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U. S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020
SRP Section: 04.03 - Nuclear Design
SRP Section: 04.04 - Thermal and Hydraulic Design
SRP Section: 04.06 - Functional Design of Control Rod Drive System
Application Section: FSAR Ch. 4

QUESTIONS for Reactor System, Nuclear Performance and Code Review (SRSB)

04.03-7

FSAR Tier 2 Section 4.3.1.1 describes the fuel burnup limit in terms of a peak fuel rod burnup limit of 62 GWD/MTU. However, FSAR Table 4.3-1 lists the burnup limit for gadolinia fuel rods as 55 GWD/MTU.

Provide additional information on the 55 GWD/MTU design peak rod burnup for gadolinia fuel rods, including technical rationale and basis, its application, and its effect on the assembly burnup limit.

This information is needed in order to clarify and ensure the adequacy of the U.S. EPR fuel burnup limit, and further to ensure that the design will be consistent with GDC10.

04.03-8

FSAR Tier 2 Figure 4.3-6 through Figure 4.3-9 show the placement of fuel rods within the seven different fuel assembly designs of the initial core. Fuel assembly designs B1, B2, C2, and C3 contain gadolinia fuel rods that are located face-adjacent to water-filled guide tubes. Such placement of gadolinia-bearing fuel rods could result in preferential uneven depletion of the gadolinia, thus posing a challenge to the lattice physics model calculations and hence introduction of modeling error.

Provide justification for the placement of gadolinia-bearing fuel rods face-adjacent to the water-filled guide tubes, including description of the validation of the MICBURN-3/CASMO-3G codes for such design. Benchmark results or reference to applicable licensing topical reports should be provided.

This information is required in order to ensure that the analytical methods utilized in the design of the U.S. EPR fuel assembly are capable of accurately modeling the design and that they are thus consistent with GDC 10.

04.03-9

FSAR Tier 2 Section 4.3.2.1 states that the core average enrichment is set by the fuel cycle length and energy requirements.

Explain the rationale for that statement and state if and how the fuel assembly design burnup limit, which is based on the peak rod burnup limit, is considered in the determination of core average enrichment.

This information is required in order to ensure accuracy and completeness of the technical description contained in the FSAR.

04.03-10

In FSAR Section 4.3.2.2.5, the applicant states that fuel manufacturing practices for modern nuclear fuel have largely eliminated the potential for significant fuel densification and that a power spike factor of 1.0 is used for the U.S. EPR fuel, based on referenced topical report BAW-10163P-A, "Core Operating Limit Methodology for Westinghouse-Designed PWRs."

Explain how fuel densification is largely eliminated due to modern fuel manufacturing practices, including a description of the pellet manufacturing process, densification propensity, analytical predictions of fuel densification, and post-irradiation examinations results, if available.

This information is necessary in order to determine whether the effects of fuel densification on local power peaking need to be included in the U.S. EPR nuclear design analysis, in accordance with criteria contained in NUREG-0800 SRP Section 4.3.

04.03-11

The applicant states in FSAR Section 4.3.2.2.7 that F_Q is continuously monitored using the fixed incore SPND detectors, but there is insufficient detail for the staff to complete its evaluation of how F_Q is monitored. There is no Technical Specification limiting condition for operation or surveillance requirement on F_Q. The Technical Specifications Bases Section B 3.2.2 states that F_Q is a direct input parameter to the LOCA analysis; however, the Technical Specification surveillance requirement is on $F^N_{\Delta H}$.

Describe how F_Q is continuously monitored, and explain why there is no Technical Specification surveillance requirement on F_Q. Also, clarify the relationship of F_Q to $F^N_{\Delta H}$ relative to the LOCA analysis.

This information is needed in order to clarify the basis for hot channel factor F_Q and to ensure adequate core monitoring for protection of the core against postulated LOCA events, per the requirements of GDC 13.

04.03-12

In FSAR Tier 2 Section 4.3.2.2.6, the applicant states that $F^N_{\Delta H}$ is used for establishing control rod patterns and control rod bank sequencing, as well as fuel loading patterns. FSAR Tier 2 Section B 3.2 describes the basis for the $F_{\Delta H}$ Technical Specification limiting condition for operation to be also associated with the LOCA analysis.

Provide a complete description of the design basis for hot channel factor $F^N_{\Delta H}$, including its relationship to hot channel factor F_Q, and how it is monitored during power operation.

This information is needed in order to clarify the basis for hot channel factor $F^N_{\Delta H}$ and to thus ensure compliance with GDC 13.

04.03-13

FSAR Tier 2 Section 4.3.2.2.6 states that crud deposition or boron buildup on fuel rods can affect core power distribution and that continuous monitoring of DNBR, LPD, and axial offset would detect changes in power distribution caused by these phenomena.

Provide an explanation of the crud and boron deposition phenomena and describe the effects on core power distribution and core reactivity, how it is accounted for in the design analysis, and how it is detected during reactor operation. This information is required in order to ensure that the requirements of GDC 10 and GDC 12 are met.

04.03-14

FSAR Tier 2 Section 4.3.2.7 states that Technical Specifications require periodic comparison of measured versus calculated power distributions throughout cycle lifetime.

Provide reference to the FSAR Tier 2 Chapter 16 section that requires periodic comparison of measured versus calculated power distributions throughout cycle lifetime.

This information is needed in order to ensure accuracy of the technical description contained in the FSAR.

04.03-15

In FSAR Tier 2 Section 4.3.2.2.7 the applicant states that uncertainties in the aeroball measurement system readings are addressed through comparison of symmetric channels.

Provide a description of how the symmetric aeroball measurements are utilized to address the aeroball measurement system uncertainties. Also, explain how the uncertainties are treated in the determining the measured power distributions and associated fuel thermal parameters such as DNBR, LPD, and hot channel factors.

This information is needed in order to ensure adequate treatment of on-line power distribution measurement uncertainties and hence compliance with GDC 10, GDC 12, and GDC 13.

04.03-16

FSAR Tier 2 Section 4.3.2.2.8 provides a description of the physics tests that are included in initial test program described in FSAR Section 14.2. The staff has evaluated the physics tests described in FSAR Section 14.2 against the Tier 1 FSAR ITAAC requirements and against the guidance contained in Regulatory Guide 1.68, and finds the following testing not included:

a. Determination of core linear power density (LPD) and DNBR in order to verify capability of the core monitoring system for use in complying with Technical Specification limiting conditions for operation,

b. Performance of a pseudo-rod-ejection test to verify calculational models and accident analysis assumptions, and

c. Demonstration of the capability of the incore neutron flux instrumentation to detect rod misalignment equal to or less than the Technical Specifications limits for control rod misalignment. (The staff notes that FSAR Tier 2 Section 16 B 3.1.7 states that detection of a rod misalignment may be done through use of the incore aeroball measurement system.)

Address each of the test items listed above by either adding a test requirement to the initial test program in FSAR Tier 2 Section 14.2, or by justifying not including such test requirement.

The information is needed in order to ensure adequacy of the core physics tests which are part of the initial test program described in FSAR Tier 2 Section 14.2.

04.03-17

In FSAR Tier 2 Section 4.3.2.4, the applicant states that the soluble boron used for reactivity control is natural B^{10} abundance, whereas in FSAR Tier 2 Section 6.8.3 the applicant states that enriched boron is utilized in the extra borating system. FSAR Tier 2 Table 6.8-1 lists the extra boron system B^{10} enrichment to be 37%.

Confirm the use of natural boron for normal (chemical and volume control system) core reactivity control. Also, describe the plant controls for keeping the natural boron separate from the enriched boron, the station procedures for verifying the isotopic level of B^{10} and the plant controls required to ensure the correct boron solutions are used in their respective systems.

This information is required in order to clarify the isotopic boron solutions used for both normal plant operation (chemical and volume control system) and for accident conditions (extra borating system) and to ensure adequate controls are in place to manage the boron isotopic solutions.

04.03-18

FSAR Tier 2 Section 4.3.3.2.4 states that the steady-state NEMO methodology has been benchmarked against the PRISM methodology, and refers to ANP-10286P, "U.S. EPR Rod Ejection Accident Methodology" for comparison of results. The steady-state NEMO methodology was approved by the staff in BAW-10180A, "NEMO-Nodal Expansion Method Optimized;" however, it is not explicitly included in the staff's approval of ANP-10263P-A, "Codes and Methods Applicability Report for the U.S. EPR." Moreover, ANP-10286P, "U.S. EPR Rod Ejection Accident Methodology" has not yet been approved by the staff.

Provide an explanation of the application of the NEMO methodology to the U.S. EPR and its validation basis.

This information is needed in order to clarify which analytical methodologies are utilized in the nuclear design of the U.S. EPR and to ensure that only approved methodologies are utilized.

04.04-21

FSAR Tier 2 Section 4.4 does not address thermal-hydraulic conditions during shutdown or low-power operations, per the guidance contained in SRP Section 4.3.

Provide an analysis of the thermal-hydraulic conditions of the U.S. EPR reactor system during shutdown and low-power operation, including analysis of a rapid boron dilution event during shutdown conditions, in accordance with NUREG-1449.

This information is needed in order to ensure adequate evaluation of the thermal-hydraulic conditions under all reactor operating conditions, including shutdown and low-power operation.

04.04-22

In FSAR Tier 2 Section 4.4.1, the applicant refers to “a small fraction of fuel rods damaged” following a postulated accident. GDC 10 and GDC 12, applicable to FSAR Tier 2 Section 4.4, require that specified acceptable fuel design limits not be exceeded during normal operation and AOOs. Guidelines for acceptable levels of fuel damage following a postulated accident are provided in NUREG-0800 SRP 15.0 in terms of off-site dose relative to 10 CFR 100 criteria, also consistent with the criteria stated in FSAR Tier 2 Chapter 15.

Explain what is meant by “a small fraction of fuel rods damaged” following a postulated accident and how this criterion relates to the applicable regulatory requirements.

This information is needed in order to ensure the correctness of the design criteria and consistency with regulatory requirements.

04.04-23

GDC 10 and GDC 12 specify requirements in terms of the specified acceptable fuel design limits. In FSAR Tier 2 Section 4.4.1, the applicant states that specified acceptable fuel design limits are not exceeded during normal operation and AOOs; however, there is no definition of specified acceptable fuel design limits applicable to the U.S. EPR provided in the Tier 2 FSAR.

Define the specified acceptable fuel design limits applicable to the U.S. EPR.

This information is needed to ensure correctness of the thermal-hydraulic design evaluation provided by the applicant in FSAR Tier 2 Section 4.4.4.3 and compliance with GDC 10 and GDC 12 requirements.

04.04-24

In FSAR Tier 2 Section 4.4.4.1, two critical heat flux correlations, ACH-2 and BWU-N, are described via reference to the applicant’s licensing topical reports ANP-10269P and BAW-10199P-A, respectively. (Note that the former is now approved and should carry the suffix –A.) Although the staff has approved the application of these two correlations as part of the LYNXT methodology to the U.S. EPR fuel in ANP-10263P-A, the applicant states in ANP-10263P-A that the longer length fuel rod of the U.S. EPR fuel assembly will be accounted for in the development of the U.S. EPR CHF correlation.

Confirm the applicability of the ACH-2 and BWU-N critical heart flux correlations to the U.S EPR fuel and explain the requirements, plans, and schedule for development of the U.S. EPR critical heat flux correlation.

This information is necessary to ensure adequacy of the DNBR calculations used to ensure compliance with GDC 10 and GDC 12 requirements.

04.04-25

In FSAR Tier 2 Section 4.4.9.3 the applicant refers to a flow mal-distribution penalty applied to the hot assembly.

Provide a description of the flow mal-distribution penalty, including technical basis, derivation, and application in the calculation of DNBR. Also, address whether the mal-distribution penalty includes the effects of asymmetry of the vessel inlet nozzles.

This information is needed to ensure adequacy of the DNBR calculations used to ensure compliance with GDC 10 and GDC 12 requirements.

04.04-26

The applicant states in FSAR Tier 2 Section 4.4.2.6.4 that hydraulic loads on the fuel assemblies were evaluated, but no description of the analysis or results are presented.

Provide a description of the hydraulic loads analysis for the U.S. EPR fuel and vessel components for normal operating conditions and design basis accident conditions, including the effects of assembly lift.

This information is needed in order to satisfactorily evaluate fuel hold down requirements, as called for in NUREG-0800 SRP 4.4 area of review.

04.04-27

In FSAR Tier 2 Section 4.4.2.9.1, the applicant states that uncertainties due to manufacturing tolerances are included in the heat flux hot channel factor, F_Q used in the COPERNIC code calculation of fuel temperature.

Describe the fuel fabrication uncertainties and tolerances that are accounted for in the COPERNIC calculations, and state specifically how pellet chipping is treated.

This information is needed to ensure adequacy of the analyses used to ensure compliance with GDC 10 and GDC 12 requirements.

04.04-28

In FSAR Tier 2 Section 4.4.2.9.4, reference is made to Reference 7, the applicant's licensing topical report ANP-10269P, "The ACH-2 CHF Correlation for the U.S. EPR." This report has been approved by the staff since issuance of the U.S. EPR FSAR.

Attach the suffix" – A" onto the FSAR Tier 2 Section 4.4 Reference 7 licensing topical report number designation.

This information is needed to update the FSAR with current information.

04.04-29

The material provided by the applicant in FSAR Tier 2 Section 4.4 does not address the effect of crud on DNBR and reactor system pressure drop, as called for in NUREG-0800 SRP 4.4 guidance.

Describe how the thermal-hydraulic design accounts for the effects of crud in the DNBR and core pressure drop calculations throughout the reactor coolant system, including how associated drops in reactor coolant system flow are monitored.

This information is needed to ensure adequate thermal-hydraulic analyses have been performed for the U.S. EPR and that the requirements of GDC 10 and GDC 12 are satisfied.

04.04-30

In FSAR Tier 2 Section 4.4.2.10, the applicant states that the design value of the heat flux hot channel factor, F_Q, does not include a specific allowance for azimuthal tilt.

Provide justification for not including an allowance for azimuthal tilt on the heat flux hot channel factor, F_Q.

This information is needed to ensure that specified acceptable fuel design limits are not exceeded during normal operation and AOOs, as required in GDC 10 and GDC 12.

04.04-31

The plant configuration data for the thermal-hydraulic systems are incorporated by reference to the applicable sections of the Tier 2 FSAR. However, some of the data called for in Regulatory Guide 1.206 are not included.

Provide the additional data called for in Regulatory Guide 1.206, in particular the following items as listed in Section C.I.4.4.3.1 of the regulatory guide:

a. Item (4) total volume of each plant component, including emergency core cooling system (ECCS) components, with sufficient detail to define each part (e.g., downcomer, lower plenum, upper head) of the reactor vessel and steam generator,

b. Item (5) length of the flowpath through each volume,

c. Item (6) height and liquid level of each volume,

d. Item (8) lengths and sizes of all safety injection lines, and

e. Item (10) steady-state pressure and temperature distribution throughout the system.

This information is needed to ensure adequacy of the thermal-hydraulic systems described in the Tier 2 FSAR.

04.04-32

The FSAR Tier 1 Section 2.2.1 ITAAC 7.2 requires a four-pump coastdown test to verify compliance against design coastdown flow given in the accompanying Table 2.2.1-4. The FSAR Tier 2 Section 15.3.2 four-pump loss of flow safety analysis is performed assuming a flow coastdown given by FSAR Tier 2 Figure 15.3-5. The test criterion does not appear to bound the flow coastdown data assumed in the Chapter 15 safety analysis. In addition, the staff notes that although Initial Plant Test # 183 "Post-Core Reactor Coolant System Flow Baseline" described in FSAR Tier 2 Section 14.2 includes flow coastdown measurements, there is no criterion on four-pump flow coastdown.

Provide a resolution of the apparent discrepancy between the FSAR Tier 1 Section 2.2.1 ITAAC 7.2 acceptance criterion and the FSAR Tier 2 Figure 15.3-5 data utilized in the FSAR Tier 2 Chapter 15 safety analysis. Also, either justify not including four-pump coastdown as a criterion in Initial Plant Test # 183 or add the criterion.

This information is necessary to ensure adequacy of the DNBR transient analysis of the design basis four-pump loss of flow event described in FSAR Tier 2 Chapter 15 and compliance with GDC 10.

04.04-33

The applicant's description of the effects of fuel rod bow references two licensing topical reports, BAW-10147P-A and BAW-10186P-A. These two reports are also referenced by the applicant's licensing topical report ANP-10263P-A, "Codes and Methods Applicability to the U.S. EPR." In its discussion of fuel rod bow, ANP-10263P-A states that because the U.S. EPR fuel assembly design is characterized by longer length with two additional spacer grids, the fuel rod bow correlation for the U.S. EPR fuel will be provided in a later submittal.

Considering the apparent need for a U.S. EPR-specific fuel rod bow correlation that has not yet been developed, explain the statements made in FSAR Tier 2 Section 4.4.4.1.5 concerning the application of fuel rod bow penalties on the hot channel factors defined for the U.S. EPR. Also provide additional information on the U.S. EPR correlation, including methodology, application, and development schedule.

This information is needed in order to ensure proper treatment of fuel rod and assembly bow in the thermal-hydraulic analysis of the U.S. EPR fuel.

04.04-34

FSAR Tier 2 Section 4.4.2.9.1 states that the effect of fuel densification is included in the calculation of the total uncertainty associated with fuel and cladding temperatures.

Provide an explanation of the effect of fuel densification in the U.S. EPR fuel and describe how its effects are accounted for in the thermal margin analyses for linear power density and DNBR.

This information is needed in order to ensure proper treatment of fuel densification in the thermal margin analysis of the U.S. EPR fuel.

04.04-35

Initial Plant Test # 188, “Post-Core Incore Instrumentation,” includes testing of the core exit thermocouples; however, no acceptance criterion is given to confirm functionality of the core exit thermocouples. The core exit thermocouples are part of the reactor instrumentation required under 10 CFR 50.34 (f) (2) (viii).

Either discuss the reason for not including an acceptance criterion on core exit thermocouple functionality in Initial Plant Test # 188, or add an acceptance criterion.

This information is needed to ensure adequacy of the initial plant testing described in FSAR Tier 2 Section 14.2 and compliance with 10 CFR 52.47 (b) (1).

04.06-3

In Section 4.6.1 of the FSAR, it is stated that the CRDM equipment is designed and qualified to operate in the reactor vessel cavity environment.

Please provide the reference(s) that document the qualification of the CRDM to operate in the reactor vessel cavity environment. This information is necessary to confirm that the CRDM equipment is capable of performing its design function within the environmental conditions specified under GDC 4.

04.06-4

According to Section 4.6.1, the protection of control rod drive mechanism from internally generated missiles is provided by concrete secondary shield wall and reinforced concrete missile shield slabs.

What analysis confirms the protection to the CRDMs from internally generated missiles? This information is required to confirm the compliance that SSCs be designed to meet the requirements of GDC 4 as it relates to protection against dynamic effects including the effects of missiles.

04.06-5

The seismic protection of the control rod drive system presented in the FSAR Section 4.6.1 refers to FSAR Section 5.4.14 on component supports. These two sections describe the restricted displacements during seismic events and design-basis accidents. The natural ground motion based seismic events are part of the seismic analysis of the plant. However, large missiles, such as an aircraft crash, can also cause acceleration events inside the containment, where the direction of the force is different (i.e. from above and the side) than in the natural ground motion seismic events (from below and the side).

How does EPR design demonstrate that the impact direction has been considered in the seismic analysis of the control rod drive mechanism component supports? This information is required to confirm the compliance that SSCs be designed to meet the requirements of GDC 4 as it relates to protection against dynamic effects including external events.

04.06-6

The control rod drive system (CRDS) according to FSAR Section 4.6.2 is part of the environmental qualification program, and thus designed to operate in extreme conditions.

How has AREVA verified that the maximum design pressure and temperature of the CRDS are not exceeded in the safety analyses where the CRDS is assumed to remain operative, and what are the maximum values? This information is required to assure compliance with GDC 4 regarding the CRDM compatible under its adverse postulated environmental conditions.

04.06-7

The analog position indication system is the only safety-related measurement in the CRDS. In Section 4.6.2, it has been written that failure of the analog rod position indicators to operate properly would not prevent the RCCAs from inserting into the core or resulting in inadvertent withdrawal of the core. Being the only safety related measurement, the analog position indicator failure would be worthwhile to consider in the safety analysis. In the safety analysis in Section 15.4 of the FSAR, this assumption was not found to have been considered in connection with any of the analysis presented.

What are the possible safety analyses or other reasoning behind the statement that failure of the analog rod position indicators to operate properly would not prevent the RCCAs from inserting into the core or resulting in inadvertent withdrawal from the core? This information is needed to assess that the single failure criteria requirements of GDC 25 are met.

04.06-8

The Compliance to the Single Failure Criterion of the Reactor Trip system is discussed in Section 7.2.2.3.1 of the FSAR and in the Table 7.2.2 FMEA Summary for Reactor Trip, which conclude that the protection system maintains the ability to perform the reactor trip function in the presence of any credible single failure of an input sensor, functional unit of the protection system or reactor trip device. Table 7.2.2 includes as one failure type single failure of sensor-RCCA position measurement. However, it is unclear, if the failure of the safety-related analog position measurement or the non-safety related digital position measurement failure has been considered.

Has the failure of the safety-related analog position measurement or the non-safety related digital position measurement failure been considered as the failure type? This information is needed to assess that the single failure criteria requirements of GDC 25 are met.

04.06-9

The mechanical adequacy Section 4.6.3 refers to Section 3.9.4.4 where it is stated that to confirm the mechanical adequacy of the CRDS, a prototype testing program was created. According to Section 3.9.4.4, this program comprises performance tests, stability tests and endurance tests. No reference to the results of this test program is given, and it is not clear from Section 3.9.4.4 how the status and results of this test program support Section 4.6 and other sections of the FSAR. The tests are reported to verify the performance of the equipment under a broad range of conditions. However, the range and conditions are not specified.

What is the status and results of the CRDS prototype testing program and the range of environmental conditions that support the FSAR? This information is necessary to evaluate the performance of the CRDS to ensure an extremely high probability of accomplishing their safety functions in accordance with GDC 29.

04.06-10

In Section 4.6.4, it is written that mechanical overheating of the CRDM causes failure of only one RCCA from inserting into the core by gravity, and the other CRDMs remain functional.

What is the rationale for this statement and the reference to the analysis that supports this conclusion? This information is required to confirm the CRDM is designed to fail into a safe state in accordance with GDC 23 in the event of adverse conditions or environments.