CENPD-284-NP-RAI
(RPA-89-112-NP-RAI
(RPA-89-053-NP-RAI)
(NRC TAC No. M88025,M75965,M75966)

Control Rod Drop Accident Analysis Methodology for Boiling Water Reactors: Response to Request for Additional Information

October 1994

ABB Combustion Engineering Nuclear Operations

ABB

TABLES OF CONTENTS

This supplemental report contains responses to the NRC Requests for Additional Information regarding References 1, 2, and 4 which were transmitted to ABB by the NRC letters identified in Reference 3.

Reference 1 provided a summary of the ABB methodology for analysis of the Control Rod Drop Accident (CRDA) using RAMONA-3B as well as a sample analysis for a 532-assembly BWR. Reference 2 provided an additional sample CRDA analysis illustrating the impact of a postulated CRDA in a plant equipped with high worth control rods. Reference 4 was submitted in 1993 to clarify and summarize the ABB CRDA methodology as well as to provide further information supporting the qualification of the ABB CRDA analysis methodology using RAMONA-3B.

A. ADDITIONAL INFORMATION REQUIRED FOR THE REVIEW OF RPA-89-112

<u>NRC Question A1</u>

Describe any significant differences between the typical BWR/4-6 fuel and control rod designs and the designs to which this methodology will be applied.

<u>ABB Response to Question A1</u>

The ABB methodology described in Reference 4 will be applied to BWR/2 through BWR/6 plants loaded with commercially available reload fuel. As discussed in Reference 4, nuclear data for RAMONA-3B will be calculated with a lattice code and three-dimensional core simulator accepted by the NRC for licensing applications. Specifically, the three dimensional static core simulator POLCA, in conjunction with the cross section generator code PHOENIX, are utilized for this evaluation and are documented in Reference 5. These are the same codes used for reload design purposes, and benchmark calculations are performed relative to available plant data to confirm that predictions of core reactivity and power shape are adequate. Section 5.2 of Reference 4 provides a summary of the benchmark information in Reference 5.

The methodology will be applied to BWR/2 through BWR/6 plants equipped with commercially available control rods. The only significant sensitivity to control rod design expected for the CRDA might be the reactivity worth of the control rods installed in the reactor. For example, the "high worth" control rods referred to in Reference 2 might be installed rather than the "standard" control rods referred to in Reference 1. As discussed in Reference 4, the ABB static methods as well as the RAMONA-3B code are used to predict reactivity worths which are sufficiently accurate for evaluation of the CRDA. Reference 4 describes in detail the ABB methodology for establishing limiting dropped rod candidates and evaluating the impact of a postulated CRDA for those candidates.

Please also see the response to Question C10 for a discussion of differences between ABB and GE control rod designs.

<u>NRC Question A2</u>

What are the differences between the GE methods of References 1-4 and the methods of RPA-89-112?

ABB Response to Question A2

The ABB methodology is described in detail in Reference 4. This methodology utilizes state-of-the-art methods based on the RAMONA-3B code and a systematic approach based on well-established sensitivities and application-specific calculations to identify and evaluate the consequences of a worst-case postulated CRDA. To the extent that the results in References 1, 2, and 4 can be compared with those in References 1 through 4 in Reference 1, the results of the two methodologies are considered to be consistent.

NRC Question A3

Does PHOENIX use a pre-ENDF/B-V value for β and, if not, justify the value used?

ABB Response to Question A3

The delayed neutron fractions in the cross section library used for the calculations in References 1, 2, and 4 were taken from [Proprietary Information Deleted.]

The sensitivity of the peak fuel enthalpy following a postulated CRDA to delayed neutron fraction is discussed in Section 4.5 of Reference 4. This discussion indicates that the severity of the CRDA is [Proprietary Information Deleted.]

NRC Question A4

What are the differences between RAMONA-3B and RAMONA-3B-SCP2, and what is their effect on the modeling, benchmarking and analysis of the rod drop accident (RDA)?

ABB Response to Question A4

ABB utilizes the Scandpower version of RAMONA-3B as explained in Section 4.3.1 of Reference 4. This version of RAMONA-3B, referred to as the "Scandpower version of RAMONA-3B" in Reference 4 and RAMONA-3B-SCP2 in References 1 and 2, includes the features in BNL Level 10 and can be considered to be upgraded to BNL version "Level 10". The most important extensions relative to the version described in Reference 6 can be summarized as follows:

(1) The nuclear cross-section data representation is made compatible with Scandpower's static 3-D core analysis methods (FMC) and the ABB CORE MASTER system which includes POLCA.

(2) The option to input a non-equilibrium xenon distribution which provides the capability to initiate the transient from a non-equilibrium xenon state.

(3) The option to treat effective delayed neutron fractions as nodal variables as a function of fuel design and burnup has been installed.

(4) The nodal coupling method dealing with the thermal flux diffusion has been upgraded to that of the static three-dimensional nodal simulator, PRESTO.

(5) The modeling of reverse flow conditions has been improved.

(6) The time integration of the hydraulics has been improved to optionally allow for higher order explicit methods as well as implicit integration of some of the equations.

A major effect on modeling of these improvements is [Proprietary Information Deleted.] The Scandpower version of RAMONA-3B used by ABB is considered to represent a substantial improvement relative to the code version discussed in Reference 6. Therefore, while ABB has not performed benchmark calculations for the code version described in Reference 6, it is expected that such a benchmark [Proprietary Information Deleted.]

NRC Question A5

How is the time dependence of the local rod-to-bundle power peaking factor accounted for?

ABB Response to Question A5

As discussed in Section 4.3 of Reference 4, at the state point for which the CRDA is to be simulated, [Proprietary Information Deleted.]

NRC Question A6

Provide References 10, 19, 30, 31, 35, 36, and 42.

ABB Response to Question A6

Copies of References 19, 30, 31, and 35 are included with this transmittal.

Reference 10 is RPA-89-112 is RPA-89-053 for which responses to NRC questions are provided in Section B.

Reference 36 is an Institute of Energy Technology Report documenting an early application of RAMONA-3 to control rod insertion. The report conclusions are that uncertainties in the fuel burnup for the test conditions made the comparison with data difficult, however reasonable agreement was obtained for the full scram test. For the single rod scrams, the simulations showed best agreement with experimental data obtained in positions close to the scrammed channels. The work and report are property of the Institute of Energy Technology and the document describes early benchmark work with RAMONA-3. In light of the information provided in Reference 4 and this document, the contents of Reference 36 does not contain any more relevant information. The response to Question A9 explains the RAMONA versions used in the Reference 36 and other benchmarks efforts.

Reference 42 was submitted to the NRC in August of 1987, and a revised version is scheduled for submittal by ABB in November of 1994. CENPD-284-P, Reference 4, has been provided to the NRC to give much more detailed and updated description of the CRDA application methodology than that provided in Reference 42 of RPA-89-112 (Reference 1). Therefore, Reference 4 should be utilized for an explanation of the application methodology.

<u>NRC Question A7</u>

Describe the qualification of the Version-SCP2 thermal diffusion option that has been performed for transients like the RDA in which strong spatial peaking occurs.

<u>ABB Response to Question A7</u>

As noted in the response to Question A4, the nodal coupling method dealing with the thermal flux diffusion has been upgraded to that of the static three-dimensional nodal simulator, PRESTO. Therefore, as discussed in Section 5.3.1 of Reference 4, the time-dependent analytical models for performing the coupled neutron flux-coolant void calculations in the version of RAMONA-3B utilized by ABB, the Scandpower version of RAMONA-3B, are equivalent to those in the PRESTO three dimensional core simulator (References 7 and 8) in the steady-state. Consequently, the capability of PRESTO to predict steady-state power distributions provides a good indication of the capability of the Scandpower version of RAMONA-3B to predict power distributions. The PRESTO code has been reviewed and accepted for steady-state neutronics applications by the NRC. As shown in Section 5.3.1 of Reference 4 and References 7 and 8, PRESTO, and therefore, RAMONA-3B, provide a state-of-the-art capability to provide reliable power shapes under hot and cold conditions.

NRC Question A8

Describe the results of the Muhleberg and Brunswick-1 RAMONA-3B turbine trip test comparisons. How do the methods and models used in these calculations compare to the ABB-Atom RDA licensing analyses?

ABB Response to Question A8

The Muhleberg and Brunswick-1 turbine trip test comparisons were performed to confirm the capability of RAMONA-3B to predict power excursions caused by a core void collapse initiated by a turbine trip. Unfortunately, formal documentation of these comparisons is not readily available. Reference to applications of RAMONA-3B to occurrences other than the CRDA in References 1 and 2 were intended to indicate the general reliability of the code to predict the response to a broad range of different occurrences and the broad experience of ABB and Scandpower in applying the code.

NRC Question A9

Were the Scandpower Peach Bottom-2, Muhleberg, Brunswick-1 and Gundremmingen A (KRB) comparisons made with Version-SCP2 and, if not, discuss the applicability of these models/comparisons as qualification for the RPA-89-112 RDA methodology?

ABB Response to Question A9

The Scandpower version of RAMONA-3B used for the calculations in References 1, 2, and 4 is the same as that used for the Peach Bottom-2, and Brunswick-1 comparisons. The Gundremmingen A (KRB) comparison was performed with a previous code version. Reference to these applications of RAMONA-3B were intended to indicate the general reliability of the code to predict the response to a broad range of different occurrences and the broad experience in applying the code. The simulations of the SPERT-IIIE power excursion tests in Reference 4 is more applicable to the CRDA methodology.

NRC Question A10

Discuss the results and applicability of the ABB-Atom and Scandpower RAMONA-3B RDA "actual plant applications" (referenced on p. 27) as qualification for the RDA licensing analyses.

ABB Response to Question A10

Reference to applications of RAMONA-3B to occurrences other than the CRDA in References 1 and 2 were intended to indicate the general reliability of the code to predict the response to a broad range of different occurrences and the broad experience of ABB and Scandpower in applying the code.

Specific qualification of the ABB CRDA methodology is provided in Reference 4. The application to CRDAs referred to in References 1 and 2 generally involved analyses to support plant operation rather than benchmark calculations. Reference was made to these applications to indicate the general usefulness and reliability of the Scandpower version of RAMONA-3B for this accident. Qualification of the ABB CRDA methodology RAMONA-3B is addressed in detail in Reference 4.

NRC Question A11

While applications of BNL and Scandpower are discussed in RPA-89-112, what qualification comparisons have been performed by ABB-Atom with the licensing Version-SCP2 of RAMONA-3B using the PHOENIX/POLCA/POLGEN cross section calculation.

ABB Response to Question A11

Reference 4 contains a detailed discussion of the qualification basis for the ABB CRDA methodology using the Scandpower version of RAMONA-3B in conjunction with the PHOENIX/POLCA/POLGEN cross section calculation. Specifically, Reference 4 contains information to verify that the ABB CRDA methodology is sufficiently accurate and conservative for licensing applications. The verification is provided by [Proprietary Information Deleted]

[Proprietary Information Deleted]

NRC Question A12

Discuss the quality assurance program and application under which RAMONA-3B-SCP2 was developed and qualified.

ABB Response to Question A12

The Scandpower version of RAMONA-3B was developed by Scandpower. ABB and Scandpower have jointly qualified the code as discussed in Reference 4. [Proprietary Information Deleted]

[Proprietary Information Deleted]

NRC Question A13

Provide the details of the calculations and results of the RAMONA-3B-SCP2 comparisons to the SPERT-III E-Core transient measurements.

ABB Response to Question A13

Simulations of six of the SPERT-IIIE power excursion tests using the ABB CRDA methodology are provided in Section 5.3.2 of Reference 4.

NRC Question A14

In representing an off-center control rod with a central rod, how are the local peaking and feedback at the off-center location preserved in the center rod drop calculation?

ABB Response to Question A14

As discussed in Reference 4, the dropped control rod is [Proprietary Information Deleted.]

[Proprietary Information Deleted]

NRC Question A15

Are the same procedures used to model the core loading (number of fuel types, axial fuel zones, etc.) in RAMONA-3B-SCP2 as are used in the PHOENIX/POLCA/POLGEN modeling? If not, discuss the effect these differences will have on the RDA modeling, analysis and benchmarking.

ABB Response to Question A15

[Proprietary Information Deleted]

NRC Question A16

How is the coupling between the void and fuel temperature dependence accounted for in the RAMONA-3B-SCP2 cross section representation?

ABB Response to Question A16

As discussed in Section 4 of Reference 4, at the core conditions for which the CRDA is to be simulated [Proprietary Information Deleted.]

In addition, fast absorption, removal, and fission cross sections are assumed to vary as the square root of the fuel temperature. Specifically, the dependence on coolant density and the coupling between the coolant density and fuel temperature in the Scandpower formulation of these cross sections is expressed in the form:

$a + b\rho + c\rho^2 + (d + e\rho + f\rho^2)(\sqrt{T_f} - \sqrt{T_{fo}})$, where

ρ = coolant density

T_f = fuel temperature

T_{fo} = reference fuel temperature, and

a, b, c, d, e, and f are constants at a given burnup and void history.

The moderator density is updated for the power calculation at each time step, thereby accounting for moderator temperature feedback. The fast group cross sections are updated by the current fuel temperature at each time step to account for Doppler feedback.

NRC Question A17

As validation of the POLGEN/RAMONA-3B-SCP2 cross section representation, provide comparisons of RAMONA-3B-SCP2 and POLCA calculated power distributions and feedback reactivities for typical RDA statepoints.

ABB Response to Question A17

As discussed in Section 4.5 of Reference 4, this type of validation is part of the ABB methodology. [Proprietary Information Deleted]

The type of global check suggested by the question is also performed. [Proprietary Information Deleted]

[Proprietary Information Deleted]

As indicated in the sensitivity studies in Section 4 of Reference 4 and the response to Question C17, [Proprietary Information Deleted.]

NRC Question A18

Provide an estimate of the uncertainty introduced into the feedback coefficient by combining the moderator temperature and moderator void dependence, and the impact on the RDA.

ABB Response to Question A18

The moderator void and temperature affects the impact of the dropped control rod through the feedback which these coolant properties have on the core power and power distribution as well as the thermal conductivity from the fuel rod to the coolant. It is our judgment that combining the moderator temperature and moderator void dependence in the cross section description does not introduce any significant uncertainties into the predicted feedback from these coolant properties.

[Proprietary Information Deleted]

The thermal-hydraulic model in RAMONA-3B treats both the moderator void and moderator temperature explicitly. The impact which the moderator void and temperature have on the heat conductance from the fuel rod to the coolant is treated in RAMONA-3B by utilizing convective heat transfer coefficients in the solution of the coupled heat conduction differential equations which depend on the fluid properties in the coolant.

NRC Question A19

Recognizing that a larger transient increase in fuel temperature results in an increased Doppler feedback, how is a conservative gap conductance determined for the RDA? How is the fuel design dependence and fuel burnup dependent gap closure and fission gas release accounted for in the determination of the gap conductance?

ABB Response to Question A19

Gap conductance is modeled as a quadratic function of average fuel temperature in the Scandpower version of RAMONA-3B used by ABB. [Proprietary Information Deleted]

[Proprietary Information Deleted]

NRC Question A20

If moderator feedback is to be included in licensing analyses of the RDA, recognizing the substantial degree of uncertainty in the magnitude and timing of the moderator voiding under the highly transient conditions of the RDA, provide detailed justification and model qualification for the relaxation of this conservatism.

ABB Response to Question A20

[Proprietary Information Deleted]

As discussed in Section 5.3.1 of Reference 4, the time-dependent hydraulic models in the Scandpower version of RAMONA-3B are equivalent to those in the PRESTO three dimensional core simulator under steady-state conditions. Therefore, the capability of PRESTO to predict steady-state hydraulic conditions provides an indication of the capability of the Scandpower version of RAMONA-3B to reliably predict hydraulic conditions during a transient. Detailed descriptions and qualification of the PRESTO hydraulic models were submitted to the NRC in References 11 and 12. It is demonstrated in these documents that the hydraulic modeling of the BWR two-phase system under steady-state conditions in PRESTO is a state-of-the-art representation. [Proprietary Information Deleted.]

[Proprietary Information Deleted.] As discussed in Section 3.4 of Reference 6, RAMONA-3B contains state-of-the-art the heat conduction models. The thermal energy distribution and heat conduction from the pellet to the coolant is performed in RAMONA-3B by solving standard coupled, time-dependent radial heat conduction differential equations in the pellet and clad. The pellet and clad are nodalized into concentric rings. [Proprietary Information Deleted.]

[Proprietary Information Deleted]

NRC Question A21

How are conservative initial conditions (cycle burnup, power level, inlet subcooling, etc.) and modeling parameters (Doppler coefficient, delayed neutron fraction, scram worth, etc.) selected for RDA licensing analyses?

ABB Response to Question A21

The selection of initial conditions is addressed in some detail in Reference 4. [Proprietary Information Deleted]

[Proprietary Information Deleted]

NRC Question A22

Do any of the sensitivities provided in Section-6 change significantly for Version-SCP2?

<u>ABB Response to Question A22</u>

The designation "RAMONA-3B SCP2" in Reference 1 refers to the Scandpower version of RAMONA-3B which has been upgraded relative to the code version described in Reference 6. Section 4.3.1 of Reference 1 lists some of the more important upgrade features. All of the ABB calculations reported in Reference 1, specifically the results in Sections 5 and 6, were performed with this upgraded Scandpower version of the RAMONA-3B. Therefore, the sensitivity calculations in Section 6 as well as the base case calculations in Section 5 were performed with "RAMONA-3B SCP2".

B. ADDITIONAL INFORMATION REQUIRED FOR THE REVIEW OF RPA-89-053

NRC Question B1

What is the basis for assuming the worth of the HWCR is 15% greater than a standard control rod? Does this 15% difference provide a bound for all variations in core conditions (rod insertion, moderator density, etc.)?

ABB Response to Question B1

The 15% difference in total reactivity worth between the high worth control rod and the standard rod was selected as typical. The purpose of the analysis in Reference 2 was to provide an indication of the impact on a CRDA for a U.S. reactor containing high worth control rods relative to one containing standard control rods. [Proprietary Information Deleted]

Performance of licensing basis calculations is more clearly described in Reference 4 and clarified in this document.

NRC Question B2

Provide quantitative justification for the spatial nodalization of the fuel rod heat transfer equations. How is the pellet/clad gap described?

ABB Response to Question B2

As discussed in Reference 6, the thermal energy distribution and heat conduction from the pellet to the coolant is performed by solving standard coupled, time-dependent radial heat conduction differential equations for the pellet, gap and cladding. The pellet and clad are nodalized into concentric rings. [Proprietary Information Deleted]

[Proprietary Information Deleted]

NRC Question B3

Provide a quantitative estimate of the uncertainty introduced by using the fuel Type-2 fuel temperature, moderator density and control rod insertion dependence for all fuel types. Does fuel Type-2 have the most conservative feedbacks and control dependence?

ABB Response to Question B3

Depletion calculations providing nuclear data (e.g. cross sections and local peaking factors) for the same void and void history were performed for each fuel type. The differentials in the nuclear data caused by changing the coolant void, fuel temperature, and control state calculated for Fuel Type 2 were applied to the appropriate mainline depletion results for each of the other fuel types. The U-235 enrichments for Fuel Types 1 and 3 are the same as for Fuel Type 2. The only difference in the fuel types was in the Gd_2O_3 design, and the calculations were performed sufficiently late in the cycle that this difference is considered to be minor. Therefore, it is judged that conclusions regarding the relative impact of a core containing high worth control rods relative to one containing standard control rods would not be altered by this approximation. Fuel Type 4 is sufficiently unimportant to the determination of the peak fuel enthalpy that this approximation will not significantly impact the calculated peak fuel enthalpy.

It should be noted that the analyses in Reference 2 were performed as a sensitivity to evaluate the impact of installing high worth control rods. The ABB methodology for CRDA licensing applications is summarized in Reference 4. [Proprietary Information Deleted]

<u>NRC Question B4</u>

What flux shape is used to determine the importance-weighted delayed neutron fraction? Is this conservative for the RDA?

<u>ABB Response to Question B4</u>

As noted in Reference 4, beta-effective is provided as a function of burnup for each fuel type on a nodal basis in the Scandpower version of RAMONA-3B used by ABB. Optionally, it can be provided on a core-average basis. [Proprietary Information Deleted] The average importance weighted core value of beta referred to on page 13 of Reference 2 is a core average value edited by the code for information and is not used in the calculations. The value quoted corresponds to the core power distribution reflected by the average axial power shape shown in Figure 6.3 of Reference 2.

[Proprietary Information Deleted] The current delayed neutron fractions used by ABB are also discussed in the response to Question A3.

<u>NRC Question B5</u>

Recognizing that the Doppler coefficient decreases at higher temperatures and the RDA is sensitive to the initial power level, how will ABB-Atom insure that the hot-zero-power and low-power events are bounded by the cold-zero-power RDA?

ABB Response to Question B5

[Proprietary Information Deleted]

Please also see the response to Question C17.

NRC Question B6

How are inoperable rods accounted for in the selection of the maximum worth rod? How will the limiting control rod pattern be selected in the RDA licensing analysis?

ABB Response to Question B6

[Proprietary Information Deleted]

Please also see the response to Question C2.

NRC Question B7

What is the effect of assuming a "linear" rod insertion on the peak fuel enthalpy?

ABB Response to Question B7

[Proprietary Information Deleted]

NRC Question B8

Are the conditions of Section-6, at which the rod worth is calculated to be 0.01174, the same as the conditions that were used to calculate the 0.012 rod worth of Reference-5? If not, how do these rod worths compare at identical conditions?

ABB Response to Question B8

The conditions in Section 6 of Reference 2 are not the same as those in Reference 5 of Reference 2. For example, the analyses were performed for different cores. Reference 5 of Reference 2 identified this reactivity worth as the maximum incremental dropped rod worth when the maximum number of rods are bypassed in the particular 748-assembly core for which the analysis was performed. The reference to Reference 5 of Reference 2 was intended only to put in perspective the value of 0.012 as a relatively high reactivity worth which would not be expected to be encountered under normal conditions.

The specific core evaluated for the CRDA in Reference 5 of Reference 2 has not been evaluated by ABB. [Proprietary Information Deleted]

Reference 4 provides further discussion of conclusions based on the ABB methodology relative to those of previous work.

NRC Question B9

The extrapolation of the fuel enthalpy in the RDA with feedback, from 2 seconds to the time at which the peak occurs (typically 5 seconds), based on the time dependence of the RDA without feedback is highly uncertain. Therefore, if the licensing calculations will be performed with feedback, provide the RAMONA-3B-SCP2 calculation beyond the time of the peak fuel enthalpy.

ABB Response to Question B9

[Proprietary Information Deleted]

NRC Question B10

Provide a quantitative estimate of the uncertainty introduced by the limited number of thermal-hydraulic channels used to represent the core thermal-hydraulics (e.g., in Figure 5.3).

ABB Response to Question B10

[Proprietary Information Deleted]

NRC Question B11

Describe the benchmarking and testing performed to validate the PHOENIX/POLCA and RAMONA-3B-SCP2 neutronics schemes for application to HWCRs.

ABB Response to Question B11

As discussed in Section 5.3.1 of Reference 4, the time-dependent analytical models for performing the coupled neutron flux-coolant void calculations in the version of RAMONA-3B utilized by ABB, the Scandpower version of RAMONA-3B, are equivalent to those in the PRESTO three dimensional core simulator (Reference 7 and 8) in the steady-state. Therefore, the capability of PRESTO to predict control rod worths and power distributions provides a good indication of the capability of the Scandpower version of RAMONA-3B to predict control rod worths and power distributions. The PRESTO code has been reviewed and accepted for steady-state neutronics applications by the NRC. Qualification of PRESTO is addressed in depth in References 7 and 8. [Proprietary Information Deleted]

[Proprietary Information Deleted]

<u>NRC Question B12</u>

Has the axial expansion of the fuel pellet been accounted for in the determination of the Doppler coefficient?

<u>ABB Response to Question B12</u>

[Proprietary Information Deleted]

C. ADDITIONAL INFORMATION CONCERNING REPORT CENPD-284-P

NRC Question C1

Recognizing the substantial degree of uncertainty in the magnitude and timing of the moderator voiding under the highly transient conditions of the RDA, provide detailed justification and model qualification for the use of this previously unapproved additional transient feedback.

ABB Response to Question C1

Please see the response to Question A20.

NRC Question C2

Discuss how the single equipment malfunction and operator error are included in the determination of the maximum worth rod. For plants using the rod worth minimizer, the rod sequence control system or the rod pattern control system, how are bypassed rods selected and accommodated?

ABB Response to Question C2

The existing plant-specific worst-case credible single equipment malfunction and operator error allowed by the design and administrative procedures are utilized by ABB in the CRDA evaluation. The existing worst-case situation is defined by the plant reactor control system, the plant technical specifications, and current licensing basis CRDA evaluation. Since the limiting assumptions for equipment malfunction and operator error are not fuel-type specific, substantial revisions to existing assumptions regarding single equipment malfunction and operator error are not anticipated to be required for most applications when ABB reload fuel is installed in a particular plant. [Proprietary Information Deleted]

As noted in the question, the assumed worst-case single equipment malfunction and operator error will depend on the control rod withdrawal system utilized for a given plant. The example provided in Section 4 of Reference 4 is for a Banked Position Withdrawal Sequence (BPWS) plant. Technical Specifications for BPWS plants typically require that no more than 8 rods be inoperable and that each inoperable control rod be separated from all other inoperable rods by at least two control cells in all directions. [Proprietary Information Deleted]

NRC Question C3

How is it assured that CRDAs like the case in Figure 4.4.3, with a ~990 pcm rod worth and a nodal peaking of ~64, are not limiting?

ABB Response to Question C3

[Proprietary Information Deleted]

NRC Question C4

In Step-6 of the cycle-specific evaluation, how will conservative values for the scram worth, velocity and delay be determined?

ABB Response to Question C4

[Proprietary Information Deleted]

NRC Question C5

Describe and justify the important RAMONA-3B core and systems modeling assumptions made in the CRDA licensing analyses (neutronic and thermal-hydraulic channels, fuel rod nodalization, etc.)

ABB Response to Question C5

[Proprietary Information Deleted]

The fuel rod nodalization in the RAMONA-3B calculations is discussed in the response to Question B2.

NRC Question C6

Describe how the fuel burnup and void history dependence is included in the nodal neutronics data.

ABB Response to Question C6

[Proprietary Information Deleted]

NRC Question C7

Does PHOENIX use a pre-ENDF/B-V value for β and, if so, justify the value used?

ABB Response to Question C7

Please see the Response to Question A3.

NRC Question C8

Discuss the modeling of reverse flow and the effect on the CRDA licensing analyses.

ABB Response to Question C8

[Proprietary Information Deleted]

NRC Question C9

For what specific fuel/core designs is the CRDA methodology intended? Has the PHOENIX/FOBUS/PHOEBE/PHIPO/POLCA code system been approved for these applications? Has the STAV fuel performance code system been approved for these applications?

ABB Response to Question C9

[Proprietary Information Deleted]

NRC Question C10

What are the significant differences between the ABB-CE and GE fuel and control rod designs, and what is their impact on the CRDA?

ABB Response to Question C10

A detailed description of the ABB SVEA-96 assembly being utilized in the U.S. can be found in Reference 11. As discussed in Reference 4, nuclear data for RAMONA-3B will be calculated with a lattice code and three-dimensional core simulator accepted by the NRC for licensing applications. [Proprietary Information Deleted]

A description of the ABB control rod and a comparison with GE blades are provided in Attachment C10-1. [Proprietary Information Deleted]

NRC Question C11

The reactivity insertion rate and resulting transient peak fuel enthalpy are increased if the reactivity is inserted over a reduced axial span (assuming a constant rod drop speed). In the case that the control rod drops to a rod drive located above the bottom of the *core, is the total reactivity in the RAMONA-3B calculation (which may have been precalculated) inserted over this reduced axial span? If not, justify the method used.*

ABB Response to Question C11

[Proprietary Information Deleted]

NRC Question C12

Discuss how a maximum rod drop speed less than 3.1 ft/sec is justified.

ABB Response to Question C12

[Proprietary Information Deleted]

NRC Question C13

Describe the difference between a power scram and a flux scram, and provide an estimate of the conservatism included by using the power scram rather than a flux scram.

ABB Response to Question C13

The "power scram" assumes that the process to initiate a scram is started when the core power reaches a certain level. The "flux scram" assumes that the process to initiate a scram is started when the core flux reaches a certain level. [Proprietary Information Deleted]

NRC Question C14

When the RAMONA-3B model is expanded to full core geometry, are reductions made in the number of neutronic and thermal-hydraulic channels (per octant)? If so, how have these approximations been validated?

ABB Response to Question C14

[Proprietary Information Deleted]

NRC Question C15

Can the control rod insert additional reactivity by dropping past the axial location of the drive mechanism?

ABB Response to Question C15

[Proprietary Information Deleted]

NRC Question C16

How will conservative values be determined for the Doppler coefficient, gap conductance and thermal conductivity for a given cycle statepoint in the licensing analyses?

ABB Response to Question C16

Please see Attachment A19-1.

NRC Question C17

The 5% and 10% power cases of Figure 4.5.19 indicate a substantial increase in the CRDA peak fuel enthalpy with increasing power. In order to justify the licensing analysis at low power, provide an evaluation of the effect of the reduction in Doppler feedback reactivity and the reduction in CRDA rod worth that occurs at higher powers.

ABB Response to Question C17

[Proprietary Information Deleted]

NRC Question C18

While it is recognized that the minimum service limit pressure is at ~20°C, can a CRDA initiated from low subcooling conditions result in a closer approach to the service limit than the assumed CRDA from 20°C?

ABB Response to Question C18

[Proprietary Information Deleted]

NRC Question C19

In Step-1 of the cycle-specific evaluation, what specific criteria are used to conclude from existing CRDA analyses that a dynamic analysis is not necessary?

ABB Response to Question C19

[Proprietary Information Deleted]

NRC Question C20

In Step-4 of the cycle-specific evaluation, in cases where all parameters of a previous dynamic analyses do not bound the cycle-specific CRDA, will a cycle-specific CRDA be performed? If not, describe how a bounding CRDA will be determined.

ABB Response to Question C20

[Proprietary Information Deleted]

NRC Question C21

Does moderator voiding have a significant effect on the CRDA calculated for the "analysis condition" statepoint of Figure 4.7.1?

ABB Response to Question C21

[Proprietary Information Deleted]

NRC Question C22

Provide justification for the conservatisms that are relaxed in the "more realistic" calculations that will be performed when the design criteria are not satisfied by the bounding CRDA.

ABB Response to Question C22

[Proprietary Information Deleted]

NRC Question C23

As validation for the RAMONA-3B core analysis capability, provide comparisons of the RAMONA-3B and POLCA prediction of power distribution, bank and rod worths, and Doppler defect.

ABB Response to Question C23

Please see the response to Question A17.

NRC Question C24

What specific version of the RAMONA-3B code will be used in the CRDA licensing analyses?

ABB Response to Question C24

ABB utilizes the Scandpower version of RAMONA-3B. Significant differences relative to the version of RAMONA-3B described in Reference 6 are described in the response to Question A4.

[Proprietary Information Deleted]

D. REFERENCES

1. ABB Report RPA-89-112, "ABB Atom Control Rod Drop Accident Analysis Methodology for Boiling Water Reactors, The RAMONA-3B Computer Code," November, 1989.

2. ABB Report RPA-89-053, "ABB Atom High Worth Control Rods for US BWRs, Rod Drop Accident Analysis," August, 1989.

3. NRC letter from Timothy E. Collins (NRC) to Derek Ebeling-Koning (ABB), "Request for Additional Information for Topical Reports RPA-89-112 and RPA-89-053", May 19, 1994.

   NRC letter from Timothy E. Collins (NRC) to Derek Ebeling-Koning (ABB), "Request for Additional Information for ABB-CE Topical Report CENPD-284-P (TAC. NO. M88025)," August 5, 1994.

4. ABB Report CENPD-284-P, "Control Rod Drop Accident Analysis Methodology for Boiling Water Reactors: Summary and Qualification," October 1993.

5. "ABB Atom Nuclear Design and Analysis Programs for Boiling Water Reactors Programs Description and Qualification," BR 91-402-P-A (proprietary), BR 91-403-NP-A (non proprietary), May 1991.

6. W. Wulff et. al., "A Description and Assessment of RAMONA-3B Mod. 0 Cycle 4: A Computer Code with Three Dimensional Neutron Kinetics for BWR System Transients," NUREG/CR-3664, 1984.

7. S. Borresen, L. Moberg, J. Rasmussen, "Methods PRESTO-B-A Three Dimensional BWR Core Simulation Code," NF-1583.03, US-NRC Topical Report Submitted by Carolina Power & Light Company, 1983.

8. K. E. Karcher, W. K. Cantrell, and D. W. Schroeder, "A Description and Validation of Steady-State Analysis Methods For Boiling Water Reactors," NF-1583.01, US-NRC Topical Report Submitted by Carolina Power & Light Company, 1983.

9. "Fuel Rod Design Methods for Boiling Water Reactors," ABB Report CENPD-285-P (proprietary), CENPD-285-NP (nonproprietary), May, 1994.

10. C. J. Paone, R. C. Stirn, J. A. Woolley, "Rod Drop Accident Analysis for Large Boiling Water Reactors," GE Report NEDO-10527, March 1972.

11. "Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors," ABB Report CENPD-287-P (proprietary), CENPD-287-NP (nonproprietary), June, 1994.

12. "Quality Assurance Program, A Description of the CENO Nuclear Quality Assurance Program" ABB Report CENPD-210A, latest revision.

13. ANL-5800, Reactor Physics Constants, July, 1963.

14. "ABB BWR Generic Control Rod Design Methodology", CENPD-290-P (proprietary), CENPD-290-NP (nonproprietary), February, 1994.

TABLE A17-1

PROPRIETARY INFORMATION DELETED

TABLE C17-1

PROPRIETARY INFORMATION DELETED

TABLE C18-1

PROPRIETARY INFORMATION DELETED

[Proprietary Information Deleted]

Figure A3-1 Peak Fuel Enthalpy Sensitivity to Effective Delayed Neutron Fraction

[Proprietary Information Deleted]

Figure A17-1 Axial Power Distributions Predicted by POLCA and RAMONA-3B For Initial Conditions in Base Case in Reference 4

[Proprietary Information Deleted]

Figure A17-2 [Proprietary Information Deleted]

[Proprietary Information Deleted]

Figure A17-3 [Proprietary Information Deleted]

[Proprietary Information Deleted]

Figure A17-4 Scram Worth for Base Case in Reference 4

[Proprietary Information Deleted]

Figure A17-5 Initial Control Rod Configuration for Base Case in Reference 4

[Proprietary Information Deleted]

Figure A17-6 [Proprietary Information Deleted]

[Proprietary Information Deleted]

Figure A19-1 [Proprietary Information Deleted]

[Proprietary Information Deleted]

Figure C17-1 [Proprietary Information Deleted]

[Proprietary Information Deleted]

Figure C17-2 [Proprietary Information Deleted]

ATTACHMENT A19-1 TREATMENT OF BOUNDING VALUES AND UNCERTAINTIES

This attachment provides a clarification and expansion of the treatment of bounding values and uncertainties relative to the discussion in Reference 4.

[Proprietary Information Deleted]

TABLE A19-1.1

PROPRIETARY INFORMATION DELETED

[Proprietary Information Deleted]

Figure A19-1.1 [Proprietary Information Deleted]

ATTACHMENT C10-1 ABB CONTROL RODS

ABB Control Rod Description

The ABB control rod design consists of neutron-absorbing materials, B_4C and Hafnium, contained in horizontally drilled holes in solid sheets of high grade 316L stainless steel. This results in a design where the blades act both as structural and containment elements. Thus, the design is completely free from crevices and other cavities, which is optimum from the corrosion viewpoint. In addition, this design has one-third the surface area of GE original equipment control rods, which yields a benefit in terms of cobalt activation.

Spacing within the gap between fuel assemblies is maintained by the use of Inconel X-750 buttons. These wear resistant buttons serve as the contact points between the control rod and adjacent fuel channels during operation. When used in a GE designed BWR, the ABB control also has a velocity limiter identical to that used on the GE original equipment control rods.

[Proprietary Information Deleted]

A very detailed description of the ABB control rod design, adopted to the various GE BWR lattice types, is contained in Reference 14.

[Proprietary Information Deleted]

ABB Combustion Engineering Nuclear Operations
Combustion Engineering, Inc.
1000 Prospect Hill Road
Post Office Box 500
Windsor, Connecticut 06095-0500
Telephone: (203) 285-9678
Fax: (203) 285-4117