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Revision 0

**AURORA-B: An Evaluation Model for
Boiling Water Reactors; Application to
Control Rod Drop Accident (CRDA)**

March 2014

AREVA Inc.



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Control Rod Drop Accident (CRDA)**

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Nature of Changes

Item	Page	Description and Justification
1.	All	This is the initial release.

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Nomenclature

ABWR	advanced boiling water reactor
ASME	American Society of Mechanical Engineers
AST	alternate source term
BOC	beginning of cycle
BPWS	banked position withdrawal sequence
BWR	boiling water reactor
CCD	component calculational device
CET	component effects test
CFR	code of federal regulations
CHF	critical heat flux
CPR	critical power ratio
CRDA	control rod drop accident
CZP	cold zero power
EM	evaluation model
EMDAP	evaluation model development and assessment process
EPRI	Electric Power Research Institute
EPU	extended power uprate
FIST	full integral simulation test
FMA	foundation methodology assessment
FMCRD	fine motion control rod driver
FoM	figure of merit
GDC	general design criteria
IET	integral effects test
LHGR	linear heat generation rate
LOCA	loss of coolant accident
LPRM	local power range monitor
LTR	licensing topical report
LWR	light water reactor
MCPR	minimum critical power ratio
NRC	Nuclear Regulatory Commission, U.S.
PCMI	pellet clad mechanical interaction
PHE	peak hot excess reactivity
PIRT	phenomena identification and ranking table

PTD	plant transient data
PWR	pressurized water reactor
RIA	reactivity insertion accident
RLBLOCA	realistic large break loss of coolant accident
RPS	reactor protection system
SE	safety evaluation
SET	separate effects test
SRP	standard review plan
U. S. NRC	Nuclear Regulatory Commission, U. S.
ΔH	transient change in enthalpy
ΔH_p	prompt enthalpy rise
ΔH_{tot}	total enthalpy rise

Abstract

The AREVA methodology to analyze the boiling water reactor (BWR) Control Rod Drop Accident (CRDA) is presented. The methodology includes the use of a nodal three-dimensional kinetics solution with both thermal-hydraulic (T-H) and fuel temperature feedback. These models provide more precise localized neutronic and thermal conditions than previous methods to show compliance with criteria for the BWR CRDA event as presented in the U. S. NRC Standard Review Plan Section 15.4.9 (Reference 1). This report presents the CRDA requirements, followed by the code and model requirements, the application methodology, and methodology uncertainties.

1.0 Introduction

The AREVA methodology to analyze the boiling water reactor (BWR) control rod drop accident (CRDA) is presented. The methodology includes the use of a nodal three-dimensional kinetics solution with both thermal-hydraulic (T-H) and fuel temperature feedback. These models provide more precise localized neutronic and thermal conditions than previous methods to show compliance with regulatory criteria for the BWR CRDA event as presented in the U. S. NRC Standard Review Plan Section 15.4.9. (Reference 1). The methodology is structured such that it will support changes in the acceptance criteria.

The AREVA methodology for the CRDA evaluation includes both generic evaluations and cycle-specific analysis. Generic studies are used to address at power conditions and system pressurization. The cycle specific analysis includes the determination of candidate control rods that could challenge fuel failure criteria and the subsequent evaluation of these candidate rods with a three-dimensional neutron kinetics and thermal-hydraulics code system.

This methodology has been developed to support recent changes in the CRDA acceptance criteria and evaluation process as reflected in the Interim Acceptance Criteria and Guidance of Appendix B of SRP 4.2 (Reference 6).

This methodology is based on extending the model qualification of the AURORA-B system described in Reference 9. The term AURORA-B CRDA will be used throughout this document to distinguish between the AURORA-B Base Evaluation Model described in Reference 9 and the methodology presented in this document.

2.0 Summary

The models and methodology described and documented herein provides a means to show compliance with interim acceptance criteria for the BWR CRDA event as established in SRP 4.2 Appendix B (Reference 6).

The range of applicability includes BWR plant types, control blades, and fuel designs for which the AREVA lattice physics and nodal simulator methods have been validated or will be validated for. { The CASMO-4/MICROBURN-B2 methodology described in the NRC approved Topical Report, EMF-2158PA, Revision 0, (Reference 17), identifies the specific commercial reactors included within the topical in Section 7.2 . These reactors are identified as core size and the core lattice of either C or D. Application to reactors not included in the topical is demonstrated in accordance with Item 1 of Section 6 of SER for Reference 17. Benchmarking of the CASMO-4/MICROBURN-B2 methodology consistent with Item 1 of Section 6 of the SER, allows for modeling of BWRs equipped with external recirculation pump systems (BWR/2 plants), jet-pump recirculation systems (BWR/3 through BWR/6 plants), and internal recirculation pump systems (similar to ABWR plants). }

Evaluation of the event requires the establishment of initial conditions covering the range of cycle operation (exposure) and domain of coolant conditions (flow rate, pressure and temperature).

The cycle is evaluated to determine exposure points, target rod patterns, coolant conditions and candidate rods which when dropped could generate a power pulse that could potentially cause cladding failure. The selection of the initial conditions and candidate rods for evaluation is described in Sections 7.3 and 7.4.

The candidate rods are subsequently evaluated with the transient code which includes the following capabilities:

- Three-dimensional neutron kinetics which properly models the power excursions, including changes in reactivity for the dropped rod, the proper treatment of the Doppler effect in the fuel, the power peaking during the transient, and the reactivity change associated with the change in moderator conditions.
- Transient thermal energy transport in the fuel pellet capable of accommodating the rapid deposition of energy from the power excursion, and to properly represent the subsequent transport of thermal energy from the fuel pellets to the coolant and changes to the coolant state, momentum and energy.

- A control system model that properly inserts the non-faulted control rods on the receipt of a RPS trip signal.
- Determination of fuel enthalpy or CPR during the event.

The selection of the candidate rods is based on static rod worth as determined with the CASMO4/MICROBURN-B2 methodology (NRC approved Topical Report EMF-2158(P)(A) Reference 17). The location of the rod relative to fuel with potentially high hydrogen content in the cladding is also considered due to the impact on acceptance criteria.

Following selection of the candidate rods, the rod drops are then evaluated with the AURORA-B code system and the results are compared against the CRDA acceptance criteria. The methodology incorporates a hydrogen pickup model to allow assessment of the PCMI failure criterion as defined in the Interim Acceptance Criteria and Guidance of Appendix B of SRP 4.2 (Reference 6).

If fuel failures are determined, the radiological consequences are evaluated by comparing the fission product inventory release determined with revised enthalpy dependent release fractions against the actual fission product inventory release used in the plant licensing basis.

This methodology replaces the prior AREVA methodology for the CRDA (Reference 41) approved in 1983 with modern methods for fuel and plant analysis.

The methodology and documentation presented, identifies the various and current regulatory requirements applicable for the BWR CRDA event and identifies how these requirements are addressed and complied with.

3.0 Regulatory Requirements Summary

3.1 Regulatory Requirements

The acceptance criteria for the CRDA are based on General Design Criterion (GDC) 13 and 28 (Reference 2) and radiation dose limits from 10 CFR 100.11 and 10 CFR 50.67 (References 3 and 2). It is noted that specific acceptance criteria for the reactivity initiated accident are under review and that interim criteria are provided in SRP Section 4.2 Appendix B (Reference 6).

The specific acceptance criteria from NUREG-0800 Chapter 15.4.9 (Reference 1) are:

1. General Design Criterion (GDC) 13, as to the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges.
2. Acceptance criteria are based on GDC 28 requirements as to the effects of postulated reactivity accidents that result in neither damage to the reactor coolant pressure boundary greater than limited local yielding nor result in sufficient damage to impair significantly core cooling capacity.

Regulatory positions and specific guidelines necessary to meet the relevant GDC 28 requirements are in SRP Section 4.2 Appendix B and have been identified as interim criteria for the Reactivity Insertion Accident. These interim criteria are summarized in Table 3.1.

The maximum reactor pressure during any portion of the assumed excursion should be less than the value that causes stress to exceed the "Service Limit C" as defined in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code.

3. 10 CFR 100.11 and 10 CFR 50.67 establish radiation dose limits for individuals at the boundary of the exclusion area and at the outer boundary of the low population zone.

Final revised SRP design basis acceptance criteria are anticipated to be issued in the near future. The actual methodology is structured to support changes in the final criteria.

Table 3.1 Interim Criteria Summary

Criteria			SRP 4.2 Appendix B Revision 3
Fuel cladding failure	High Temperature	Zero Power	170 cal/g total enthalpy internal rod pressure below system pressure 150 cal/g total enthalpy internal rod pressure exceeding system pressure
		Power > 5%	Local heat flux exceed thermal design limit CPR
	PCMI	Cladding Hydrogen content (ppm)	
		<75	150 cal/g prompt enthalpy
		75 to 150	Linear decrease from 150 cal/g at 75 ppm to 60 cal/g at 150 ppm
		150 to 300	Linear decrease from 60 cal/g at 150 ppm to 50 cal/g at 300 ppm
Core coolability	Peak radial average enthalpy		<230 cal/g
	Peak Fuel Temperature		Below incipient fuel melting conditions
	Mechanical Energy	Non-molten fuel to coolant interaction	Address with respect to reactor pressure boundary, reactor internals, and fuel assembly structural integrity
		Rod burst	
	Cool-able geometry	Fuel pellet and cladding fragmentation	No loss of cool-able geometry
		Fuel rod ballooning	
Radiological			10 CFR 100.11 or 10 CFR 50.67 radiation dose limits met at exclusion area boundary and at the outer boundary of the low population zone

3.2 ***Compliance with NUREG-0800 Chapter 15.0.2***

NUREG-0800 SRP 15.0.2 "Review of Transient and Accident Analysis Methods" (Reference 7) describes the review process and acceptance criteria for analytical models and computer codes. This section summarizes the compliance with the requirements identified in the SRP 15.0.2.

Several elements are identified related to documentation and model requirements in SRP 15.0.2. The documentation includes the following components:

- A. An overview of the evaluation model which provides a clear roadmap describing all parts of the evaluation model, the relationships between them, and where they are located in the documentation.

A comprehensive description of the AURORA-B base EM code system is provided in ANP-10300P (Reference 9). The actual model requirements for the CRDA event are established in Section 5. The model description is provided in Section 7 along with roadmap for the CRDA analysis.

- B. A description of the accident scenario including initial conditions, the initiating event and all subsequent events and phases of the accident, and the important physical phenomena and interactions between systems and components, if appropriate, that affect the outcome of the accident.

The scenario description and discussion of phenomena interactions are provided in Section 4. The result of the phenomena identification and ranking are reflected in the evaluation model requirements in Section 5.

- C. A comprehensive description of the code assessment.

A summary of the code assessment data base is presented in Section 6.0. The results of the code assessment are provided in Section 8.0.

- D. Determination of code uncertainty for a sample plant accident calculation.

The code uncertainty bases are established within Section 8 and the application for a sample plant accident calculation is provided in Section 9.

- E. A comprehensive theory manual.

The AURORA-B code system for the CRDA evaluation contains multiple computational devices. The theory manuals for S-RELAP5, MB2-K and RODEX4 are References 19, 20, and 22.

- F. A detailed user's manual.

The user manuals for S-RELAP5, MB2-K and RODEX4 are References 57, 68, and 69. The interface requirements for interaction of the computational devices are provided in Section 9.0 along with process and procedures to ensure correct transfer of information between the various computational devices.

- G. A description of the Quality Assurance program under which the evaluation model was developed and assessed, and the corrective action program that will be used to address any error that might be discovered.

The AREVA Quality Assurance program, which complies with the requirements of Appendix B to 10 CFR Part 50 is identified in Section 10.0.

4.0 Scenario Identification

4.1 *Regulatory Basis*

From SRP 15.0.2 II.1.B the requirement for the accident scenario is presented which requires:

A description of the accident scenario including initial conditions, the initiating event and all subsequent events and phases of the accident, and the important physical phenomena and interactions between systems and components, if appropriate, that affect the outcome of the accident.

The following section identifies the accident characteristics and the sequence of events.

Detailed discussion of the phenomena identification and ranking is presented in Section 5.0 with the model requirements.

4.2 *Characteristics*

The BWR Control Rod Drop Accident (CRDA) is the result of a postulated event in which a fully inserted high worth control rod becomes decoupled from its drive mechanism. The drive mechanism is withdrawn but the decoupled control rod is assumed to be stuck. At a later optimum moment, the control rod drops at its maximum speed to the position of the control rod drive mechanism. This results in the insertion of large positive reactivity into the core resulting in a prompt critical power increase which is terminated by negative Doppler reactivity from the fuel temperature increase. With the large power excursion, the Reactor Protection System will initiate an insertion of the withdrawn control rods. The power pulse width for a BWR CRDA is approximately 45 to 75 milli-seconds such that the power has increased and decreased prior to the beginning of movement of the control rods in response to the scram. Although a small portion of the fission energy is directly deposited in the moderator, no significant change in the reactivity due to moderator feedback occurs until thermal conduction of heat from the fuel to moderator begins. Once the thermal energy transfer begins the local moderator temperature increases and creates substantial local voiding. This voiding results in additional negative reactivity insertion. Some fluctuation in the power from the void feedback may occur until the scram is completed which terminates the event.

The general characteristics of a CRDA are summarized:

- A rapid insertion of positive reactivity (the worth of the dropped rod)
- A rapid increase in localized power along with fuel temperature

- Insertion of negative reactivity through the broadening of resonance absorption cross section (Doppler effect)
- Rapid decrease in power generation in response to the negative reactivity
- Heat transfer from the fuel to coolant resulting in:
 - Small decrease in the negative Doppler reactivity with decreasing fuel temperature
 - Large increase in negative reactivity from moderator voiding
- Insertion of negative reactivity of SCRAM control rods

The analysis of CRDA is generally divided between two operating regimes: the startup range and the power range.

CRDA in the Power Range

In the power range, there are reactor characteristics that reduce the severity of the CRDA. The first is the consequences of the void distribution of the operating reactor which results in a smaller reactivity insertion rate for the dropped rod. With the presence of voids, and the coolant at saturated conditions, the direct heating of moderator has a greater potential to decrease the moderator density in the power range compared to the cold conditions for the startup range and the increase in voids results in smaller rod worths. Other characteristics are related to the Doppler Effect. The Doppler feedback in the power range occurs much more quickly relative to the magnitude of the power increase compared to that in the startup range. (In the startup range, the actual power must increase by several orders of magnitude prior to significant Doppler feedback.) Finally, with the presence of voids in the moderator the fuel to water ratio increases and the spectrum hardens. With the hardened spectrum the resonance absorption of neutrons increases.

These characteristics all tend to reduce the severity of the CRDA in the power range.

CRDA in the Start-Up Range

During the startup sequence, the rod patterns employed are permitted by constraints on rod movements by the technical specification restrictions on the order in which control rods are withdrawn including the maximum number of bypassed control rods. The Banked Position Withdrawal Sequence (Reference15) is an example of a set of restrictions intended to reduce the maximum rod worth that is used by most BWRs. These type of withdrawal sequences are

typically enforced with rod pattern control systems. The AREVA CRDA methodology presented herein can be applied to any specified rod withdrawal sequence.

Sequence of Events and System Response

The sequence of events for the CRDA may be summarized as follows (this sequence has been annotated from that given in Reference 14):

1. At some time during the withdrawal of control rods from the reactor, a complete (but not necessarily sudden) rupture, breakage, or disconnection of a random fully inserted control rod drive from the companion control blade occurs.
2. The control rod drive associated with the disconnected control blade is withdrawn as a part of the start-up process and the control rod blade sticks at the fully inserted position, rather than follow the moving control rod drive. This fault is not detected by the plant operators.
3. At some later time, under critical reactor conditions, the rod pattern causes the decoupled rod to have the maximum worth from fully inserted to the position of its drive. The faulted control blade drops at that time to the location of the companion control rod drive.
4. A rapid increase in reactor power near the location of the dropped control blade occurs, and a correspondingly sudden increase in fuel temperature in the nearby fuel assemblies. The fuel temperature reactivity feedback (Doppler) terminates the initial power burst.
5. The reactor protection system trips on a high flux signal from the Source Range Monitor, the Intermediate Range Monitor, or the Average Power Range Monitor; although the trip is generally assumed to occur at a reactor power of 120%, rather than explicitly representing the various trip systems.
6. All of the withdrawn control rods aside from the decoupled control rod insert into the core (scram) at the technical specification scram speed.
7. When all of the control rods, except for the faulted control rod, are fully inserted, the reactor is assured to be sub-critical due to the Shutdown Margin requirement. This state represents the termination of the accident.

Event Evaluation

Evaluation of the event requires the establishment of initial conditions covering the range of cycle operation (exposure) and domain of coolant conditions (flow rate, pressure and temperature).

The cycle is evaluated to determine exposure points, target rod patterns, coolant conditions and candidate rods which when dropped could generate a power pulse that could potentially cause cladding failure.

The candidate rods are subsequently evaluated with the transient code which includes the following capabilities:

- Three-dimensional neutron kinetics which properly models the power excursions, including changes in reactivity for the dropped rod, the proper treatment of the Doppler effect in the fuel, the power peaking during the transient, and the reactivity change associated with the change in moderator conditions
- Transient thermal energy transport in the fuel pellet capable of accommodating the rapid deposition of energy from the power excursion, and to properly represent the subsequent transport of thermal energy from the fuel pellets to the coolant and changes to the coolant state, momentum and energy
- A control system model that properly inserts the non-faulted control rods on the receipt of a RPS trip signal
- Determination of fuel enthalpy or CPR during the event

The result of the transient are then evaluated against the CRDA criteria to assess fuel failures, core coolability, system over pressurization, and radiological consequences.

5.0 Evaluation Model Requirements

5.1 *Regulatory Basis*

From SRP 15.0.2 II.1.B the requirement for the accident scenario is presented which includes: A description of the accident scenario including initial conditions, the initiating event and all subsequent events and phases of the accident, and the important physical phenomena and interactions between systems and components, if appropriate, that affect the outcome of the accident.

5.2 *Model Requirements*

The accident characteristics and sequence of events are presented in Section 4. This section presents a detailed discussion of the phenomena identification and ranking. The format used for the subsequent sub-sections is based on Element 1 of Reference 8. Since this methodology development is based on the AURORA-B base EM (Reference 9), the EMDAP process is focused on the changes to the AURORA-B base EM and extension of the validation for CRDA.

5.2.1 Analysis Purpose, Transient Class, Plant Class, and Fuel Designs

The purpose of the analysis is to ensure that the regulatory requirements for the BWR CRDA event are met.

The transient class for which the methodology described within this topical is defined in SRP 15.4.9 Spectrum of Rod Drop Accidents (BWR). In some instances the licensing basis for a plant may not adhere to the SRP, or may refer to prior revisions of the SRP. It is noted that revision of the acceptance criteria for the CRDA is ongoing at the time of document preparation.

The BWR plant types, control blades, and fuel designs for which the methodology is applicable, should be consistent with those for which the AREVA lattice physics and nodal simulator methods have been validated. This includes BWRs equipped with external recirculation pump systems (BWR/2 plants), jet-pump recirculation systems (BWR/3 through BWR/6 plants), and internal recirculation pump systems (similar to ABWR plants). A detailed description of the BWR plant types is provided in Reference 14. With respect to the CRDA, the primary difference between reactor designs is the incorporation of Fine Motion Control Rod Drive (FMCRD). The control blade design for FMCRD does not incorporate the velocity limiter. BWR2-6 plants all have velocity limiters.

5.2.2 Figures of Merit

The Figures of Merit (FoM) evaluated are parameters that demonstrate compliance with applicable acceptance criteria defined in the licensing basis of each plant. Specifically, the FoMs (also known as acceptance criteria measures) considered are those necessary to demonstrate compliance with the acceptance criteria for the CRDA. Ideally the FoM's would correspond identically to the regulatory requirement however, regulatory guidance permits "surrogate" standards for demonstrating compliance. The FoMs include prompt fuel enthalpy rise, total fuel enthalpy, MCPR, system pressure, fission product inventory release fractions, and core coolability. (The fission product inventory release is used as a "surrogate" FoM for the event dose consequences.) Evaluation of the MCPR and system pressurization FoM's are included in the AURORA-B base EM.

Figure of Merit 1 Fuel Enthalpy

Fuel enthalpy is used in the determination of cladding failure and gap release fractions. Two enthalpy values are considered; the prompt enthalpy rise and the total enthalpy. The prompt fuel enthalpy rise is defined as the radial average fuel enthalpy rise at the time corresponding to one pulse width after the peak of the prompt power pulse. The total enthalpy is simply the maximum enthalpy experienced during the event. The enthalpy figure of merit is used in two ways for evaluating the CRDA. The first is to determine if fuel cladding failure criteria have been exceeded and the second is used with other methods to determine radiological consequences of the CRDA if fuel cladding failure occurs.

1. The prompt fuel enthalpy rise is evaluated and compared to the criteria for PCMI (Pellet Clad Mechanical Interaction) cladding failure. The total enthalpy is compared against the high temperature cladding failure criteria.
2. If the loss of fuel cladding integrity has occurred, independent of the driving mechanism (PCMI, high temperature, MCPR) this figure of merit in conjunction with other methods can be used in the determination of the gap inventory release fraction for the failed rods.

Figure of Merit 2 MCPR

Evaluation of MCPR is performed for at power conditions to determine fuel failure. If a rod exceeds the critical power it is assumed failed and will be included in the radiological consequence evaluation. This figure of merit is addressed in the AURORA-B base EM (Reference 9).

Figure of Merit 3 Peak System Pressure

Evaluation of peak system pressure is made to assure the peak system pressure remains below applicable ASME limits for the CRDA. This figure of merit is addressed in the AURORA-B base EM (Reference 9).

Figure of Merit 4 Fission Product Inventory Released

The total fission product inventory released from fuel rods which have experienced cladding failures is used for comparison with plant licensing basis for the CRDA. The inventory released is determined by the gap inventory release fractions which are composed of an enthalpy dependent release term and a steady state term. The inventory released can then be used to determine an equivalent number of failed fuel rods relative to using only the steady state release term for comparison with a specific number of rod failures. Alternatively, the inventory released can be compared directly to the plant licensing basis source term for the CRDA.

Figure of Merit 5 Core Coolability

Evaluation of core coolability is addressed using maximum deposited enthalpy, peak fuel temperature, the maximum cladding temperature and the power pulse characteristics.

5.2.3 Identify Systems, Components, Phases, Geometries, Fields, and Processes That Will Be Modeled

The evaluation model is the same as that used for the analysis of Anticipated Operational Occurrences with the extension of qualification to cold reactor conditions, and large rapid insertions of positive reactivity. The systems, components, phases, geometries, fields, and processes modeled are unchanged from Reference 9.

5.2.4 PIRT Summary

The figures of merit were reviewed with respect to the existing capabilities of the AURORA-B base EM. The ability of the AURORA-B base EM to assess the MCPR FoM and the peak system pressure FoM is documented in Reference 9. Therefore, the CRDA PIRT is generated for the enthalpy FoM. The AURORA-B base EM PIRT (Reference 25) and NRC PWR rod ejection PIRT (Reference 16) were used in the PIRT development.

The Fission Product Inventory Release FoM and the Core Coolability FoM are based on the determination of fuel failures which are caused by exceeding an enthalpy criteria or MCPR.

This PIRT is specific to the evaluation of the CRDA event. The generation of the PIRT involved review of existing industry PIRTs for the CRDA and PWR rod ejection.

Ranking of the phenomena was completed with respect to their influence on the enthalpy FoM.

Rankings are high (H), medium (M), low (L) or not-ranked (N).

Table 5.1 Control Rod Drop Accident Analysis PIRT

Subcategory	Phenomenon	R	Rationale
Calculation of power history during pulse	Control rod worth	H	Determines the amount of reactivity insertion.
	Rate of reactivity insertion	M	Within limits, the outcome is insensitive to the rate of reactivity insertion.
	Moderator feedback	M	The moderator temperature rise is small and corresponding reactivity effect is small, but not negligible.
	Fuel temperature feedback	H	The fuel temperature feedback causes the power excursion to turn around and essentially limits the energy deposition.
	Delayed-neutron fraction	H	Determines when prompt criticality is reached
	Reactor trip reactivity	L	It is important to have the rods trip to terminate the event but the effect is minor relative to the pulse.
	Fuel cycle design	H	Determines control rod worth and core loading
	Direct Energy deposition to moderator	L	The fraction of energy deposited directly in the moderator is small.
Calculation of rod fuel enthalpy increase during pulse (including cladding temperature)	Heat resistances in fuel, gap, and cladding	M	Per Reference 16, at a maximum, 25 percent of the deposited energy is conducted out and does not contribute to the fuel enthalpy.
	Transient cladding-to-coolant heat transfer coefficient	M	Per Reference 16, at a maximum, 25 percent of the deposited energy is conducted out and does not contribute to the fuel enthalpy.
	Heat capacities of fuel and cladding	M	Enthalpy is the integral of heat capacity and temperature. Enthalpy and enthalpy increases are both highly important. Small variations in the heat capacities have little impact on the overall event.
	Pellet radial power distribution	M	This element is rated lower because it is only part of the overall heat transfer process.
	Rod-peaking factors	H	Determines how much energy is directed to the peak location.
	Pellet and Cladding dimensions	L	Reflected in the heat resistances
Initial conditions	Burn-up distribution	N	
	Cladding hydrogen content	M	Determines enthalpy failure criteria
	Power	M	Affects reactivity feedback
	Coolant temperature and pressure	M	Affects moderator properties
	Core Flow	M	Affects moderator properties during transient
	Fuel rod internal pressure	L	Determines high temperature failure threshold

5.2.4.1 Control Rod Worth

The control rod worth determines the amount of reactivity insertion. A strong correlation exists between the control rod worth and the peak fuel enthalpy. The worth depends on fuel cycle design, cycle lifetime, and initial conditions. The initial conditions are required to be a reasonable representation of the limiting conditions allowed by plant Technical Specifications with respect to inoperable control rods that maximize the worth of the dropped rod.

The MICROBURN-B2/CASMO-4 methodology described in the NRC approved Topical Report, EMF-2158, Revision 0, (Reference 17), is fully capable of determining the dropped control rod worth, and establishing the appropriate initial conditions for power, flow, temperature, rod pattern and state in cycle.

5.2.4.2 Rate of Reactivity Insertion

The rate of reactivity insertion is controlled by the actual rod worth and the rate at which the control rod drops from the core. For BWRs 2 through 6, the control rod drop velocity is maintained with the control rod velocity limiter. Based on the results of velocity limiter tests, a conservative maximum rod velocity of 3.11 ft/s is used from the Reference 10 Appendix Velocity Limiter Tests (Page A-1). This was determined to be a bounding conservative velocity for BWRs 2 through 6.

The methodology will use a bounding rod drop rate or provide measured data which can support a slower drop rate.

5.2.4.3 Moderator Feedback

For the CRDA, moderator feedback is the change in the reactivity feedback from moderator temperature and density changes in the active, by-pass and water channels/rods. This also includes the changes in void fractions in the coolant. These changes are a result of direct energy deposition to the coolant and heat transfer from the cladding. Both conditions are specifically modeled in the 3-D kinetics code with thermal hydraulics feedback.

The methodology will model both moderator temperature and moderator void feedback. This capability is demonstrated in the AURORA-B base EM.

5.2.4.4 Fuel Temperature Feedback

Fuel temperature feedback is the reactivity feedback from fuel temperature changes. This results from the heating of fuel and the associated neutronic effects, in particular the Doppler effect. The Doppler reactivity feedback is the primary mechanism which reverses the power transient.

The evaluation module will model the Doppler feedback as a function of the fuel temperature changes.

5.2.4.5 Effective Delayed-Neutron Fraction

The effective delayed neutron fraction (β_{eff}) determines when prompt criticality is reached.

The evaluation model will include the effective delayed neutron fraction.

5.2.4.6 Reactor Trip Reactivity

Reactor trip reactivity is the negative reactivity associated with insertion of control rods after receipt of a reactor trip signal and the rate at which it is inserted. The actual event is turned around by Doppler prior to the start of motion of the rods being inserted. Consequently it will ultimately terminate the event in that all control rods, except the dropped rod, are inserted into the core to bring it to a subcritical state.

The actual magnitude of the power pulse is insensitive to the reactor trip reactivity. Therefore, the model will use minimum plant Technical Specification scram rates.

5.2.4.7 Fuel Cycle Design

The fuel cycle design includes those important design elements that determine the neutronic properties of the core at event initiation, such as the loading pattern, control history (control rod pattern, spectral shift), burnup, and exposure.

The evaluation model will represent the reactor core characteristics with respect to fuel loading and operation history. This capability is demonstrated in the AURORA-B base EM.

5.2.4.8 Heat Resistances in High Burnup Fuel, Gap, and Cladding

The resistances offered by the fuel, gap, and cladding to the flow of thermal energy from regions of high temperature to regions of lower temperature. The resistance is dependent upon the path length and thermal conductivity, which change with burnup.

The model must represent the exposure dependent heat transfer properties of the fuel, gap, and cladding. This capability is demonstrated in the AURORA-B base EM.

5.2.4.9 Transient Cladding-to-Coolant Heat Transfer Coefficient

The correlation that determines transport of energy at the interface by one or more of the following modes: forced convection-liquid, nucleate boiling, transition boiling, film boiling, or forced convection-vapor.

Due to the rapid nature of the power burst, there is limited change in heat transfer during the power burst. However, subsequent to the power burst the heat transfer from the rod to the coolant will be represented.

5.2.4.10 Heat Capacities of Fuel and Cladding

The respective quantities of heat required to raise the fuel and cladding one degree in temperature at constant pressure. The heat capacity of the fuel defines the amount of absorbed energy prior to an increase in temperature. The evaluation model will represent the material heat capacities.

5.2.4.11 Direct Energy Deposition to Moderator

Direct energy deposition into the active channel fluid occurs as the fluid attenuates γ -rays and neutrons that escape the fuel rods. The amount of energy deposited in the fluid is a function of the local fluid density and locally generated fission power of the fuel. The partition of the energy deposition which is deposited directly in the fuel pellet and that deposited in the moderator and structural components may affect the reactivity feedback mechanism.

The evaluation model will partition the energy deposition between the fuel and moderator/coolant within the core.

5.2.4.12 Pellet Radial Power Distribution

The radial distribution of the power produced in the fuel rod describes the change in power intensity from the center of the pellet to its outermost edge. The pellet radial power profile could affect the rate of energy transferred from the fuel pellet to coolant and the pellet temperature distribution for Doppler effects.

5.2.4.13 Rod-Peaking Factors

Rod-peaking factors represent the nodal rod or pellet power distribution within an assembly during the transient. As the rod is withdrawn there will be a significant change in the pin peaking factors.

The peaking factor is utilized to determine the pin enthalpy and critical power. The peaking includes the effects of power tilts that may occur across the bundles during the CRDA. The pin peaking factor change during the transient must be properly modeled.

5.2.4.14 Pellet and Cladding Dimensions

Uncertainty in the pellet and cladding dimensions are well known at beginning of life. The pellet and cladding dimensions change with exposure. The primary impact of dimensional changes is on the thermal heat transfer from the fuel to the coolant. The impact of the variance in the pellet and cladding dimensions are reflected in the thermal mechanical properties of the fuel rod.

5.2.4.15 Burnup Distribution

The local radial burnup distribution is homogenized in the cross section lookup process. The power distribution uncertainty of the kinetics includes the impact of the radial burnup distribution. The burnup distribution is reflected through the core power distribution, the control rod worth, and the fuel rod thermal-mechanical properties.

5.2.4.16 Cladding Hydrogen Content

The content of hydrogen in the cladding at the start of the transient is used in the determination of the failure criteria. This does not have an impact on the actual kinetic response, but impacts fuel cladding failure threshold for PCMI.

The methodology includes a method to calculate the cladding hydrogen content at the beginning of the event.

5.2.4.17 Power Distribution

The total core power and core power distribution at the start of the event. The initial power distribution is driven by the time in cycle and the initial control rod pattern.

The evaluation model will adequately represent the core power and power distribution at the start of the event.

5.2.4.18 Coolant Initial Conditions

The coolant initial conditions include pressure, temperature and the flow distribution (active channel, bypass, and water rod flows) at the start of the transient.

The methodology will address the spectrum of initial conditions for the event. The initial conditions in the power range are addressed in the AURORA-B base EM.

5.2.4.19 Total Core Flow

The total core flow refers to the total core flow at the beginning of the event. The actual flow value is not expected to change during the event.

The evaluation model will be capable of modeling the localized flow re-distribution during the event. This is addressed in the AURORA-B base EM.

5.2.4.20 Fuel Rod Internal Pressure

The fuel rod internal pressure relative to the system pressure is used to determine the high temperature failure criteria. This parameter is only important if the less restrictive high temperature criteria are applied.

6.0 Assessment Data Base Summary

6.1 Regulatory Basis

From SRP 15.0.2 II.1.C the requirement for the documentation requires a comprehensive description of the code assessment data base.

6.2 Assessment Data Base

A summary of the Assessment Data Base is presented in this section. The actual assessment results are provided in Section 8. The comprehensive assessment data base and the evaluation of the AURORA-B base EM are provided in Section 4 and 6 of Reference 9. For evaluation of the CRDA, the assessment base is expanded to include modeling of the SPERT III RIA experiments. For the at-power CRDA, the assessment data base information provided for the AURORA-B base EM (Reference 9) includes reactivity insertion events. For the CRDA in the start-up range, a sub-set of the assessment data supplied for the AURORA-B base EM is applicable. A summary of the Assessment Data Base is presented in this section for those items identified as important for the CRDA evaluation. The actual assessment results are provided in Section 8.

Table 6.1 is the assessment matrix which shows the component effects tests that were selected and the highly ranked PIRT phenomena addressed.

Core State Conditions

The capability of the CASMO4/MICROBURN-B2 code system to provide adequate cross sections, fuel burnup, spectral history information, and static control rod worth is provided in the LTR for CASMO4/MICROBURN-B2 (Reference 17). The assessment base for the CASMO4/MICROBURN-B2 consist of several reactors and cycles of measured data for comparison of measured TIP signals and cold and hot eigenvalue benchmarking. The assessment base also includes gamma scan data as well as calculations with higher order methods.

Numerical Kinetics Benchmarks

Numerical benchmarks provide a useful test of the equations and numerical solutions because the benchmarks are mathematical problems whose solutions are well known. Specifically, they

demonstrate the performance of the kinetics equations to simulate industry-standard problems that are specifically designed to test spatial and temporal equations numerical solutions. The assessment data base for the numerical benchmark include the 2-D TWIGL seed-blanket reactor problem (Reference 30), the LMW 3-D PWR delayed critical transient problem (Reference 31) and the LRA 2D and 3D BWR Control Blade Drop Transients (Reference 32).

Integral system test of Turbine Trip Event

The assessment of the capability of the AURORA-B base EM to model global fast transient conditions for an actual reactor utilizes the Peach Bottom Turbine Trip tests (Reference 37 and 39). This provides an assessment of 3D power distribution, scram reactivity of control rods, void/reactivity relationship, fuel rod thermal & mechanical properties and Doppler reactivity.

Integral test of a CRDA

A series of reactivity tests were performed as part of the Special Power Excursion Test (SPERT) III E-core program (Reference 42). The actual tests were conducted in the 1960s as described in Reference 43. These experiments involved the rapid reactivity insertions ranging from \$0.5 to \$1.3. The experiments were performed at cold startup, hot startup, hot standby, and operating power.

The capability of the AURORA-B system to simulate the kinetic response to the large insertion of reactivity is evaluated with this assessment.

Hydrogen Pickup Model

The hydrogen pickup model used is described in Reference 58 "Response to question 1b" on pages 10 through 21. The assessment of the model against measured data is provided in Figure 2 of Reference 58 (page 18).

Table 6.1 AURORA-B CRDA Evaluation Model Assessment Matrix

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7.0 Evaluation Model Description

7.1 *Regulatory Basis*

From SRP 15.0.2 II.1.A, requires that the documentation contain an overview of the evaluation model which provides a clear roadmap describing all parts of the evaluation model, the relationships between them, and where they are located in the documentation.

A comprehensive description of the AURORA-B base EM code system is provided in ANP-10300P (Reference 9). The actual model requirements for the CRDA event are established in Section 5. The following Sections contain descriptions of the evaluation model and process for analysis of the CRDA and includes a roadmap for the CRDA analysis. This is collectively referred to as the CRDA methodology.

7.2 *CRDA Evaluation Methodology*

The AREVA methodology for CRDA analysis involves the selection of conservative initial conditions (see Section 7.3) and candidate rods (see Section 7.4) which would result in the most severe consequences for the CRDA. Following the selection of the candidate control rods, the rod drops are simulated with the AURORA-B code system. The results obtained from the simulation are then evaluated against the Interim Acceptance Criteria of Appendix B of SRP 4.2 (Summarized in Section 3.1 Regulatory Requirements of this document) to assess rod failure and subsequent fission product inventory release. The methodology includes generic components, plant specific components, and cycle specific components. The CRDA evaluation road map is presented in Figure 7.1. Detailed guidance for the analysis of the CRDA with the AURORA-B code system is provided in Reference 65.

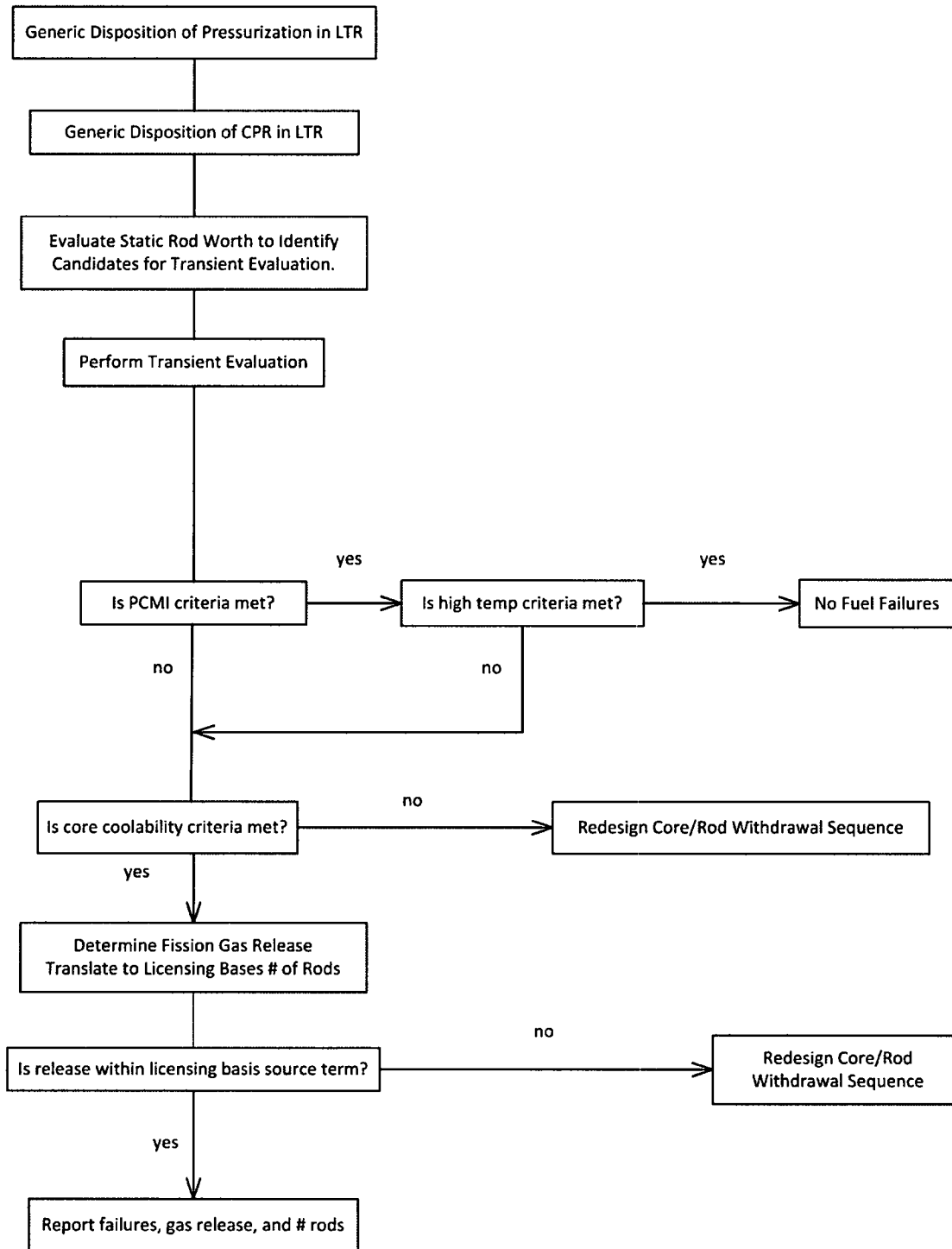


Figure 7.1 CRDA Evaluation Road Map

7.3 ***Determination of Conservative Initial Conditions and Parameters***

The CRDA analysis methodology involves the selection of initial conditions that represent the most limiting conditions for the CRDA. The selection of the conservative initial conditions differs between the startup range and the power range. The primary initial conditions for a given point in the cycle include the core coolant conditions, power, flow, and control rod pattern. The in cycle evaluation times are also determined.

7.3.1 Startup Range

In the startup range the initial conditions that must be determined are the initial coolant properties, the initial core power, the initial core flow, and the inoperable control rods. The determination of the initial core flow and the inoperable control rods is determined on plant specific bases. Sensitivity studies for initial conditions are included within the methodology assessment results provided in Section 8.

7.3.1.1 Initial Coolant Conditions

The selection criteria has been developed using simulation of reactor startup sequences with various temperatures and pressures. Although the actual static rod worth varied little with moderator temperature, the lower temperature for a given rod worth results in higher fuel enthalpy. Sensitivity studies for initial coolant conditions are included within the methodology assessment results provided in Section 8.7.2.3.

The use of the cold temperature below that at which the reactor actually goes critical is conservative for the startup range CRDA for the following reasons:

- There can be a small positive reactivity increase with modern fuel designs as the moderator temperature increase up to 300 °F.
- The actual temperature at which BWRs go critical is higher than 70°F as indicated in Table 7.1.

Table 7.1 BWR Critical Temperatures

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7.3.1.2 Initial Flow and Power

The initial flow rate and power are determined based on sensitivity studies as shown in Sections 8.7.2.4 and 8.7.2.5. The event is assumed to occur with a xenon free core.

7.3.1.3 In-Cycle Evaluation Times

The analysis is performed at beginning of cycle, peak hot excess reactivity and end of full power conditions. These in-cycle conditions represent various core conditions and reactivity arrangements that may be encountered throughout the cycle.

7.3.1.4 Inoperable Control Rods

Inoperable rods locations are defined consistent with those allowed by plant technical specifications in such a manner to maximize the worth of the candidate rods. [

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A typical A-sequence group assignment of control rods for a core with 764 assemblies is shown in Figure 7.2. The inoperable rods are located in South-East half of the core and are placed as close as allowed. Plant Technical Specification identify the maximum number of rods that can be out of service, the maximum number of rods per rod group, and the minimum separation distance between out of service rods. Although 3 inoperable rods may be allowed, two rods are assigned to each of the first 4 groups. The assignment requires two rods separation between inoperable rods.

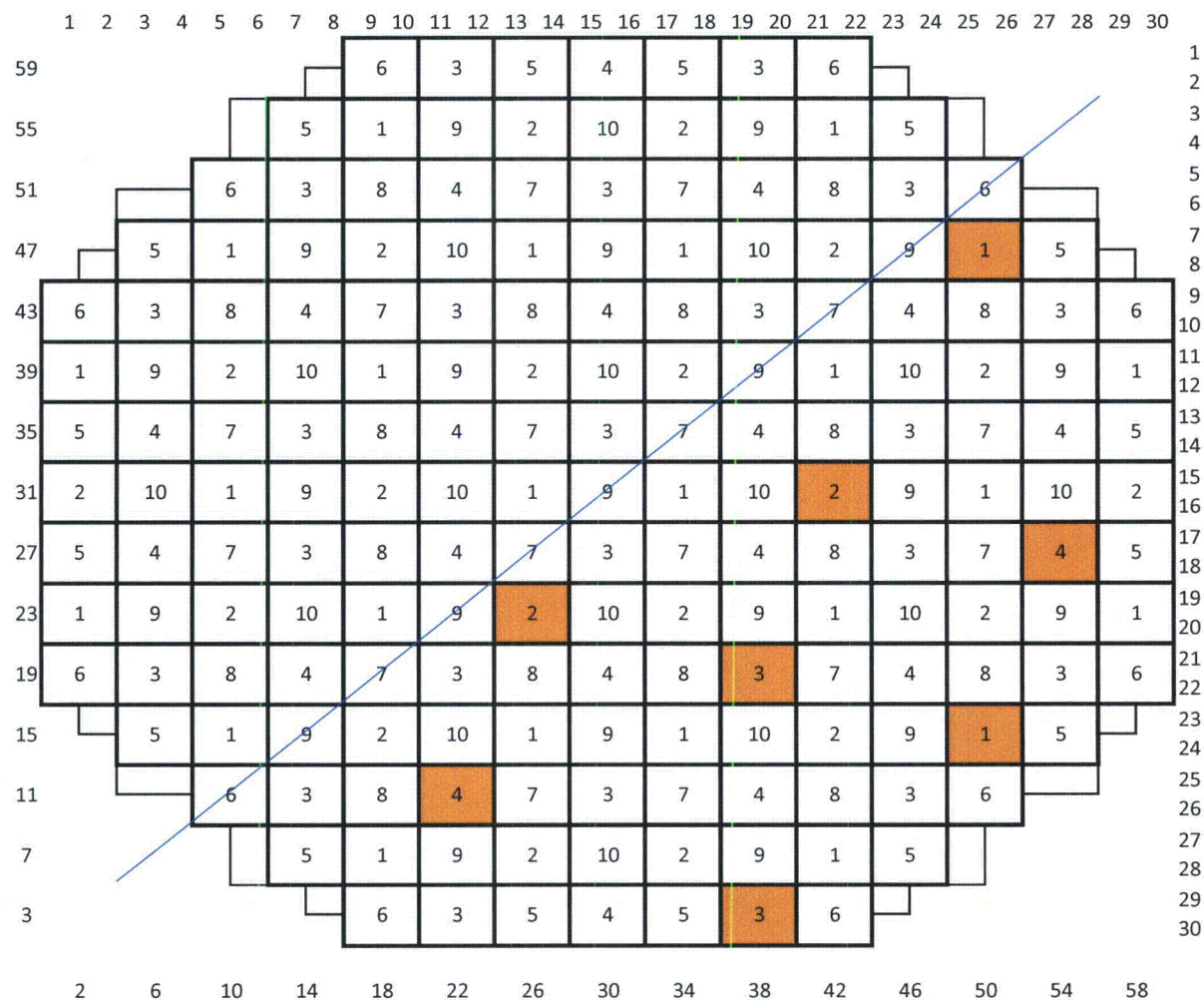


Figure 7.2 Example A-Sequence Groups with Inoperable Rod Locations

7.3.2 Power Range

For at power conditions, control rods are inserted and dropped to the current position of the control rod drive. This includes rods which are fully withdrawn. (The insertion and dropping of a rod would be similar to performing a sequence exchange and a rod of the prior sequence remains stuck in the core.) The rod drops are performed based on the licensing step through and utilizes the target rod patterns.

7.4 ***Selection of Candidate Rods***

7.4.1 Startup Range

The rod selection process for the startup range CRDA is based on the static worth of the control rods and a broad range at which criticality is assumed to occur. The reactor is assumed to be critical if the calculated k-eff for the initial rod pattern is within the range;

[

] Table 7.2 provides a summary of cold critical eigenvalue variation with the MICROBURN-B2 code system. Since the target is established based on prior cycles, the consistency of the MICROBURN-B2 results demonstrate that the selection criteria will bound the actual reactor criticality.

Table 7.2 Cold Eigenvalue Variation for Recent Cycles

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] for the CRDA in cycle

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Table 7.3 Upper Group Rods Bounded By Lower Group

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Following the establishment of the static rod worth values, the rods are identified for which detailed transient analyses are to be performed. The rod drop with the highest worth from BOC, PEAK, and EOFP are the first three rods selected. Selection of additional rods occurs after the transient evaluation of these three rods. The transient results from these first three cases are then used along with the static rod worth to define the most limiting in cycle condition. This is typically EOFP with top peaked reactivity distribution. [

] The outer ring evaluation is used to evaluate the impact of the CRDA on fuel rods with high exposure and cladding hydrogen content.

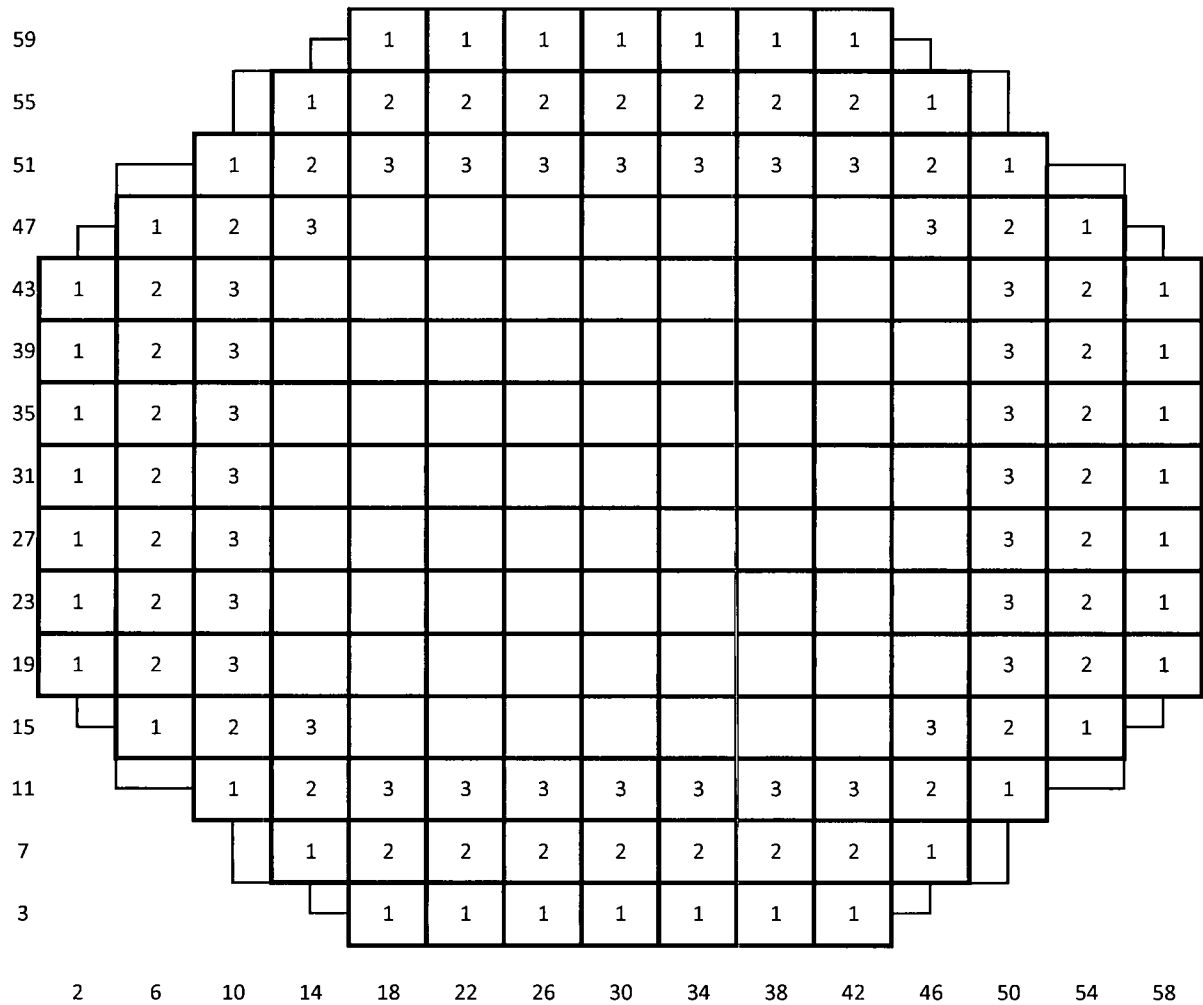


Figure 7.3 Example Control Rod Ring Locations

Table 7.4 Maximum Rod Worth (Δk) for Outer Rings

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7.4.2 Power Range

The change in steady state CPR is used for the selection criteria in the power range. The power range criteria are not burnup dependent therefore, only the rod drop with the largest change in steady state CPR is evaluated. These selections are made using the approved core simulator (MICROBURN-B2 Reference 17) and an NRC approved critical power correlation (References 33, 34, or 35).

7.5 ***Transient Evaluation Model***

The AURORA-B CRDA transient model maintains all capabilities of the AURORA-B base EM of Reference 9. The primary changes to the model include the expansion of processing cold library data, pin power reconstruction at cold conditions, and the inclusion of a peak rod heat structure. Due to the localized nature of the CRDA, a simplified plant system model is used.

The code structure, field equations, closure relations, and code numeric are the same as that described in Reference 9.

The thermal-hydraulic and thermal conduction field equations used within S-RELAP5 are the same as used in the RLBLOCA methodology described in Reference 18 and other USNRC

approved methodologies based on S-RELAP5. Extensive technical detail related to the field equations is provided in the S-RELAP5 theoretical description given in Reference 19.

The neutron kinetics equations contained within the MB2-K 3D kinetics model are described in detail within the MB2-K theoretical description (Reference 20). These field equations solve the transient three-dimensional and two-group equation set with six delayed neutron groups. The advanced nodal methods used in MICROBURN-B2 are also used in MB2-K to solve the spatial neutron flux distribution. A fully implicit numerical scheme is used in the temporal solution in which the neutron flux and the precursor density are factored into a fast varying exponential function and a smoothly varying component.

The additions to the AURORA-B code system (MB2-K and S-RELAP5) include:

- Pin power reconstruction at cold conditions.
- Inclusion of moderator temperature in cold cross section determination to support cold voided cross section feedback.
- Transfer of peak pin power from MB2-K to S-RELAP5.
- Transfer of moderator temperature from S-RELAP5 to MB2-K.
- Support a peak rod heat structure.

The peak rod heat structure and the cold pin power reconstruction allow the AURORA-B system to determine the peak pin enthalpy utilizing the realistic pin peaking determined throughout the transient. The pin power reconstruction at cold conditions is performed in the same manner as for hot operating conditions. This process takes into account the nodal surface fluxes and currents, average fluxes, and corner fluxes and flux gradients throughout the transient. The details of the pin power reconstruction are described in Reference 20 (MB2-K Theory Manual).

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7.5.1 External Data Transfer

External data transfer is an important element of the procedures for treating the input information (particularly the code input arising from the assumed plant state at transient initiation). In addition, analysis of the target scenarios requires input from numerous data

sources and engineering disciplines because of the highly coupled neutronic / fuel thermal-mechanical / thermal-hydraulic nature of the events. As a result, the amount and content of data that will be transferred to the EM from external data sources is important in assuring accurate modeling of the BWR plant at the target initial conditions. A summary schematic highlighting important elements of the external data transfer is shown in Figure 7.4.

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Figure 7.4 External Data Transfer to AURORA-B EM

7.5.2 Coupling of Component Calculational Devices

Coupling of the component calculational devices to create the code system is important to assure the relevant event characteristics are captured in the EM. Modeling the dynamic relationship between the neutron-kinetics, fuel rod models, and core thermal-hydraulics models is important aspect in modeling the CRDA. The coupling between component calculational devices for the system calculation is illustrated in Figure 7.5.

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Figure 7.5 Coupling of Component Calculational Devices for CRDA

In the coupling, the 3D kinetics equations are simulating every fuel assembly in the core at the same level of detail used in the steady state core simulator. Modeling of the water channels/rods and the core bypass are also consistent with the modeling applied in the steady state core simulator.

Coupling from the thermal-hydraulic and thermal conduction equations to the 3D kinetics equations requires passing the active channel, core bypass, the water channel/rod moderator densities, moderator temperature, and the Doppler effective fuel temperature to the 3D kinetics

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7.5.3 Fuel Grouping

The CRDA scenario is a relatively localized event around the target rod. The individual fuel assemblies in either [] around the target rod are explicitly modeled as individual channels. Both an average fuel rod and peak power fuel rod are evaluated for the individual channels. [

] Within the buffer rings the fuel assemblies are assigned in groups with similar geometric, hydraulic, thermal power, and fuel thermal properties. The remainder of the fuel assemblies in the core are assigned to groups with similar geometric, hydraulic, thermal power, and fuel neutronic properties.

Sensitivity studies were completed with the code assessment relative to channel grouping for the CRDA and are provided in Section 8.7.1.

When developing the fuel grouping assignments, excluding the assemblies around the target rod, an algorithm is applied that ensures all assemblies in a fuel group have identical geometric design, orifice design, nuclear design (e.g. identical nuclear lattice designs), and are part of the

same reload batch. Additional characteristics such as exposure, power distribution, and/or the initial CPR predicted by MICROBURN-B2 may be used to define even smaller groups if desired.

7.5.4 Time Step Control and Advancement

The time step control and advancement are consistent with the AURORA-B base EM of Reference 9. However, due to the magnitude of the CRDA reactivity insertion at [

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The time step sizes for the CRDA are provided in Table 7.5.

Table 7.5 Time Step Size for CRDA Evaluation

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7.5.5 Plant Model Nodalization

Modeling of the plant hydraulic components and heat structures is a mathematical mapping from the physical system to the computational framework of the EM. For the CRDA analysis, a core only model is used in the AURORA-B system. The primary impact of the nodalization on the CRDA is the number of individual channels included in the core model.

In order to achieve the above, detailed technical guidance has been developed (Reference 24) to define the framework under which standardized input models are prepared for the AURORA-B EM. Details for the AURORA-B CRDA EM model are provided in Reference 65. Specifically, the purpose of the technical guidance is to establish a consistent approach for the following parameters;

- Nodalization of hydrodynamic components, heat structures, and their connections, plus flow and pressure boundary conditions
- Modeling practices for components and processes (including selection of phenomenological code models)
- Control variable and trip definition, and their use

- Material properties
- Initialization of the components and structures

The following sections summarize the key aspects of the technical guidance for nodalization of hydrodynamic components and heat structures, as well as the modeling practices for key components and processes.

7.5.5.1 Core Region

The nodalization of the fuel assemblies, flow channels, water channels/rods, core bypass region, and associated heat structures are developed in a manner consistent with the steady state core simulator and fuel design data, except for consideration of “fuel grouping” as described in Section 7.5.3 and peak rod heat structures. [

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The core model components are identified in Table 7.6 and shown in Figure 7.6.

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Figure 7.6 Core Modeling Components

Table 7.6 Description of Reactor Core Components

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7.5.5.2 Plant Parameters Data

Plant parameters data are those parameters used in preparing input to the EM that describe the configuration of a given plant that could potentially change from cycle to cycle. A document defining the parameters is prepared jointly by AREVA and the utility customer for each plant and reviewed each fuel cycle. Every value is reviewed and accepted by the utility customer to ensure the most accurate data defining the plant configuration is used in AREVA analyses. This is particularly important to ensure AREVA is aware of values which have been superseded due to plant configuration changes, operation changes, or more recent information. Customer acceptance of the document validates the parameters within the document. Changes from prior cycles are typically highlighted to ensure analysts are aware of the changes.

The plant parameters data provides necessary information for the physical plant and any engineered features influencing plant performance as part of the "standard nodalization". The data are obtained from plant drawings, plant design data and specifications, component/equipment design data and specifications, plant system manuals, Technical Specifications, startup test data, plant operation data, and other similar sources.

The following list summarizes key plant parameters data used in developing input to the EM:

- Core thermal power, core flow, steam flow, dome pressure at rated conditions, and allowable power/flow map for off rated conditions.
- Safety system performance and set points including; scram set points for high neutron flux, high dome pressure, scram delays for the instrumentation, control rod insertion speeds.
- Allowed control rod sequences and inoperable control rod restrictions.

The approach for selecting specific values for plant parameters data that assure plant operations are bounded is described in Section 9.0 for the application methodology.

7.5.5.3 Individual Channel Model for CRDA

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7.6 ***Determination of Prompt and Total Enthalpy Response***

The specific acceptance criteria make a distinction between prompt and total enthalpy rise. If acceptable results can be obtained using total enthalpy, then total enthalpy rise may be used for the evaluation. However, if there is a significant difference between the prompt and total enthalpy, then both the prompt and total enthalpy may be used for event evaluations. The prompt enthalpy is the radial average fuel enthalpy rise at the time corresponding to one pulse width after the peak of the prompt pulse as defined on page 4.2-33 SRP4.2 Appendix B

(Reference 6). [

] of the transient is also illustrated in Figure 7.8 which contains the local peaking factors for all rods in a lattice. Each color represents a different time step in the transient. Prior to the rod drop, the lower rod numbers have higher peaking than the higher rods in the lattice. As the blade drops out the peaking flattens across the lattice. The peaking for all rods remains constant during the peak pulse at approximately 0.8 seconds. Therefore, the enthalpy rise during the power pulse is proportional to the peaking factors during the time of the peak power. A linear adjusted pin peaking factor distribution is obtained by increasing the average enthalpy by [] and leaving the peak constant.

[

]

The application of this process is illustrated in Figure 7.9. The nominal values identified in Figure 7.9 assumes linear interpolation and extrapolation based simply on the peak and average enthalpy increases whereas the enhanced values are based on the increased average value.

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Figure 7.7 Pin Peaking Factor Data for Select Rods and Core Power

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**Figure 7.8 Pin Peaking Factor Distribution for All Rods with
Transient Time**

[

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Figure 7.9 Enthalpy Determination

Additional information on the pin peaking factor distribution during the transient is provided in Section 8.7.3.12.

7.7 *Evaluation of Event MCPR Response*

The single channel model is applied for evaluating the fuel assembly MCPR for the [] assemblies around the dropped rod location. The MCPR response is obtained from the AURORA-B system during the transient. Evaluations of rod drops were performed in the power range. The evaluation consisted of determining the rod which when dropped would have the largest impact on CPR with the static calculation. All rods in one quadrant of the core were evaluated to determine the impact on CPR. If the rod was withdrawn, it was inserted and then dropped. The rods which had the largest impact on the steady state CPR were then evaluated with the AURORA-B system.

The power pulse in the power range is much broader than that in the startup range due to the presence of voids in the core as illustrated in Figure 7.10.

[

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(Reference 11 page 3-2).

Table 7.7 Power Range Minimum CPR

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Table 7.8 Rods in Boiling Transition versus CPR Safety Limit EOC

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Table 7.9 Rods in Boiling Transition for SL=1.0 Near BOC

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Figure 7.10 Power Pulse for CRDA in Power Range

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Figure 7.11 Power Range CPR Response CRDA

7.8 ***Evaluation of Event Pressure Response***

The most severe power pulse for the CRDA occurs at the cold conditions at atmospheric pressure. Therefore, a significant amount of energy is required to simply pressurize the core. Although local pressure increase occurs within a few assemblies, there is little change in the core system pressure in either the startup range or the power range.

The pressure response in the lower plenum for rod drops in the power range are provided in Figure 7.12 and the pressure response for a rod drop in the startup range is provided in Figure 7.13.

Therefore, the CRDA would not result in a reactor pressure which would cause increased stress to exceed the "Service Limit C" ASME Boiler and Pressure Vessel Code.

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Figure 7.12 Power Range Lower Plenum Pressure Response

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Figure 7.13 Startup Range Lower Plenum Pressure Response

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7.9 Evaluation of Radiological Consequences

The direct determination of the radiological consequences is not addressed in this LTR. However, the method to determine the appropriate release fractions and fission product inventory is provided which can then be used in the evaluation against the source term used in the plants licensed radiological evaluation for the CRDA.

7.9.1 Nodal Rod Release Fraction

The release term for RIA includes an enthalpy dependent term. Therefore, the evaluation will be more intrusive/sophisticated than simply evaluating the number of failed rods. If a rod is determined to have failed as result of the CRDA, then the transient fission gas release term is determined for each failed rod.

Steady state release fractions will be utilized consistent with Regulatory Guidance and requirements. The steady state release terms are provided from Reference 12 and 13. For the AREVA CRDA methodology the steady state release terms are based on the appropriate steady state release term of Reference 12 and 13. The steady state gap release fractions are provided in Table 7.10. Transient enthalpy dependent release fractions from SRP 4.2 Appendix B

(Reference 6) are provided in Table 7.11. A revised set of release fractions for the CRDA event are provided in Reference 45. The total release fractions for the BWR CRDA are provided in Table 7.12 based on SRP 4.2 Appendix B and Reference 45 (PNNL-18212 Revision 1).

(It is recognized that the potential revision of the release fractions may occur as indicated in References 44 and 45. Upon adoption of revised release fractions, the AREVA methodology will incorporate release fractions consistent with the plant licensing basis. The inclusion of the values from Reference 45 are provided for example.)

The final release fraction for a failed rod is steady state release fraction and the average of the nodal transient release fractions.

$$RRF_i = SSRF_i + \frac{\sum TRF_i(\Delta H_k)}{N} \quad (4)$$

Where;

RRF_i the rod release fraction for isotope i including steady state and transient release

$SSRF_i$ the steady state release fraction for isotope i

$TRF_i(\Delta H)$ the enthalpy dependent transient release term for isotope i

ΔH_k the enthalpy increase at node k

Table 7.10 Steady State BWR Fuel Rod Peak Gap Release Fractions

Isotope	Current RG 1.183 Table 3 (Reference 12)	Current RG 1.195 Table 2 (Reference 13)
I-131	0.08	0.08
I-132	0.05	0.05
Kr-85	0.10	0.10
Other Nobles	0.05	0.05
Other Halogens	0.05	0.05
Alkali Metals	0.12	-

NOTE: I-132 is set to the release fraction of other Halogens

Table 7.11 Transient BWR Fuel Rod Gap Release Fractions

Isotope	SRP 4.2 Appendix B (Reference 6)	
	%	Fraction
All	$\text{Max}(0, 0.2286 \cdot \Delta H - 7.1419)$	$\text{Max}(0, 0.002286 \cdot \Delta H - 7.1419 \text{E-}2)$

Table 7.12 Local Gap Release Fractions for BWR CRDA

Isotope	SRP4-2 Appendix B	PNNL-18212 Revision 1
I-131	$0.08 + \max(0, 0.00286 \cdot \Delta H - 7.1419E-2)$	$0.08 + 0.00073 \cdot \Delta H$
I-132	$0.05 + \max(0, 0.00286 \cdot \Delta H - 7.1419E-2)$	$0.09 + 0.00073 \cdot \Delta H$
Kr-85	$0.10 + \max(0, 0.00286 \cdot \Delta H - 7.1419E-2)$	$0.38 + 0.0022 \cdot \Delta H$
Other Nobles	$0.05 + \max(0, 0.00286 \cdot \Delta H - 7.1419E-2)$	$0.08 + 0.00073 \cdot \Delta H$
Other Halogens	$0.05 + \max(0, 0.00286 \cdot \Delta H - 7.1419E-2)$	$0.05 + 0.00073 \cdot \Delta H$
Alkali Metals RG 1.183	$0.12 + \max(0, 0.00286 \cdot \Delta H - 7.1419E-2)$	$0.50 + 0.0031 \cdot \Delta H$
Alkali Metals RG 1.195	$0 + \max(0, 0.00286 \cdot \Delta H - 7.1419E-2)$	

7.9.2 Nodal Rod Fission Product Inventory

The nodal fission product inventory is assumed to be uniform over the length of the entire fuel rod and is determined in a manner consistent with Section 3.1 of Reg Guide 1.183 (Reference 12) or Section 3.1 of Reg Guide 1.195 (Reference 13). The fission product inventory is based on the maximum allowed power and the maximum exposure at the time of the event.

7.9.3 Fission Product Inventory Released

The total fission product inventory released is determined by the summation of the individual products of the release for each failed rod.

7.10 ***Evaluation of Core Coolability***

There are four specific requirements for core coolability contained in SRP 4.2 Appendix B. This section describes how the coolability criteria are addressed and complied with in the AREVA CRDA methodology.

7.10.1 Peak Radial Average Fuel Enthalpy

The fuel enthalpy is calculated with the AURORA-B code system and the total enthalpy is compared with the 230 cal/g limit. The total enthalpy is combined with an enthalpy uncertainty for the AURORA-B code system to demonstrate the 230 cal/g limit is not exceeded.

7.10.2 Peak Fuel Temperature

The peak fuel temperature is calculated with the AURORA-B code system. The resulting temperature is then compared with the melting temperature for the UO₂. The fuel melting temperature utilized for failure determination will be [] rods (Section 7.1.8 Figure 7.16 Reference 22).

7.10.3 Mechanical Energy

The typical power pulse width for the BWR CRDA is estimated between 45 and 75 ms for cold zero power conditions and 45 and 140 ms for hot zero power conditions (Table 1 of Reference 47). A review of the experimental data was provided in Reference 48 and 49. Table 7.13 provides a summary of test results presented in Appendix A of Reference 47. No UO₂ fuel experienced dispersal for tests with pulse widths above 10 milliseconds and enthalpy deposits below 230 cal/gram. []

] See Table 3.1 Interim

Criteria Summary, Page 3-2 of this document.

Table 7.13 RIA Test with Failures (Below 230 cal/g)

Test	Burnup	Pulse Width	Peak Enthalpy		Failure Enthalpy		Fuel Loss
	GWd/MT	ms	J/g	cal/g	J/g	cal/g	
PBF RIA 1-2	4.4	16	775	185	586	140	No
NSRR JM-4	21	5.5	743	177	743	177	No
NSRR JMH-5	30	6.2	910	217	790	189	Yes
NSRR JM-5	26	5.6	697	167	682	163	No
NSRR TK-2	48	4.4	448	107	251	60	Yes
NSRR HBO-1	50	4.4	306	73	251	60	Yes
NSRR HBO-5	44	4.4	335	80	322	77	Yes
NSRR TK-7	50	4.3	398	95	360	86	Yes
REP Na-8	60	75	410	98	184	44	No
NSRR FK-6	61	4.3	548	131	293	70	Yes
NSRR FK-7	61	4.3	540	129	260	62	Yes
NSRR FK-9	61	5.7	377	90	360	86	Yes
NSRR FK-10	61	5.1	430	103	335	80	Yes
NSRR FK-12	61	5.8	373	89	301	72	Yes
REP Na-10	63	31	410	98	338	81	No
REP Na-1	64	9.5	481	115	117	28	Yes
LS-1	69	4.4	469	112	222	53	Yes

7.10.4 Coolable Geometry

The design of the BWR fuel assembly and channel helps mitigate the loss of coolable geometry from the CRDA. Each BWR fuel assembly is channeled such that flow enters the bottom of the assembly and exits the top. The BWR fuel assembly includes axial spacers or grids which maintain the positions of the fuel rods within the assembly. Bypass coolant flows around the assembly channel and through the internal water rods/channels within the fuel assembly. Modern BWR assemblies utilize part length rods which provide an increased flow area in the upper portion of the fuel assembly. This increase in flow area reduces the two-phase pressure drop in the top of the assembly during power production.

The bypass flow and coolant would not be affected by the CRDA. Prior to the control rod dropping, the fuel adjacent to the control rod is either producing no power, or less than core average since it is controlled. []

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7.11 ***Methodology Uncertainty***

The uncertainty for the AREVA CDRA methodology is determined in a manner similar to that used for the AURORA-B base EM. In that work, the treatment of biases and uncertainties was divided among three classes. The first were those related to the evaluation model structure relative to the interactions among the Component Calculations Devices (CCD's) and time step sensitivities. The second was the selection of plant parameters and initial conditions. The third and final were those involved in the prediction of highly ranked PIRT entries. The actual evaluation of the uncertainty is provided in Section 8.7.

7.11.1 Evaluation Model Structure

For the coupling between the computational tools and the time step specification, the time steps and CCD interactions are limited to those which do not vary, except in a trivial way. Thus, for example, the time step size for a particular transient is selected such that further reductions in that size results in virtually no changes (achieved convergence) in the results produced by the evaluation model. Table 7.14 identifies the model interactions that are evaluated. (The results of the model structure evaluation are provided in Section 8.7.1 of this document.)

7.11.2 Plant Parameters & Initial Conditions

Plant parameters are generally a mixture of realistic and conservative values. While the conservative values may generally be relied upon for use, the evaluation model may be exercised in sensitivity studies to determine appropriate values for parameters available as realistic values. Initial conditions are selected to be appropriate to the accident or transient, but are treated in the same fashion as realistically specified plant parameters to ensure that the results are appropriately conservative or insensitive to variations in the input values.

7.11.3 Highly Ranked PIRT Entries

For highly ranked PIRT entries sensitivity studies are performed and presented in Section 8.7.3 of this document. For these studies, the effect of the propagation of these perturbations and the resulting uncertainties are quantified. Based on these results, suitable treatments of these entries are formulated to ensure that net conservative FoM results are produced.

Table 7.14 Evaluation Model Structure Interactions

Model Element	Comment
Time step sensitivity	The time step sensitivity will be repeated for CRDA analysis.
Hydraulics to Kinetics Coupling Scheme	This represents the computational coupling between the two major Component Computational Devices for the methodology
Core Flow Distribution	The flow distribution will be evaluated for application at startup conditions.
Fuel Channel Grouping	This includes the number of fuel assemblies that are represented explicitly around the dropped rod and the grouping of the remainder of the assemblies within the core.

8.0 **Assessment Results**

8.1 ***Regulatory Basis***

From SRP 15.0.2 II.1.A and B, a comprehensive description of the code assessment is required along with the determination of the code uncertainty for a sample plant accident calculation.

A summary of the code assessment data base is presented in Section 6.0. The results of the code assessment are provided within this Section. The code uncertainty basis are also determined within this Section while the actual determination of the uncertainty for a sample plant accident calculation is presented in Section 9.

8.2 ***AURORA-B CRDA Assessment Results***

A comprehensive description of the AURORA-B base EM code system assessment is provided in ANP-10300P (Reference 9). The additional assessments of the AURORA-B system for the analysis of the CRDA are presented in this section along with the results of sensitivity studies.

The base assessment criteria were established for the AURORA-B base EM code system and are applied for the AURORA-B CRDA methodology. The primary scope of the assessment for the AURORA-B CRDA is to address modeling of the CRDA in the startup range.

8.3 ***AURORA-B CRDA Model Fidelity or Accuracy Assessment***

This Model Fidelity Assessment for the AURORA-B Base EM is presented in Section 6.2 of Reference 9. Following are the assessment conclusions from Reference 9.

8.3.1 Rod Bundle Void Tests and Christensen Void Tests

The assessment data base includes test from different facilities and provides void information for a wide range of system pressures (400 to 1260 psia) and flow conditions that cover the typical operating BWR conditions. These assessments show excellent code-data agreement of the given tests for void distributions. The results are applicable for modeling phenomena associated with void generation for the CRDA in the power range. Void sensitivity studies are provided in Section 8.7.3.2 for the start-up range.

8.3.2 Summary of MICROBURN-B2 Qualification

A summary of the MICROBURN-B2 qualification is provided in Section 6.2.5 of Reference 9 for the AURORA-B system. This summary is based on the complete qualification of the NRC approved CASMO4/MICROBURN-B2 methodology (Reference 17). The MICROBURN-B2 qualification assessment summary as provided in Reference 9 is applicable for the CRDA in the power range.

In addition to the operating power conditions, the CASMO4/MICROBURN-B2 methodology includes extensive qualification against cold critical measurements for numerous reactor startups. From this it is inferred that the CASMO4/MICROBURN-B2 core simulator provides excellent prediction of []

8.3.3 Summary of RODEX-4 Qualification

The summary of the RODEX-4 qualification as provided in Section 6.2.6 of Reference 9 is applicable for the CRDA in the power range. The fuel rod history effects are independent of the event initial conditions. Therefore, the assessment of the RODEX-4 codes and models remains applicable for the CRDA.

8.4 ***AURORA-B CRDA Model Field Equation and Numeric Solutions Assessment***

The assessment of the field equations and numerical techniques presented in Reference 9 is directly applicable for the CRDA analysis. The numerical kinetic benchmark results from the 2-D TWIGL seed-blanket reactor problem (Reference 30), the LMW 3-D PWR delayed critical transient problem (Reference 31) and the LRA 2D and 3D BWR Control Blade Drop Transients (Reference 32) are presented in Section 6.4 of Reference 9. The results predicted by MB2-K showed excellent agreement with other published results for all of the problems.

8.5 ***AURORA-B CRDA Model Applicability to Simulate System Components***

The assessment of rod bundle pressure drop and critical power test is presented in Section 6.5 of Reference 9 along with other components. The pressure drop and the critical power assessments from Reference 9 are applicable for the rod drop in the power range.

8.6 ***AURORA-B CRDA Model Integral Tests***

8.6.1 Peach Bottom Turbine Trip Tests

The test results from the Peach Bottom Turbine Trip tests (Reference 37 and 39) are used to assess the global capability of the AURORA-B EM in predicting the neutron kinetics response to typical pressurization events. In addition, the Peach Bottom tests provide sufficient information for validating the capability of S-RELAP5 in predicting several phenomena in dynamic scenarios. This assessment is provided in Section 6.6.2 of Reference 9.

Direct measurement of the indicated PIRT phenomena cannot be made in the tests, so the validation is performed via measured and calculated pressure and LPRM response of the plant in three tests. The validation shows reasonable to excellent code-data comparisons of pressure and LPRM response. From this it is inferred that the EM makes excellent predictions of the indicated PIRT phenomena. In addition, the global capability of the EM to predict the events is shown to be excellent. The specific phenomena that are included in the Reference 9 assessment which are [

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8.6.2 SPERT III RIA Tests

The Special Power Excursion Reactor Test III (SPERT III) reactor was a small pressurized-water, nuclear research reactor. The reactor was constructed as a facility for conducting reactor kinetic behavior and safety investigations. The facility description is provided in Reference 42 and the descriptions of the tests are provided in Reference 43. This section presents the results of the code assessment against the selected SPERT III experiments. The SPERT III initial experiment conditions represented cold startup, hot startup, hot standby, and hot operating conditions as listed in Table 8.1. It is noted that the average linear heat generation rate (LHGR) for the SPERT III core at operating power is close to that of commercial BWR reactors.

Table 8.1 SPERT III E-Core Tests Conditions

Range	°F	MWT	Avg Kw/ft
Cold Startup	70	5e-5	-
Hot Startup	260, 500	5e-5	-
Hot Standby	500	1	0.23
Hot Operating	500	20	4.7

The actual core configuration of the SPERT reactor is quite different than that of commercial BWRs. The SPERT core is relatively small and the core consists of only fresh fuel. The actual control rods are a combination of flux trap and fuel follower. A single transient rod is located at the center of the core which can be rapidly ejected out of the core for the reactivity insertion.

The modeling of the SPERT core required assumptions with respect to some of the components and initial conditions. The initial positions for the control rods and the transient rod are not identified for each experiment. However, critical positions were specified for static conditions along with the control rod worth and transient rod worth.

8.6.2.1 SPERT III Static Conditions

The model for the SPERT core was generated with the CASMO4/MICROBURN-B2 codes system and the static calculations were performed and compared with the static results provided in the reports (References 42, 43, and 66). Comparisons of the control rod and transient rod worth are provided in Figure 8.1 and Figure 8.2. These figures show good agreement between the MICROBURN-B2 results and the reported rod worth values.

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Figure 8.1 SPERT III E Core Control Rod Worth

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Figure 8.2 SPERT III E Core Transient Rod Worth

8.6.2.2 Transient Simulation

Transient simulations for SPERT test case were completed for each of the operating ranges. The initial conditions are set and the rod positions are iterated on to match the power pulse.

The nodalization for the transient cases were set to SPERT bundle size. To generate the [

] The results of the transient cases are provided in
Figure 8.3 through Figure 8.9.

The results of the magnitude of the power peak and the time at which it occurred are provided in Table 8.2. All results obtained are within the published experimental uncertainties.

Table 8.2 SPERT III E-Core Tests Results

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Figure 8.3 Hot Operating Case 86 (\$1.17,19MW)

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Figure 8.4 Hot Standby Case 82 (\$1.29,1.2MW)

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Figure 8.5 Hot Startup Case 56 (\$1.04,50W)

[

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Figure 8.6 Hot Startup Case 58 (\$1.15,50W)

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Figure 8.7 Hot Startup Case 29 (\$1.10,50W)

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Figure 8.8 Cold Startup Case 41 (\$1.13,50W)

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[

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Figure 8.9 Cold Startup Case 43 (\$1.21,50W)

Assessment Conclusions

The AURORA-B system can model the SPERT reactivity events within the uncertainty of the published experiment results. If detailed input conditions were known, the assessment would be considered an excellent agreement. Due to the lack of sufficient detail of the experiments, an explicit assessment of the model cannot be obtained. However, assessment conclusions are implied from the modeling of the SPERT experiments.

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The range of temperatures and power levels evaluated are consistent with BWRs from cold shutdown to hot operating conditions. The AURORA-B codes system predicted the results of the selected SPERT-III test within the published uncertainties.

8.7 ***AURORA-B CRDA Model Biases and Uncertainties***

The determination of the AURORA-B CRDA Model Biases and Uncertainties is provided in this section following the process provided in Section 7.11. The summary of the model biases and uncertainty is provided in Section 8.7.4.

8.7.1 Evaluation Model Structure

The impact of the specific elements of the AURORA-B system for power conditions is provided in Section 6.8.2 of Reference 9. This section reviews the model impact with respect to the CRDA. The evaluation and conclusions are summarized below:

Time Step Sensitivity

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Hydraulics to Kinetics Coupling Scheme

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Core Flow Distribution

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Fuel Channel Grouping

The reference analyses for the sample BWR-4 plant evaluated the impact of the channel grouping around the target control rod in the startup range. The grouping consisted of the explicit target array and the buffer region size. The results from the channel grouping are provided in Table 8.3 and shown graphically in Figure 8.10, Figure 8.11, and Figure 8.12. The nomenclature [

]

The core peak power decreases as the effective number of channels increases. The decrease nearly flattens at [

] channels and above.

When only [

] This generates lower fuel temperature (less Doppler feedback) and less moderator heating/voiding in these adjacent assemblies.

Based on these results [

] area is acceptable.

[

Table 8.3 Fuel Channel Grouping Results

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Figure 8.10 Peak Core Power versus Channel Grouping

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Figure 8.11 Prompt Enthalpy Rise versus Channel Grouping

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Figure 8.12 Prompt Enthalpy Rise versus Channel Grouping

8.7.2 Plant Parameters & Initial Conditions

The impact of plant parameters and initial conditions are presented in this section.

8.7.2.1 Scram Speed

For the CRDA, the power pulse is turned around prior to the actual movement of the control blades. Therefore, use of conservative delay times and SCRAM speed is appropriate for the CRDA analysis.

8.7.2.2 Control Rod Drop Velocity

For BWRs 2 through 6, the control rod drop velocity is maintained with the control rod velocity limiter. Based on the results of velocity limiter tests, a conservative maximum rod velocity of 3.11 ft/s is used from the Reference 10 Velocity Limiter Tests. This was determined to be a bounding conservative velocity for BWRs 2 through 6. The use of the conservative drop velocity will be used for BWRs 2 through 6.

For BWRs without velocity limiters, an appropriate drop velocity must be determined for the application of this methodology.

8.7.2.3 Initial Coolant Temperature

Sensitivity studies were completed with respect to the impact of the initial coolant conditions on the CRDA figures of merit. In the startup range, the largest impact on the enthalpy FoM was observed using the cold startup conditions. Comparisons of the enthalpy response versus temperature were made for two rod drops one rod at peak reactivity and the other at end of full power conditions.

The end of cycle rod drop 12 was used in the sensitivity study consistent with the other sensitivity studies. The results for fuel group 3 of rod drop 12 are shown in Figure 8.13 and Table 8.4. The rod 127 drop is at peak reactivity and the response is similar to that of rod 12 as indicated in Figure 8.14.

For both rods, the largest enthalpy response occurred starting from the lower temperature conditions. Utilization of 68°F for the initial temperature provides bounding conditions for higher temperatures.

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Figure 8.13 Peak Enthalpy Response Rod 12 at End of Full Power

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Table 8.4 Core Initial Temperature Sensitivity Rod 12 EOFP

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Figure 8.14 Peak Enthalpy Response Rod 127 at Peak Reactivity

8.7.2.4 Initial Core Power Level

Sensitivity studies were performed with respect to the power level. Relative to the reference case, the magnitude of the event decreased as the initial power was increased.

Table 8.5 Core Initial Power Sensitivity

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8.7.2.5 Initial Core Flow

Sensitivity studies were performed with respect to the coolant flow rate and the results are provided in Table 8.6. As the core initial core flow increase the peak power and prompt enthalpy rise decrease. The total enthalpy rise increases as the initial core flow increases. The typical startup conditions for BWR plants are carried out with both recirculation pumps at minimum speed (Reference 60 page 6.2.6, Reference 61 Section 4.4.3.1.2 Page 4.4-11, Reference 62 Section 4.4.3.3.1 Page 4.4-10). With the recirculation pumps at minimum pump speed the core flow would be greater than 10%. Therefore, use of an initial core flow of 10% is appropriate for the determination of the prompt enthalpy rise for startup conditions.

Table 8.6 Core Initial Flow Sensitivity

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8.7.2.6 Fuel Rod Power History

The AURORA-B methodology utilizes nodal average powers in constructing the rod power history effects. To assess the impact of this, rod power histories were interchanged for six fuel channels as indicated in Table 8.7. The core contained three reload batches. The sensitivity to the rod power history was evaluated by switching the power history profiles for different channels. A summary of modified gap properties are provided in Table 8.8.

The impact of the exchange of the power history files is provided in Table 8.9

Table 8.7 Power History Exchanges

Channel	Power History Modification
1	2nd cycle power history replaced with 3rd cycle power history
2	2nd cycle power history replaced with 1st cycle power history
3	1st cycle power history replaced with 3rd cycle power history
9	3rd cycle power history replaced with 2nd cycle power history
10	3rd cycle power history replaced with 1st cycle power history
11	1st cycle power history replaced with 2nd cycle power history

Table 8.8 Summary of Gap Properties

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An increase in the prompt enthalpy rise can occur when the power history for lower exposed assembly is used for that of a higher exposed assembly. This indicates that the use of lower exposed and correspondingly lower powered rod history will produce a larger enthalpy rise.

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for the evaluation of the CRDA. The impact on the enthalpy responses are provided in Figure 8.15, Figure 8.16, and Figure 8.17.

Table 8.9 Power History Exchange Results

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Figure 8.15 Fuel Group 1 and 2 Enthalpy Sensitivity to Power History

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Figure 8.16 Fuel Group 3 and 9 Enthalpy Sensitivity to Power History

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Figure 8.17 Fuel Group 10 and 11 Enthalpy Sensitivity to Power History

8.7.2.7 Allowed Rod Withdrawal Sequence

The plant specific rod withdrawal sequence affects the maximum dropped rod worth. Each plant utilizes a specific control rod pattern sequence control system which minimizes the worth of the control rods for the CRDA. The Banked Position Withdrawal Sequence (Reference 15) is typically used for controlling the allowed startup sequences. The analyzed withdrawal sequence must be consistent with the plant limitations. To conservatively bound rods in higher order groups, the pull sequence can be analyzed with each higher order group being the second group pulled.

8.7.2.8 Inoperable Control Rods

Each plant's Technical Specifications allow for inoperable control rods. Since the actual location of the inoperable control rods would not be known at the time of core licensing process, a conservative assignment of the inoperable control rods is assigned for a plant. Assigning the inoperable control rod to one half of the core resulted in the largest increase in rod worth for the remaining control rods.

8.7.3 Analysis of Biases and Uncertainties from Highly Ranked PIRT Phenomena

The impact of propagating the biases and/or uncertainties of the highly ranked PIRT phenomena through the EM was investigated through a series of sensitivity cases.

8.7.3.1 Control Rod Worth

The actual worth of a control rod is dependent on the actual control blade design, the fuel cycle design, cycle lifetime, and initial conditions. The methodology assumes that the rod drop is always the first rod in a group. This is conservative in that the worth of the dropped rod is increased by assuming it is the first rod to drop in a group. The rod worths were tabulated and ranked for each potential withdrawal sequences through banking of the third group at notch 12. The results shown in Table 8.10 show that the highest worth rod for a given sequence is more [] the rod drops in the sequence. Eight different pull sequences were evaluated, four for the A sequence rods and four for the B sequence rods. The sequence identifier indicates if it is an A or B sequence and then the order in which the rod groups are pulled. (See Figure 8.18 for an illustration of all the rod pulls and their respective worths for one sequence.) Utilization of the highest worth rods ensures that the potential rod drops are bounded for criteria which are not burn up dependent.

Table 8.10 Rod Worth Ranking with Third Group Banked at 4, 8, and 12

[

[

]

Figure 8.18 Control Rod Drop Worth for Sequence A1234

The actual uncertainty of the cold rod worth is determined in the following manner to address the actual blade worth uncertainty, and the modeling of the blade worth uncertainty.

Control blades designed by both Westinghouse and GEH are designed to match original equipment blades within $\pm 5\%$ (Reference 51 page 7-3 Section 7.2.4, and Reference 52 page 10 Section 3.2.1). As such the uncertainty for the blade worth is 5%. As the control blade is irradiated, there is uncertainty associated with the depletion of the absorber material. For the CRDA, only beginning of life (maximum) blade worth is used. Therefore, no uncertainty is assigned to control blade depletion for the CRDA.

A modeling uncertainty for control blade worth was established in Reference 17 [

] The control blade worth uncertainty is summarized in Table 8.11.

Table 8.11 Control Blade Worth Uncertainty

[

]

The evaluation of the impact on the enthalpy response due to a change in rod worth was evaluated. Figure 8.19 shows the relation between fuel enthalpy and static rod worth to be fairly linear, consistent with Reference 46. [

]

Data were then evaluated over additional exposure points and temperatures and are shown in Figure 8.21 [

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[

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Figure 8.19 Peak Enthalpy versus Control Rod Worth EOFP

[

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Figure 8.20 Relative Enthalpy versus Relative Rod Worth EOFP

[

]

Figure 8.21 Relative Enthalpy versus Relative Rod Worth

8.7.3.2 Moderator Feedback

For the CRDA, moderator feedback is the change in the reactivity feedback from moderator temperature and density changes in the active, by-pass and water channels/rods. This also includes the changes in void fractions in the coolant. These changes are a result of direct energy deposition to the coolant and heat transfer from the cladding. Both conditions are specifically modeled in the 3-d kinetics code with thermal hydraulics feedback.

The moderator feedback is a fundamental part of the cross sections. The sensitivity to the moderator density change during the transient was evaluated by increasing and decreasing the density feedback for both the active channel flow and the bypass flow. The results of the sensitivity studies (Table 8.12 and Table 8.13) show that the variation in the moderator feedback has insignificant impact on the prompt fuel enthalpy rise. The moderator feedback does have a minimal effect on the total enthalpy rise. Based on the results of the sensitivity studies an uncertainty of [] on the enthalpy is used for the moderator feedback.

Table 8.12 Active Channel Moderator Density Feedback

[

]

Table 8.13 Bypass Channel Moderator Density Feedback

[

]

8.7.3.3 Fuel Temperature Feedback

Fuel temperature feedback is the reactivity feedback from fuel temperature changes. The coupling of the kinetics and the thermal hydraulics codes results in the use of fuel temperature determined on the average rod rather than the peak rod for the given lattice. A [] perturbation is utilized for the Doppler feedback based on Reference 56 and 63. The [] perturbation on Doppler reactivity feedback is chosen [

]. The impact of the CRDA is approximately inversely proportional to the Doppler feedback as indicated in Table 8.14. The sensitivity study shows that Doppler feedback is very important for the CRDA event and that the change in enthalpy is consistent or slightly more than the change in the Doppler feedback. An uncertainty of [] will be used for the enthalpy rise based on the Doppler uncertainty.

Table 8.14 Fuel Temperature Feedback

[

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8.7.3.4 Effective Delayed-Neutron Fraction

The total delayed neutron fraction (β_{eff}) determines when prompt criticality is reached. The effective delayed neutron fraction varies depending on the heavy nuclide inventory or the nodal exposure. The neutronics data libraries generated, for each lattice represented in the AURORA-B system contain the delayed neutron fraction data for six groups tabulated as a function of exposure and spectral history. A node-specific delayed neutron fraction is obtained for each node by interpolating on the table with available nodal exposure and spectral history.

A sensitivity study was performed by modifying the cross section data libraries to increase and decrease the delayed neutron fraction. The effective delayed neutron fraction was increased and decreased by [] A reference transit calculation was run and the impact on the peak enthalpy is inversely proportional to the change in the total delayed neutron fraction as shown in Table 8.15. An uncertainty of [] is utilized for the total delayed neutron fraction consistent with Reference 50 (page 7-2 Section 7.1.4). Based on the sensitivity study, an uncertainty of [] is determined for the enthalpy rise.

Table 8.15 Delayed Neutron Fraction Sensitivity

[

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8.7.3.5 Reactor Trip Reactivity

Reactor trip reactivity is the negative reactivity associated with insertion of control rods after receipt of a reactor trip signal and the rate at which it is inserted. Since the actual power pulse is turned around by Doppler prior to the start of motion of the rods being inserted, the rate of SCRAM and delay time are based on minimum Technical Specifications requirements.

With the exception of the dropped rod, all other rods are inserted by the scram and the core is in a sub-critical state which terminates the event.

By utilizing the minimum technical specification scram times, no uncertainty is assigned to the reactor trip reactivity.

8.7.3.6 Fuel Cycle Design

The fuel cycle design includes those important design elements that determine the neutronic properties of the core at event initiation, such as the loading pattern, control history (control rod pattern, spectral shift), burnup, and exposure. The impact of the fuel cycle design is the key parameter in determining the control rod worth. Therefore, [

]

8.7.3.7 Heat Resistances in High Burnup Fuel, Gap, and Cladding

The resistances offered by the fuel, gap, and cladding to the flow of thermal energy from regions of high temperature to regions of lower temperature. The resistance is dependent upon the path length and thermal conductivity, which change with burnup.

Sensitivity calculations were performed for fuel thermal mechanical properties. The calculations were performed with adjusting the gap alone, and adjusting the gap coefficient. The gap width was increased and decreased to show the impact on heat transfer as indicated in Table 8.16. Sensitivity studies on the fuel conductivity were also performed by enhancing and reducing the conductivity by 20% as shown in Table 8.17. The fuel thermal conductivity uncertainty from the MATPRO model (Reference 55) is less than 10%. The actual impact on the fuel enthalpy is minimal and an uncertainty of [] is assigned to the prompt fuel enthalpy rise for heat resistances based on the results of the sensitivity studies.

Table 8.16 Gap Width Adjustments

[

]

Table 8.17 Fuel Heat Transfer Coefficient Sensitivity

[

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8.7.3.8 Transient Cladding-to-Coolant Heat Transfer Coefficient

The correlation that determines transport of energy at the interface by one or more of the following modes: forced convection-liquid, nucleate boiling, transition boiling, film boiling, or forced convection-vapor.

In the startup range, the rapid nature of the power burst, there is limited change in heat transfer during the power burst. In this range the failure criterion is enthalpy based. The heat transfer has little impact on the fuel enthalpy since the event is turned around by Doppler prior to any significant heat transfer from the rod.

Use of steady state heat transfer correlations have been shown to be conservative or appropriate for transient conditions (References 33, 34, 35, 53, and 54). Therefore, no uncertainty is applied for the transient heat transfer coefficient.

Reference	Location within Reference
33	Last part of Section 5.0 of SER top of Page 6
34	Page 60 Response to 19, ANP-10249Q1P
35	Page 8 of SER Section 3.3.4 and additional information which discusses References 53 and 54 is contained on Page 13 of ANP-10298Q3P Item 8 Round 2 RAI question 7.
53	Chapter IIA addresses use of steady-state correlations. Page 11: "Comparison with flow and power transient experiments indicated that the prediction was quite conservative."
54	Section 9.6.2, Page 405, addresses critical heat flux under transient conditions.

8.7.3.9 Heat capacities of Fuel and Cladding

The heat capacity of UO₂ and cladding are established based on the RODEX4 fuel properties established for the fuel rods. The temperature change during the prompt pulse is based on the deposited energy and the heat capacity. The variation of heat capacity of the UO₂ is only a function of temperature. The [

] which is considered a standard for defining heat capacity for UO₂. No error estimate or special treatment is used for the UO₂ heat capacity.

The actual heat capacity of the cladding does not impact the initial prompt power pulse since the event is turned around by Doppler prior to any significant heat transfer from the rod. Therefore, no error estimate or special treatment is used for the cladding heat capacity.

8.7.3.10 Direct Energy Deposition to Moderator

Direct energy deposition into the active channel fluid occurs as the fluid attenuates γ -rays and neutrons that escape the fuel rods. The amount of energy deposited in the fluid is a function of the local fluid density and locally generated fission power of the fuel. The partitioning of the energy deposition which is deposited directly in the fuel pellet and that deposited in the moderator and structural components may affect the reactivity feedback mechanism.

Sensitivity studies were performed with respect to the energy partitioning between fuel and moderator. The fraction of power deposited in the coolant and structural material was increased and decreased by modifying the energy deposition fractions in the cross section libraries. The energy deposition fractions to the coolant/moderator were increased and decreased by 20% in the cross section libraries. The CRDA evaluations were repeated with the revised cross section libraries. The results are provided in Table 8.18 The change in energy deposition to the moderator has little impact on the actual fuel enthalpy. An uncertainty value of [] is assigned for the impact on the peak enthalpy for this parameter.

Table 8.18 Heat Deposition Sensitivity

[

]

8.7.3.11 Pellet Radial Power Distribution

The radial distribution of the power produced in the fuel rod is determined with the RODEX4 (Reference 21) modules in the S-RELAP5 code system. Sixteen radial regions are utilized to predict the power and temperature profile for the pellet.

Sensitivity studies were performed varying the coefficients for determining the Doppler effective temperature. Variations in the temperature weighting coefficients were performed to assess the impact on the effective Doppler temperature and the results are provided in Table 8.19. The default weighting for the Doppler effective temperature, [] are used as the reference in the sensitivity study.

Based on the results of the sensitivity study a [] uncertainty is assigned to the enthalpy rise.

Table 8.19 Doppler Effective Temperature Coefficient Sensitivity

[

]

A Pellet average temperature weighting factor

B Pellet surface temperature weighting factor

C Pellet centerline temperature weighting factor

8.7.3.12 Rod-Peaking Factors

Rod-peaking factors represent the nodal rod or pellet power distribution within an assembly during the transient. The MB2-K code utilizes pin power reconstruction to update the local rod peaking factors for each time step of the transient. As the rod is withdrawn there will be a significant change in the pin peaking factors from the initial controlled state. However, the actual peaking factors are fairly constant during the power pulse as illustrated in Figure 7.7 and Figure 7.8. The inclusion of the pin power reconstruction based on the flux reconstruction model provides an accurate estimate of local peaking during the transient.

The assessment of the methodology pin power reconstruction and pin peaking factor determination is included within Reference 17 (CASMO/MICROBURN-B2 Methodology). The same methodology for pin power reconstruction is employed within the MB2-K system. The local peaking uncertainty was determined based on [

]

[] (from Reference 17 page 2-6) is applicable for the control rod drop evaluation since the same methodology is used to construct the pin power distribution.

8.7.3.13 Cladding Hydrogen Content

The content of hydrogen in the cladding at the start of the transient is used in the determination of the failure criteria. This does not have an impact on the actual kinetic response, but impacts the fuel cladding failure threshold for PCMI. The hydrogen model is a best-estimate nodal hydrogen model and no uncertainty is applied. From page 12 of Reference 49, "a best-estimate nodal hydrogen concentration is judged sufficient to address the local cladding properties."

8.7.3.14 Power Distribution

The total core power and core power distribution at the start of the event. The power distribution during the transient is driven by the local rod worth and fuel characteristics. The nodal power uncertainty [

] The nodal power distribution is reflected in the nodal isotopic concentration which is used in the determination of the cross sections for the transient calculation. Therefore, the CASMO4/MICROBURN-B2 power distribution uncertainty is applied for the rod drop.

The initial power level uncertainty is addressed in the startup range by selecting conservative initial conditions (Section 8.7.2.4). As such there is no specific uncertainty on total core power.

8.7.3.15 Coolant Initial Conditions

The coolant initial conditions includes pressure, temperature and the flow distribution, active channel, bypass, and water rod flows at the start of the transient. Conservative parameters are selected for the initial coolant conditions as described in Section 8.7.2. Therefore, there is no uncertainty term attributed to the coolant initial conditions.

8.7.3.16 Fuel Rod Internal Pressure

The fuel rod internal pressure relative to the system pressure is used to determine the high temperature failure criteria. This parameter is of primary importance if the less restrictive high temperature criteria are applied.

In the power range, the system pressure exceeds that of the rod internal pressure for low burnup fuel. The approved RODEX-4 is used for determining the rod internal pressure if the less restrictive high temperature criteria is applied. The RODEX-4 is an approved rod mechanical model which predicts the internal rod gas pressurization.

8.7.4 Uncertainty Summary

The uncertainties determined in the prior Sections are combined to provide the enthalpy rise uncertainty. The uncertainties are identified in Table 8.20. [

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Table 8.20 Uncertainty Summary for Enthalpy

[

]

9.0 Evaluation Model Implementation

9.1 *Regulatory Basis*

From SRP 15.0.2 II.1.A, D, and F; the documentation requirements include an overview of the evaluation model, the determination of code uncertainty for a sample plant, and instructions and guidance for the code usage and limitations.

This section presents an example plant analysis of the CRDA with the AURORA-B methodology.

9.2 *Steady State Evaluations Candidate Rod Selection*

The AREVA CRDA methodology is demonstrated on a BWR4 reactor with 764 assemblies, 2 year cycles and EPU.

The sample plant adheres to BPWS for rod pattern control during startup. The A and B sequence groups defined for the sample plant are given in Figure 9.1 and Figure 9.2. For this sample plant, all BPWS allowed pull orders are supported.

9.2.1 Initial Conditions

The determination of the conservative initial conditions is performed for the first application of the methodology for a given plant class, reload size, and fuel product line. The evaluation of initial conditions from Section 8.7.2 is used for this sample application. The initial conditions as determined in Section 8.7.2 are shown in Table 9.2.

9.2.2 Inoperable Rod Positions

A maximum of 8 inoperable rods are allowed for this plant. [

] The inoperable rod locations are
indicated in Figure 9.1 and Figure 9.2.

9.2.3 Group Critical Position

The first step is to determine the end of group or bank position k-effective values to determine where criticality is anticipated to occur for the given rod withdrawal sequence. Figure 9.3, Figure 9.4, and Figure 9.5 show the calculated k-effective values at the end of groups 1 through 4 for an A and B sequence withdrawal sequence.

[

]

9.2.4 Determination of Static Rod Worth

Based on the results of the group worth, static rod worths were then determined. The highest worth rods for each cycle exposure are selected for further evaluation.

For EOFP more rods are selected due to the increased cladding hydrogen content, the increased rod worth, and the top peaked power shape. [

]

Table 9.1 Candidate Rod for Transient Evaluation

[

]

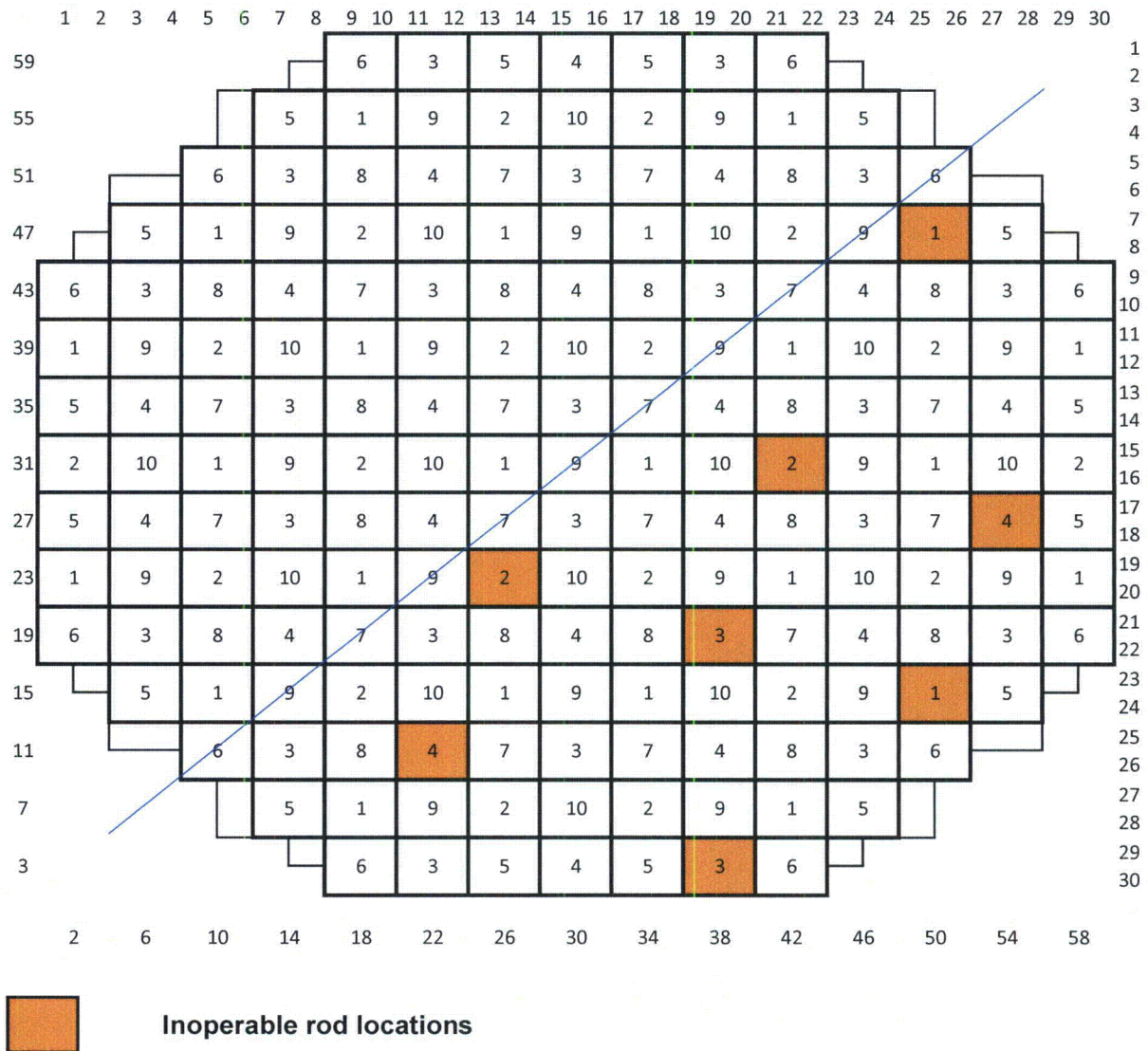
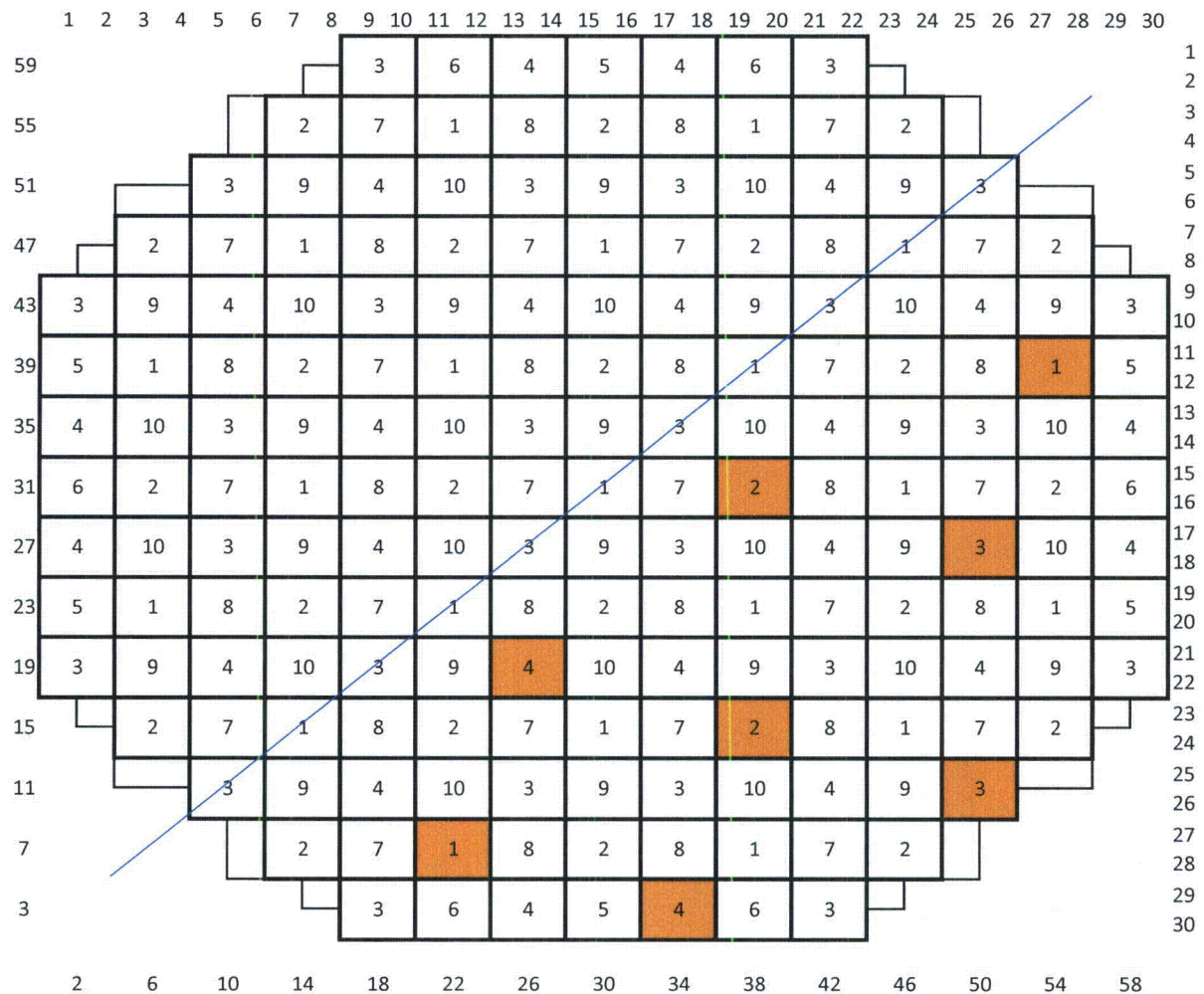


Figure 9.1 Sample Plant A-Sequence Rod Groups



 Inoperable rod locations

Figure 9.2 Sample Plant B-Sequence Rod Groups

[

]

Figure 9.3 BOC K-Effective for A and B Sequences Groups 1 Through 4

[

]

Figure 9.4 PEAK K-Effective for A and B Sequences Groups 1 Through 4

[

Figure 9.5 EOFP K-Effective for A and B Sequences Groups 1 Through 4]

9.3 Transient Evaluation

The evaluation of each rod drop is performed with the AURORA-B system. The initial pre-rod drop state point is established with the MICROBURN-B2 core simulator. The initial conditions used for the transient calculation are identified in Table 9.2.

Table 9.2 Initial Conditions

Parameter	Value
Core Power	0.001 MW
Core Flow	10%
Core Temperature, coolant and fuel	68°F
Core Pressure	15 psia

The channel grouping with a [] is used for this sample plant. Once the channel grouping is defined, the power history information is processed to obtain the fuel rod characteristics for use in the RODEX-4 rod mechanical models.

The rod drops were performed and the maximum prompt and total enthalpy rise determined for the drops are provided in Table 9.3. The reported enthalpy includes the maximum average rod

enthalpy increase (both prompt and total) as well as the peak rod enthalpy increase (both prompt and total). The maximum enthalpy increase with an [

] for this sample evaluation.)

Table 9.3 Maximum Enthalpy Rise for Sample Rod Drops

[

]

Table 9.4 Maximum Enthalpy Rise for Sample Rod Drops with Uncertainty Multiplier

[

]

The first letter of the sequence identifier indicates “A” or “B” sequence and the last four digits indicate the group pull order.

Since the PCMI failure criterion is dependent on the hydrogen content, a detailed evaluation of all nodal enthalpy increases and rod nodal hydrogen content is required. The result of this evaluation is provided in Figure 9.6 and is based only on the peak rod nodal enthalpy increase []. The same data provided is also provided in Figure 9.7 segregated based on reload.

For this sample evaluation there are no rod failures identified. Therefore, no radiological consequence would be required.

[

]

Figure 9.6 Prompt Enthalpy versus Cladding Hydrogen Content

[

]

Figure 9.7 EOFP Prompt Enthalpy Rise by Reload

9.4 ***Evaluation Against Failure Criteria***

9.4.1 Fuel Cladding Failure

9.4.1.1 High Temperature Criteria

The total enthalpy of the peak rod is compared against the high temperature criteria as indicated in Figure 9.8. The high temperature enthalpy criterion is met.

[

]

Figure 9.8 Total Enthalpy versus High Temperature Failure Threshold

9.4.1.2 PCMI Cladding Failure

The evaluation of PCMI cladding failure is evaluated in Section 9.3 and comparison to the failure threshold is provided in Figure 9.6. The prompt enthalpy did not exceed the PCMI failure criteria.

9.4.2 Core Coolability

Since no fuel failures are identified in this sample calculation the four elements of the core coolability criteria are met:

- Peak radial average enthalpy <230cal/g
- Below incipient fuel Melting conditions

- Reactor pressure boundary, reactor internals, and fuel assembly structural integrity is maintained
- No loss of coolable geometry

However, the following discussion is provided.

9.4.2.1 Peak Radial Enthalpy

Evaluation of the high temperature criteria, Figure 9.8, shows that the peak radial enthalpy remains below 230 cal/gram.

9.4.2.2 Peak Fuel Temperature

The fuel temperature was evaluated for the event based on the node with the highest enthalpy. The mesh point temperatures of the peak rod of the node with the highest enthalpy are provided in Figure 9.9, Figure 9.10, and Figure 9.11. The maximum temperature is [

] This is well below incipient fuel melting.

[

]

Figure 9.9 Mesh Point Temperatures Across Peak Rod

[

]

Figure 9.10 Central Mesh Point Temperatures for Peak Rod

[

]

Figure 9.11 Outer Mesh Point Temperatures for Peak Rod

9.4.2.3 Mechanical Energy

The power pulse width for the rod drop was evaluated. [

] that reactor pressure
boundary, reactor internals and fuel assembly structural integrity are maintained.

[

]

Figure 9.12 Rod Drop 118 Power Pulse

9.4.2.4 Cool-able Geometry

The cladding temperature for the peak fuel rod is shown in Figure 9.13 and it remains below 2200°F. [

] coolable geometry is maintained.

[

]

Figure 9.13 Cladding Temperature for Peak Rod

9.5 *Example Radiological Evaluation*

The PCMI failure criterion was reduced for the evaluation provided in Section 9.3 to demonstrate the radiological evaluation when failed rods are identified. The reduced failure threshold for demonstration of the Radiological Evaluation is provided in Figure 9.14. With the reduced failure threshold, two assemblies indicated fuel failures as identified in Table 9.5.

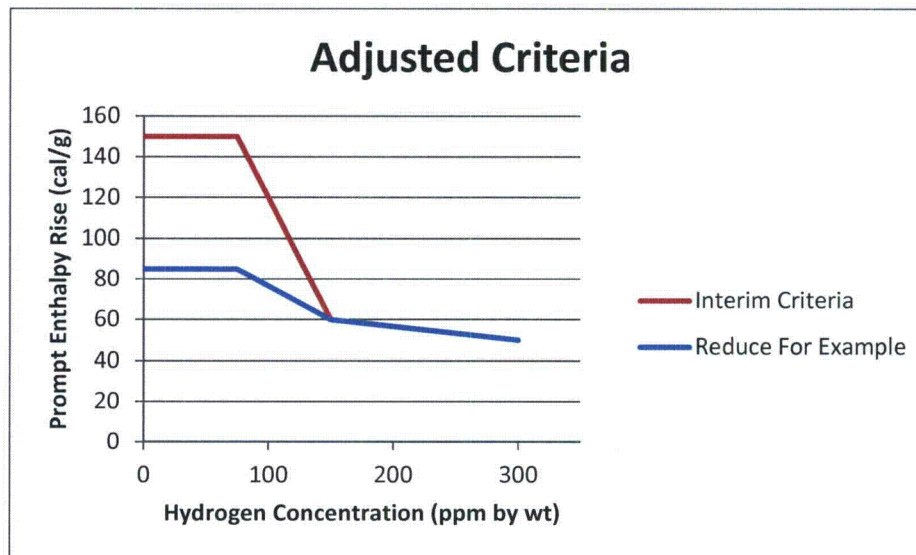


Figure 9.14 Adjusted Failure Criteria for Example Evaluation

Table 9.5 Assemblies with Failed Rods

Exposure	Sequence	Rod	Assemblies with Failures
EOFP	AG1234	118	V17118, V17134

In this example evaluation, a simplified assumption is made that all rods are at the maximum rod nodal hydrogen content rather than utilizing the individual pin hydrogen concentration values.

The number of failed rods were determined for each node by rearranging (Equation 3) from Section 7.6 and substituting the failure threshold Δh_f for Δh_i and n_{fail} for n_{prank} . This allows determination of number of rods that fail.

[

]

Where:

Δh_f	Enthalpy failure threshold
Δh_{max}	Peak rod enthalpy increase
n_{fail}	Number of rods that fail
Δh_{avg}	Average rod enthalpy increase
n_{rods}	The number of rods in the lattice

The equation used to determine the enthalpy for each rod is based on the peaking factor rank with [] conservatism applied to the average enthalpy and assumed that all rods have the maximum hydrogen concentration for the given node for this example.

Table 9.6 Determination of Rod Failures

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* Failed rods are determined based on prompt enthalpy rise (ΔH_p).

The enthalpy dependent release fractions are determined based on the deposited enthalpy as determined from rearranging Equation 5 and determining the average enthalpy increase of the failed rods.

[

](6)

The average pin rank for the failed rods is equal to one half of the total number of rods failed.

The example determination of release fractions is based on both SRP 4.2 and PNNL-18212 Revision 1.

Table 9.7 Fission Gas Release Fraction (SRP 4.2)

[

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Table 9.8 Fission Gas Release Fraction (PNNL-1812 Rev. 1)

[

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For each assembly, the enthalpy dependent release terms are determined along with the ratio of the release fractions to the Reg Guide 1.183 release fractions for the rod drop.

Table 9.9 Fission Gas Release Fraction Ratios

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The number of failed rods are tabulated for those assemblies which have failed rods indicated. This number of failed rods is then multiplied by the ratio of the transient release fraction to the reference release fraction in order to establish an equivalent number of failed rods. The maximum ratio of all groups (I-131, I-132, Kr85, Nobles, Halogens, Alkali Metals) is used to determine the equivalent number of rods.

For assembly V17118, 19 rods are identified as exceeding the reduced failure threshold. Utilizing the ratios of the release fractions, the actual number of failed rods is determined using the two different transient release fractions. The maximum ratio of release fraction to the reference release fraction is [] based on other Halogens using the steady state release fractions of Table 3 of RG1.183 and the transient release term from SRP4.2 Appendix B. The maximum ratio of release fraction to the reference release fraction is [] for Alkali Metals using the transient release term from PNNL-1812 Rev1.

Utilizing the release fraction ratios, an equivalent number of rod failures is determined for the two different transient release terms. The same process is repeated for each of the assemblies for which rod failures are identified for the given rod drop. Table 9.10 contains the total equivalent rod failures for all bundles for the example drop.

The approach of using the maximum ratio implies that release isotope groups are increased by this ratio. No additional radiological evaluation is required past demonstrating that the release term is bounded by that used within the radiological evaluation. For this sample plant 2000 rod failures are utilized in the source term for the CRDA. Therefore, in this sample evaluation, the dose would not exceed the licensing basis.

Table 9.10 Equivalent Rod Failures

[

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10.0 **Quality Assurance Program**

10.1 ***Regulatory Basis***

From SRP 15.0.2 II.1.H, a description of the Quality Assurance program under which the evaluation model was developed and assessed, and the corrective action program that will be used to address error that might be discovered is required.

10.2 ***AREVA QA Program***

The quality assurance plan is not discussed within the topical report. The AREVA quality assurance plan is contained in Reference 64.

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