Design Flow													
Min. Measured Flow													
Reactor Avg Temp.													
Reactor Inlet Temp.													
Reactor Coolant Temp.													
RCS Pressure													
Bypass Flow													
Avg Linear Power Density													
	Generic Parameters												

Table 3-5 Event Matrix (5) (Sheet 2 of 2)

Section	**15.4.7**				**15.4.8**							
Event*	Inadvertent Loading				Rod Ejection							
Event Classification	PA				PA							
Computer Codes Used	ANC				TWINKLE-M, VIPRE-01M, MARVEL-M, ANC							
Analysis / Case**	Case A	Case B	Case C	Case D	HFP,BOC			HFP,EOC		HZP,BOC	HZP,EOC	
					Fuel	RCS	DNBR	Fuel	DNBR	Fuel	Fuel	PCMI
	Event Specific Parameters											

Table 3-6 Event Matrix (6) (Sheet 1 of 2)

Section	**15.5.2**	**15.6.1**	**15.6.3**		**15.6.5**
Event*	CVCS	DEP	SGTR		LOCA
Event Classification	AOO	AOO	PA		PA
Computer Codes Used	MARVEL-M	MARVEL-M	MARVEL-M		WCOBRA/TRAC HOTSPOT
Analysis / Case**	PZR	DNBR	Dose Input	SG overfill	PCT
Nominal Conditions	**Design Basis Parameters**				
Reactor Power					
Thermal Design Flow					
Min. Measured Flow					
Reactor Avg Temp.					
Reactor Inlet Temp.					
Reactor Coolant Temp.					
RCS Pressure					
Bypass Flow					
Avg Linear Power Density					
	Generic Parameters				

Table 3-6 Event Matrix (6) (Sheet 2 of 2)

Section	**15.5.2**	**15.6.1**	**15.6.3**		**15.6.5**
Event*	CVCS	DEP	SGTR		LOCA
Event Classification	AOO	AOO	PA		PA
Computer Codes Used	MARVEL-M	MARVEL-M	MARVEL-M		WCOBRA/TRAC HOTSPOT
Analysis / Case**	PZR	DNBR	Dose Input	SG overfill	PCT
	Event Specific Parameters				

*

FTD	Decrease in Feedwater Temperature
EXFW	Increase in Feedwater Flow
EXL	Increase in Steam Flow
CRB	Inadvertent Opening of a Steam Generator Relief or Safety Valve
SLB	Steam System Piping Failures
LOL/TT	Loss of Electrical Load / Turbine Trip
LOOP	Loss of Non-Emergency AC Power to the Station Auxiliaries
LONF	Loss of Normal Feedwater Flow
FLB	Feedwater System Pipe Break
PLOF	Partial Loss of Forced Reactor Coolant Flow
CLOF	Complete Loss of Forced Reactor Coolant Flow
UF	Frequency Decay
LOR	Reactor Coolant Pump Rotor Seizure
SHB	Reactor Coolant Pump Shaft Break
RWS	Uncontrolled Control Rod Assembly Withdrawal from a Subcritical
RWP	Uncontrolled Control Rod Assembly Withdrawal at Power
DROP	One or more dropped RCCAs within a group or bank
Misaligned	One or more misaligned RCCAs
SRWP	Uncontrolled withdrawal of a single RCCA
DIL	Inadvertent Decrease in Boron Concentration in the RCS
Inadvertent Loading	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position
Rod Ejection	Spectrum of Rod Ejection Accidents
CVCS	CVCS Malfunction that Increases Reactor Coolant Inventory
DEP	Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve
SGTR	Steam Generator Tube Rupture
LOCA	Loss-of-Coolant Accidents

**

DNBR	minimum DNBR
RCS	maximum RCS Pressure
MSP	maximum Main Steam Line Pressure
PZR	maximum Pressurizer Water Level
Hot Leg	Hot Leg Boiling
PCT	Peak Cladding Temperature
TCL	Pellet Centerline Temperature
Fuel	Pellet Centerline and Average Temperature, Cladding Temperature, Fuel Enthalpy

EXL Case A	Manual rod control, min. moderator feedback
EXL Case B	Manual rod control, max. moderator feedback
EXL Case C	Automatic rod control, min. moderator feedback
EXL Case D	Automatic rod control, max. moderator feedback
SLB Case A	Hot standby with offsite power
SLB Case B	Hot standby without offsite power
SLB Case C	Power condition
Inadvertent Loading Case A	Assembly interchange with a large reactivity difference
Inadvertent Loading Case B	Assembly interchange with a small reactivity difference
Inadvertent Loading Case C	Assembly interchange with and without burnable poison
Inadvertent Loading Case D	Burnable poison assembly loaded in incorrect location

4.0 EVALUATION OF NUCLEAR RELATED PARAMETERS

4.1 Introduction

In the nuclear design process, the preliminary loading pattern for the specific reload core is determined based on the requirement in design initialization process described in Section 2.2. In order to make the preliminary loading pattern be finalized, key safety parameters are selected and evaluated according to the safety evaluation described in Section 3.
In this section, more detail process of determining nuclear related key safety parameters is discussed together with their general characteristics in view of nuclear design. Key safety parameters for specific events are also discussed together with the assumptions and sequences of the nuclear calculation.

4.2 Determination of Nuclear Related Key Safety Parameters

Reload cores primarily change the characteristics of the plant in the following ways:

- Fuel inventory (core average enrichment and burnup)
- Fuel placement (loading pattern)
- Burnable absorbers (inventory, type, and location)

Depending on the combination and extent of changes in the above areas, both local and global core characteristics may change to a different degree for different safety analysis considerations.

These changes, in turn, affect the following nuclear input to safety analyses in three main areas.

- Core kinetic characteristics: global core reactivity parameters and coefficients that include, if necessary, spatial effects during an AOO or PA
- Control rod worths, both local and global core effects, depending on the AOO or PA
- Core power distributions for initial and 'statepoint' conditions, depending on the AOO or PA

In addition, other key safety input parameters or data may be required and are described for specific events; for example, the inlet moderator temperature distribution for a steamline break accident. More detail on specific applications and input requirements is provided in Section 3.

Generally, to bound the effects of the core reload, anticipated operation, and previous cycle operation, the following conditions are considered when calculating or evaluating key safety parameters for a reload cycle:

- BOC, MOC and EOC of the reload cycle,
- Full power, part power (if required), and zero power operation,
- RCCA insertions permitted by technical specifications
- Previous cycle operation, such as operation outside of the assumed cycle burnup window, or extended operation at part power
- Any other unanticipated plant behavior, startup results, or significant deviations from predictions for the previous cycle
- Any planned operation for the reload cycle which significantly deviates from baseload operation.

Key safety input parameters may be determined by a variety of methods:

- An explicit calculation for the specific reload design. This is usually required when the reload core differs significantly from the previous cycle in terms of fuel inventory, loading pattern, and/or burnable absorber inventory, type, and location, and cycle length and operation.
- An evaluation based on comparisons to (or use of) identical, similar, or bounding designs. In this case, a simple comparison can be made to demonstrate that the previously evaluated parameters are applicable and no additional calculations are required.
- Conservative evaluations or adjustments based on margin to safety limits
- “Default” bounding values based on generic analyses

The need for an explicit calculation or an evaluation is based on the sensitivity of a given safety input parameter to the core changes. For a typical reload evaluation, a combination of calculations and engineering judgment is usually performed. Reload cores possess varying degrees of similarity with previously evaluated reload cores and evaluation methods recognize this fact.

The primary nuclear analysis code used for safety analyses is the ANC code (Reference 4-1). ANC is a three dimensional two-group diffusion core calculation code based on nodal expansion method. Using few-group constants generated by PARAGON (Reference 4-1), ANC calculates nuclear parameters such as critical boron, power distribution, exposure, reactivity coefficients etc.

4.2.1 Core Reactivity Parameters and Coefficients

“Core kinetics” generally refers to a time-dependent change in core power level due to a change in core conditions. For any transient, the rate and magnitude of power level change is a result of many factors, including inherent core characteristics, initial operating conditions, and the nature of the transient. This section describes the inherent core characteristics and some of the initial operating conditions, referred to as “kinetic characteristics”, assumed for providing the nuclear design input to the transient analyses. As described below, many factors influence the core kinetics, and these factors contribute to the overall rate and magnitude of power level changes differently.

Reactivity coefficients are calculated based on the change in reactivity due to a change in a given parameter, which may be isolated or combined with other parameters. Examples of useful isolated plant parameters, which are calculated independently of other changes, are moderator temperature, fuel temperatures (Doppler), and boron worth. An example of a combined coefficient is the power coefficient, which includes moderator, Doppler, and power redistribution effects.

Reactivity coefficients and defects are calculated using an explicit 3D model at the desired conditions. The coefficients are typically generated for a given reference condition by calculating core reactivity at two points which represent a small change from the reference condition. Power distribution (and redistribution) at the two points is implicitly included in the 3D calculation. Reactivity coefficients are usually “snapshots” of core response at specific core conditions, and can be sensitive to operational variables such as cycle (core average) burnup, power level, moderator temperature, boron concentration, control rod insertion, and power distribution. The selection of appropriate reactivity coefficients for a given AOO or PA depends on the sensitivity of the coefficient to a given condition, and at which conditions the transient of interest is considered to be limiting. For example, a transient may be evaluated at BOC or EOC (or both), and the most conservative reactivity coefficient chosen may be either the least negative or most negative value that produced the most limiting result.

The transient analyses discussed in Section 3 use appropriate or conservative combinations of reactivity coefficients.

4.2.1.1 Moderator Reactivity Coefficient

Moderator reactivity coefficients may be expressed in terms of either temperature (reactivity change per moderator temperature change pcm/°F) or density (reactivity change per moderator density change pcm/g/cc), depending on the application or transient analyzed. At the highest level, the moderator reactivity coefficient is sensitive to the neutron flux spectrum: a harder flux spectrum results in a more negative moderator reactivity coefficient, and vice-versa. The neutron flux spectrum is affected by moderator density, moderator temperature, fuel burnup, boron concentration, and control rods and/or burnable absorbers.

Moderator temperature coefficients are calculated using 3D models for various conditions as described in Section 4.2, and therefore any power redistribution effects are implicitly accounted for. The coefficients are determined by varying the moderator temperature a small amount from a reference value.

4.2.1.2 Doppler Coefficient

Doppler feedback is typically expressed as the fuel temperature coefficient, which is the change in core reactivity per degree change in 'effective' fuel temperature (pcm/°F). The effective fuel temperature takes into account the thermal conductivity and radial power distribution in the pellet as well as the local isotopic effects in the pellet. A major contributor to the fuel temperature coefficient is the change (broadening or narrowing) of ^{238}U and ^{240}Pu resonance absorption peaks as the effective fuel temperature changes.

Doppler coefficients are calculated using 3D models for various conditions as described in Section 4.2, and therefore implicitly include any effects due to power redistribution. The coefficients are determined by varying core power level or effective fuel temperature a small amount from a reference value while maintaining a constant moderator temperature.

Power coefficients, which are expressed as a change in reactivity per percent change in power, include moderator, Doppler, and power redistribution effects. The ‘Doppler-only’ component of the power coefficient can be determined from the calculations described above.

4.2.1.3 Boron Worth

Boron worth is typically expressed in terms of a core reactivity change for a given change in boron concentration (pcm/ppm). Being primarily a thermal neutron absorber, boron worth depends both on the amount of soluble boron present, as well as the effect of changes in moderator temperature, fuel burnup and control rods and/or burnable absorbers.

Boron worths are calculated using 3D models for various conditions as described in Section 4.2, and therefore implicitly include any effects due to power redistribution. The coefficients are determined by varying boron concentrations a small amount from a reference value while maintaining a constant power level and moderator temperature.

4.2.1.4 Delayed Neutron Data and Prompt Neutron Lifetime

The kinetic (dynamic) response of the core during transient conditions is also largely determined by the ratio of ‘prompt’ (essentially instantaneous) fissions to ‘delayed’ neutrons, and the prompt neutron lifetime; this is particularly important for AOOs or PAs that reach limiting values in a very short timeframe (such as a rod ejection). Delayed neutrons are contributed from fission products (precursors), at a relatively short time after fission occurs. The delayed neutron contribution from different fissionable isotopes varies due to their different production of precursor isotopes. To determine an ‘effective total’ delayed neutron inventory and prompt neutron lifetime for a core, the various fissionable isotopes must be weighted both by the total amount of the isotope and the ‘power sharing’ of the isotope (i.e., the fraction of fissions contributed by the given isotope).

The effective core delayed neutron fractions and prompt neutron lifetime are calculated using power distributions from 3D models. The delayed neutron fractions and prompt neutron lifetime for each fuel region or sub-region (defined by common enrichment, burnup, etc.) are determined by the burnup/isotope inventory data coming from the lattice code (Reference 4-2) and power weighted by the 3D results.

4.2.2 Control Rod Worth Parameters

RCCAs are used to apply fairly rapid reactivity insertion to the core for operation and to ensure that the plant can be shutdown at any point in the cycle with sufficient shutdown margin to meet safety limits in the case of an AOO or PA.

Control banks are used for axial power distribution control particularly during Xenon transients, and to compensate for reactivity changes due to temperature, power level, boron concentration, and total Xenon inventory. The allowable insertion of control banks at power is restricted by the rod insertion limit defined in technical specifications, as described in more detail below. Shutdown banks are held ‘in reserve’ and are fully withdrawn during power operation.

Rod worths are calculated for each reload cycle. Changes in fuel and burnable absorber inventories, loading patterns, and previous cycle operation affect the core power distribution and flux spectrum. Therefore, the reactivity worth of individual rods, banks, and the total worth of all RCCAs can vary for each reload. As a result, key safety input parameters may be affected by the reload and must be evaluated for each reload; these parameters include total rod worth, differential rod worth, rod insertion limits, and trip reactivity worth and shape. This section discusses how the parameters described above are evaluated for each reload. Specific events such as ejected rod and dropped rod are discussed separately in other sections of this report. Note that these parameters should be evaluated considering margins in accordance with the criteria for Rod Worth Measurement during the Startup Physics Tests (Reference 4-2).

4.2.2.1 Total Rod Worth

Total (integral) rod worth is the amount of negative reactivity available from all control and shutdown banks at HZP conditions, assuming the ‘worst’ (highest worth) RCCA is fully withdrawn (stuck) out of the core. The calculation is performed, at a minimum, at BOC and EOC conditions, with a 3D core model. An initial case is established by fully inserting all control and shutdown banks. A ‘stuck rod search’ is performed to determine the highest worth RCCA by withdrawing each RCCA individually. The worst stuck rod worth is subtracted from the total rod worth determined from the initial case, and the result is the “N-1” rod worth. A conservative adjustment or multiplier is applied to the N-1 rod worth for conservatism. The N-1 rod worth is the basis for determining minimum shutdown margin and is an input to safety analysis as described in Section 3.

4.2.2.2 Differential Rod Worth

Differential rod worths, defined in pcm/step, are evaluated or calculated for each reload. At a minimum, BOC and EOC conditions are considered, assuming both HFP and HZP initial conditions. For full power cases, the control banks are assumed to move in normal sequence with the programmed control bank overlap, and all control bank insertions permitted by technical specifications are considered. For hot zero power conditions, two sequential control banks are withdrawn simultaneously with 100% overlap. The differential rod worths are calculated with 3D core models.

4.2.2.3 Rod Insertion Limits

Control rod insertion limits establish the maximum control bank insertion as a function of core power. As power level is reduced, the allowable control rod insertion is increased, and control bank overlap is assumed. The insertion limits are established based on the following considerations:

- Peaking factor limits are met during normal operation
- Minimum required shutdown margin is maintained
- Rod Ejection limits are met

Insertion limits are established or confirmed for each reload using 3D models.

4.2.2.4 Trip Reactivity

Trip reactivity is defined in terms of total rod worth as well as the shape (inserted rod worth versus rod position) during a trip of the control banks at full power. The worst stuck rod (N-1 configuration) is assumed for the trip. Because the trip reactivity shape is affected by the axial power distribution (and burnup) of the core, it is evaluated based on full power axial power distributions for the entire cycle; in general, the cycle burnup with the most negative axial offset is the limiting point. The calculation is performed with a 3D core model at the most adverse condition of the axial offset allowed by the technical specifications for the initial condition. To simulate the trip, all control rods are inserted in stepwise fashion to obtain the worth versus rod position. In general, the trip reactivity shape that is most ‘bottom’ skewed (i.e., with the rod worth insertion delayed as much as possible) is considered conservatively.

A normalized curve of worth versus position is generated, and a minimum total worth value is used with this curve as input to the safety analysis (See Section 3).

4.2.3 Peaking Factors

Core power distribution varies with RCCA insertion, power level, and core burnup, etc. In general, core power distribution can be discussed with the combination of radial and axial power distribution. Radial power distribution is mainly dependent on the fuel loading pattern, and the core designer tries to flatten the power distribution by an adequate arrangement of new and burnt fuel assemblies, gadolinia doped assemblies, and burnable absorber assemblies. The degree of flatness of radial power distribution can be expressed as the radial peaking factor $F_{\Delta H}$ which is a maximum rod power relative to the average rod power. On the other hand, axial power distribution can vary with the operating condition and can be easily controlled by the RCCA movements. The degree of flatness of axial power distribution can be expressed as the axial peaking factor F_Z ,however, more commonly used index is the three dimensional peaking factor F_Q which is the maximum local power relative to the average rod power.

In safety analysis, bounding $F_{\Delta H}$ and F_Q values throughout the entire cycle duration are used as the hottest rod power and hottest point power, respectively.

4.3 Key Safety Parameters for Specific Events

This section addresses the determination of key safety input parameters for following specific events or classes of events.

- Loss-of-Coolant Accidents
- Inadvertent Decrease in Boron Concentration in the Reactor Coolant
- Control Rod Misoperation (System Malfunction or Operator Error)
- Uncontrolled Control Rod Assembly Withdrawal from a Subcritical
- Spectrum of Rod Ejection Accidents
- Steam System Piping Failure

In the evaluation of the key safety parameters for the above events, the worst case conditions or inputs including the effects of previous cycle burnup and operating history are considered regarding the event specific nature.

4.3.1 Loss-of-Coolant Accidents

The statistical methodology (ASTRUM) is applied to the large break LOCA analysis of the US-APWR. The key safety input parameters for LOCA are

The engineering hot channel factor F_Q and the nuclear uncertainty factor F_Q are set to certain values for the US-APWR as bounding values, however, justification of using these bounding value is done for each reload.

4.3.2 Inadvertent Decrease in Boron Concentration in the RCS

An inadvertent decrease of the boron concentration in the reactor coolant can occur due to the addition of low-boron-concentration water into the reactor coolant due to a malfunction or improper operation of the chemical and volume control system (CVCS). This transient results in a positive reactivity addition to the core. At this event, the maximum dilution rates from charging/discharging sources are assumed.

The key safety input parameters for boron dilution events are

For each reload, it is confirmed that sufficient operator response time is provided in order to deal with the event from boron dilution initiation at each operation mode defined in technical specifications.

4.3.3 Control Rod Misoperation (System Malfunction or Operator Error)

Individual RCCA events include one or more dropped RCCAs, one or more misalignment RCCAs, and uncontrolled withdrawal of a single RCCA at full power.

The key safety input parameters for events that involve individual (single) RCCAs are Events considered are static misalignment, dropped rods, and rod withdrawal. The evaluation is performed to determine the RCCA that produces The evaluation is performed, at a minimum, for BOC and EOC burnup conditions, The analysis is performed with a 3D core model.

4.3.4 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical

The key safety input parameter for the uncontrolled control rod assembly withdrawal from subcritical is The evaluation is performed, at a minimum, for BOC and EOC burnup conditions at HZP, although other points in the cycle may also be chosen. Two RCCA banks are assumed to be withdrawn simultaneously with 100% overlap.

The analysis is performed with a 3D core model.

4.3.5 Spectrum of Rod Ejection Accidents

For fuel performance analysis of rod ejection events, the key safety input parameters are the total peaking factor F_Q and the ejected rod worth, which determine the maximum local power during the transient. The evaluation is performed, at a minimum, for BOC and EOC burnup conditions, although other points in the cycle may also be chosen depending on peaking factor behavior (i.e., cycle burnup(s) where the maximum peaking factor occurs). At each cycle burnup point chosen, both hot zero power and hot full power are analyzed, assuming the control rods are at the insertion limits defined by the technical specifications.

For DNBR analysis,

The analysis is performed with a 3D core model which incorporates the following assumptions:

4.3.6 Steam System Piping Failure

Steamline break accidents are typically divided into two categories:

- A “credible” break, in which the main steam system experiences an accidental depressurization either from a small pipe or valve failure, and

- A "hypothetical" break, in which a main steamline experiences a major rupture.

Generally, the hypothetical break is the limiting event, and therefore reload evaluations consider only this event.

The key safety input parameters for steamline break are

The evaluation is performed at EOC conditions, where shutdown margin becomes minimum, due to maximum positive moderator feedback during a cooldown event. The plant is assumed to be in hot shutdown with control bank RCCAs at fully inserted position except one stuck rod. The stuck RCCA with the maximum rod worth at these conditions is assumed to be fully withdrawn from the core (N-1 stuck rod conditions).

The analysis is performed with a 3D core model that incorporates global core information from a reference transient analysis of the event, at the point of limiting margin to safety limits; these are referred to as "statepoints". This information includes core power level, boron concentration, flow rate, inlet temperature, the local inlet temperature distribution, and system pressure.

The analysis is performed with a 3D core model.

Based on these initial conditions, the results of the cycle-specific 3D solution for either reactivity (in the event of subcriticality) or power level (in the event of return-to-power) are compared to the initial transient analysis statepoints. Criteria are established to confirm that either the reference transient analysis results are conservative and bounding, or the transient analysis is revised to bound the cycle-specific results.

In addition to confirmation of transient analysis results, the data generated from the 3D model is provided as input to Thermal/Hydraulic DNBR calculations.

4.4 References

4-1 Qualification of Nuclear Design Methodology using PARAGON/ANC, MUAP-07019-P Rev. 0 (Proprietary) and MUAP-07019-NP Rev.0 (Non-Proprietary), December 2007.

4-2 Reload Startup Physics Tests for Pressurized Water Reactors, ANSI/ANS-19.6.1-2011, January, 2011

5.0 THERMAL AND HYDRAULIC EVALUATION

5.1 Introduction

The thermal-hydraulic evaluation of reload core is performed mainly for the following two objectives. The first objective is to check or re-evaluate the core thermal limit and local linear heat rate (LHR) limit, and provide the core operating limits report (COLR) information expected in Technical Specification. The second objective is to confirm thermal-hydraulic parameters used in the safety analysis are applicable to the reload cycle concerned. If those limit and/or the output parameters for safety analysis are affected considerably, re-design of the reload core or re-evaluation of safety analysis should be initiated.

The reference analyses are performed based upon the 'bounding analysis' concept, in which a set of key parameters are generated to bound all future anticipated operating cycles. For each reload, the key parameters are reevaluated, recalculated, and compared to the bounding values used for the reference analyses. If the parameter is bounded, the reference safety analysis is valid; otherwise, additional evaluation or analysis is required to demonstrate that safety criteria and design bases are met. Although the reference analysis is intended to be valid for all plant reloads, changes to the plant, significant changes in anticipated fuel management, or other issues may require additional analyses or evaluations.

5.2 Key Parameters

Key parameters of thermal and hydraulic design related to reload core evaluation are briefly described below. Analytical methodologies used in each evaluation are comprehensively described in Reference 5-1 and 5-2.

- **Core thermal Limit**

Core thermal limit is a reactor core inlet temperature limit as a function of pressure and reactor thermal power, on which safety analysis limit of DNBR is reached. VIPRE-01M code and applicable DNB correlations, which are validated in Reference 5-1, are used. Safety analysis limit of DNBR is determined considering necessary design uncertainties and margin, in accordance with revised thermal design procedure (RTDP, Reference 5-3). The uncertainty of DNB correlation and key input parameters for DNBR analysis, such as plant operating conditions, core radial power distribution and fuel manufacturing data, are statistically combined into the DNBR limit.

The core thermal limit assumes a design bounding shape for the axial power distribution, which covers those from normal operation, and a corrective function of the axial offset is provided to be applicable for the anticipated operational occurrences (AOOs).

- **Fuel Temperature and Local LHR Limit**

Fuel temperature and its associated parameters, such as uncertainty and gap conductance, are calculated for the various local burnup and LHR by using FINE code (Reference 5-2). The results are used for determining local LHR limit and input for Safety Analysis.
The local LHR limit is determined as a maximum allowable LHR vs. local burnup, and corresponds to the fuel centerline melting condition, considering the design uncertainty and margin. Nuclear analyses are performed to confirm the local LHR limit is not exceeded in any conditions of the core during normal operation and AOOs under the over power protection systems.

The dominant factors which can affect fuel temperature are pellet density, pellet densification, helium backfill pressure, fuel enrichment, and fuel dimensions.

- **Rod Internal Pressure**

The rod internal pressure is maintained below the cladding liftoff pressure, at which the pellet-cladding gap increases due to outward cladding creep. This criterion allows fuel rods to have the rod internal pressure higher than the system pressure. Therefore, the DNB propagation is taken into account for normal operation, AOOs and postulated accidents.

- **Corrosion**

The pellet cladding mechanical interaction (PCMI) failure criteria is a change in radial average fuel enthalpy greater than the corrosion-dependent limit depicted in SRP 4.2 Appendix B. The criterion for hydrogen content of the US-APWR fuel limits the corrosion level within the range given in the PCMI criteria in SRP4.2 Appendix B. Therefore, the PCMI failure criteria is applicable to US-APWR fuel and the fuel enthalpy change is compared with the PCMI criteria as described in Section 3.3.18.

Many inputs to the thermal and hydraulic design generally do not change as a result of a reload design. Examples of these inputs include the reactor coolant system pressure, core thermal power, and fuel design. Other inputs are much more likely to change for a reload design and are therefore more likely to be a basis for re-evaluation or re-analysis. A change to any of these parameters will be evaluated for the reload design.

5.3 Typical Thermal Evaluations for Reloads

Reload thermal-hydraulic design is performed using input from reload core design as well as fuel design and plant operating conditions. However, in most cases, fuel design and plant operating conditions are not changed significantly. Therefore, the reload safety evaluation methodology focuses on effects of reload core design. In the initiation of reload safety evaluation process, degree of change in the fuel design and plant operating condition are checked. If their changes are considerably large, the specific safety evaluation process should be initiated.

5.3.1 Core Thermal Limit

The following parameters depend on reload core design and may affect DNBR or its uncertainty are following:

These parameters are generally bounded by reference analysis conditions and it is confirmed in the reload core evaluation.

In addition, the safety analysis limit DNBR is confirmed to include sufficient margin to cover rod bow penalty and following cycle-specific penalties:

The DNB correlations to be used for the reload evaluation are determined by the fuel assembly designs, mainly the grid spacer designs. If the new DNB correlation is used for the reload core design, the re-analyses are required for all the DNB analyses in thermal and hydraulic design and safety analyses.

5.3.1.1 Nuclear Enthalpy Rise Hot Channel Factor, $F_{\Delta H}^{N}$

Nuclear enthalpy rise hot channel factor, $F_{\Delta H}^{N}$, represents the ratio of the hot channel enthalpy rise to core-average enthalpy rise. The value of DNBR reflects the enthalpy rise to the point where the DNBR is calculated. The design value of $F_{\Delta H}^{N}$ is determined to cover the ratio of maximum value of rod integral power to core averaged integral rod power.
It is confirmed that the design value of $F_{\Delta H}^{N}$ bounds the reload core conditions via the nuclear design.

For partial power operation, deeper control rod insertion limit than that for rated thermal power (RTP) operation requires relaxation of $F_{\Delta H}^{N}$.

$$F_{\Delta H}^{N} = F_{\Delta H}^{RTP}\left[1+0.3(1-P)\right]$$

where:

$F_{\Delta H}^{N}$: the limit at partial power

P : the power fraction to RTP

$F_{\Delta H}^{RTP}$: the limit at RTP

The permitted relaxation of $F_{\Delta H}^{N}$ is embedded in the core thermal limit and allows radial power distribution changes with control rod insertion to the insertion limits.

5.3.1.2 Axial Power Distributions

The axial power distribution changes with burnup and is affected by power level, control rod movement, and transition of xenon distribution in the axial direction. The shift of axial power distribution during normal operation can be contained by controlling the change in axial power imbalance which is monitored by the ex-core nuclear detectors.

A conservative axial power distribution is used as the design axial power distribution for the DNB analyses, in which the axial power distribution does not change significantly. It is also assumed for evaluation of core thermal limit. The conservatism of the design power distribution is to be confirmed in the reload core designs.

When the axial power distribution is skewed during AOOs, the core thermal limit is lowered by the corrective function for the axial offset, which is incorporated by the over temperature ΔT trip.DNBR analyses are performed to confirm the applicability of the corrective function, The skewed axial power distributions predicted by the nuclear evaluations of the AOO conditions for reload core are used in those analyses.

5.3.1.3 Engineering Hot Channel Factor

Engineering hot channel factors account for the effects of variations in fuel fabrication on peaking factors, as described in more detail below. The engineering hot channel factors are considered to be conservatively determined and generic in nature, and therefore do not need to be re-evaluated for a reload core.

5.3.1.3.1 Heat Flux Engineering Hot Channel Factor, F_Q^E

F_Q^E accounts for the effect of fuel fabrication variations on the local heat flux, and is determined by statistically combining the fabrication variations for fuel pellet diameter, density and enrichment as well as fuel pellet eccentricity and fuel rod diameter variation. F_Q^E represents the possibility of the local power spike caused by the local variance of above mentioned factors and is considered for peak LHR and fuel centerline temperature calculations. However, no DNB penalty need be taken for such a small heat flux spike, because DNB is based on an integrated fuel channel quantity.

5.3.1.3.2 Enthalpy Rise Engineering Hot Channel Factor, $F_{\Delta H,1}^E$

$F_{\Delta H,1}^E$ accounts for the effect of fuel fabrication variations on the hot channel enthalpy rise, and is determined by statistically combining the fabrication variations for fuel pellet diameter, density, and enrichment. This factor is directly considered in the VIPRE-01M subchannel analysis, and is one of the statistically treated input parameters in RTDP.

5.3.1.4 Axial Fuel Stack Shrinkage

Axial fuel stack shrinkage due to fuel densification results in severe DNBR, and is affected by fuel pellet sintering temperature. However, the shrinkage is compensated by thermal expansion and therefore the total shrinkage factor, which includes densification and thermal expansion, is small. Since the conservatively selected value is used in reference analysis, reload core condition is generally bounded.

5.3.1.5 Hydraulic Evaluation

The hydraulic evaluation of a reload core is performed to determine the impact of any changes to the fuel design, including the fuel rods, grids spacers, and top/bottom nozzles. The reload evaluation determines the hydraulic compatibility of the reload fuel with the fuel carried over from the previous cycles.

The inconsistent pressure drop between the fuel assemblies causes the undesirable flow redistribution and consequent DNB penalty. This transition core DNB penalty is estimated taking account of the fraction of high pressure drop fuels in the reload core.

5.3.2 Fuel Temperatures

The local LHR limit, which corresponds to the design limit of fuel temperature, is determined as a function of local burnup using the relationship between fuel centerline temperature of UO_2 fuel and local LHR.

The peak LHR for gadolinia-doped pellets is maintained lower than that of uranium dioxide pellets by reducing UO_2 enrichment. Consequently, the temperature of fuel with Gd_2O_3 is designed to be non-limiting.

Compliance with this limit is verified with the ANC core model (Reference 5-4). The peak LHR in the reload core during normal operation and AOOs is covered by the local LHR limit, which ensures that the fuel centerline temperatures remain below the melting temperature of the fuel. To avoid fuel centerline melting, the local LHR limit is selected as the overpower limit and serves as a basis for overpower protection system setpoints.

The dominant factors which can affect fuel temperature are pellet density, pellet densification, helium backfill pressure, fuel enrichment, and fuel dimensions.

If the fuel enrichment is within the scope of the reference analyses, there is no impact on the fuel temperatures. The other parameters are generally not changed from reference design, however, if the above parameters are changed, a new fuel temperature analysis is performed and compared with the reference design. If the new temperature is not bounded by the reference data, impact on the events described in Section 3.0 needs to be checked.

Fuel temperature data for input to safety analyses are calculated for reload designs if necessary. Fuel temperature profiles, which include fuel centerline, average, surface temperatures as a function of local linear heat rate, are calculated along with the fuel rod lifetime to determine the maximum fuel temperatures.

5.3.3 Rod Internal Pressure

One of the most important inputs for rod internal pressure is power history of fuel rod in the reload core. FINE code provides the rod internal pressure for rods in the reload core, regarding the power histories, fuel design parameters, such as pellet density and its densificaiton, enrichment, helium backfill pressure, and other dimensions. The DNB propagation is assessed with the rod internal pressure in the reload core. If the rod internal pressure is bounded by the reference analysis, no further safety analysis would be necessary. If it is not, the effect is taken into account.

5.4 References

5-1 Thermal Design Methodology, MUAP-07009-P Rev.0 (Proprietary) and MUAP-07009-NP Rev.0 (Non-Proprietary), May, 2007.

5-2 Mitsubishi Fuel Design Criteria and Methodology, MUAP-07008-P Rev.2 (Proprietary) and MUAP-07008-NP Rev.2 (Non-Proprietary), July 2010.

5-3 Friedland, A. J. and Ray, S., Revised Thermal Design Procedure, WCAP-11397-P-A (Proprietary) and WCAP-11397-A (Non-Proprietary), April 1989.

5-4 Qualification of Nuclear Design Methodology using PARAGON/ANC, MUAP-07019-P Rev. 0 (Proprietary) and MUAP-07019-NP Rev.0 (Non-Proprietary), December 2007.

6.0 CONCLUSIONS

The Mitsubishi safety analysis methodology is based upon the ‘bounding analysis’ concept, therefore the main key safety parameters are generated to bound all future anticipated operating cycles for the reference safety parameter. However, significant changes in fuel management, or other issues may require a change in input safety parameters and thus additional analyses.

The process of demonstrating safety of reload cycle was described in this technical report as the sequence of the reload design, the concept of selecting key safety parameters, and the way of confirming whether the key safety parameters are bounded or reevaluation is needed. Reload evaluation of plants with Mitsubishi safety analysis are to be performed according to the process and the concept described in this report.