ArevaEPRDCPEm Resource

From:	WELLS Russell D (AREVA NP INC) [Russell.Wells@areva.com]
Sent:	Monday, April 06, 2009 2:55 PM
To:	Getachew Tesfaye
Cc:	Pederson Ronda M (AREVA NP INC); BENNETT Kathy A (OFR) (AREVA NP INC); DELANO Karen V (AREVA NP INC)
Subject:	Response to U.S. EPR Design Certification Application RAI No. 196, FSAR Ch 9
Attachments:	RAI 196 Response US EPR DC.pdf

Getachew,

Attached please find AREVA NP Inc.'s response to the subject request for additional information (RAI). The attached file, "RAI 196 Response US EPR DC.pdf" provides technically correct and complete responses to all 13 questions.

Appended to this file are affected pages of the U.S. EPR Final Safety Analysis Report in redline-strikeout format which support the response to RAI 196 Questions 9.01.01-11, 9.01.01-12, and 9.05.01-68.

The following table indicates the respective pages in the response document, "RAI 196 Response US EPR DC.pdf" that contain AREVA NP's response to the subject questions.

Question #	Start Page	End Page
RAI 196 — 9.01.01-6	2	2
RAI 196 — 9.01.01-7	3	4
RAI 196 — 9.01.01-8	5	6
RAI 196 — 9.01.01-9	7	7
RAI 196 — 9.01.01-10	8	8
RAI 196 — 9.01.01-11	9	9
RAI 196 — 9.01.01-12	10	10
RAI 196 — 9.01.01-13	11	11
RAI 196 — 9.01.01-14	12	12
RAI 196 — 9.01.01-15	13	13
RAI 196 — 9.01.01-16	14	14
RAI 196 — 9.01.01-17	15	16
RAI 196 — 9.05.01-68	17	17

This concludes the formal AREVA NP response to RAI 196, and there are no questions from this RAI for which AREVA NP has not provided responses.

Sincerely,

(Russ Wells on behalf of)

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From: Getachew Tesfaye [mailto:Getachew.Tesfaye@nrc.gov]
Sent: Friday, March 06, 2009 5:23 PM
To: ZZ-DL-A-USEPR-DL
Cc: Christopher VanWert; Shanlai Lu; Joseph Donoghue; Jason Carneal; Edward McCann; Robert Radlinski; Peter Hearn; Joseph Colaccino; ArevaEPRDCPEm Resource
Subject: U.S. EPR Design Certification Application RAI No. 196 (2260, 2237),FSAR Ch. 9

Attached please find the subject requests for additional information (RAI). A draft of the RAI was provided to you on February 26, 2009, and on March 6, 2009, you informed us that the RAI is clear and no further clarification is needed. As a result, no change is made to the draft RAI. The schedule we have established for review of your application assumes technically correct and complete responses within 30 days of receipt of RAIs. For any RAIs that cannot be answered within 30 days, it is expected that a date for receipt of this information will be provided to the staff within the 30 day period so that the staff can assess how this information will impact the published schedule.

Thanks,
Getachew Tesfaye
Sr. Project Manager
NRO/DNRL/NARP
(301) 415-3361

Hearing Identifier: AREVA_EPR_DC_RAIs
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Response to

Request for Additional Information No. 196 (2260, 2237), Revision 0

03/06/2009

U. S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020
SRP Section: 09.01.01 - Criticality Safety of Fresh and Spent Fuel Storage and Handling
SRP Section: 09.05.01 - Fire Protection Program
Application FSAR Ch. 9

QUESTIONS for Reactor System, Nuclear Performance and Code Review (SRSB)
QUESTIONS for Fire Protection Team (SFPT)

Question 09.01.01-6:

Section 4.6.2 of Holtec Technical Report UN-TR-08-001(P) discusses the core operating parameters used in the fuel depletion calculations, and refers to Table 4.6.2 for a list of core parameters. The value of in-core fuel assembly pitch of 21.402 cm (8.426 inches) given in Table 4.6.2 appears inconsistent with the value of 21.5036 cm (8.466 inches) provided in FSAR Table 4.4-1.

In order to verify the appropriate fuel design data were used in the criticality analyses, verify the in-core fuel assembly pitch used in the depletion analyses and any corrections to the analyses made and the application revised, as necessary. This information is required in order to ensure compliance with 10 CFR 50.68(b) and GDC 62.

Response to Question 09.01.01-6:

The value of 8.426 inches is the grid-to-grid width of the U.S. EPR assembly at cold conditions. The assembly pitch is correctly stated in the U.S. EPR FSAR as 8.466 inches. For the criticality analyses performed by Holtec and documented in UN-TR-08-001(P), an assembly pitch of 8.432 inches was used, which is 0.034 inches less than the actual pitch of 8.466 inches. The difference is 0.034 inches/2 x 25.4 mm/inch = 0.43 mm on each side of the assembly. This has negligible impact in the neutronics calculations using CASMO.

FSAR Impact:

Neither the U.S. EPR FSAR nor Holtec Technical Report UN-TR-08-001(P) will be changed as a result of this question.

Question 09.01.01-7:

Section 4.6.3 of Holtec Technical Report UN-TR-08-001(P) states that axial burnup distributions for several assembly average burnups are used in the criticality analyses, with reference to Table 4.6.3 of the report. Section 4.8.2.1 states the calculations are performed with a constant axial burnup and with an axial burnup distribution.

The staff has reviewed the axial burnup distributions as provided in Table 4.6.3 of the Holtec report, and finds that the axial burnup distributions are essentially identical and nearly uniform, at differing assembly average burnup levels.

In order to verify the adequacy of the criticality analyses and ensure compliance with 10 CFR 50 Appendix A, General Design Criterion 62, the staff requires the following information:

a. Justify the axial burnup distributions given in Holtec Technical Report UN-TR-08-001(P) Table 4.6.3 adequately bound the range of burnup distributions and assembly average burnups anticipated for the U.S. EPR fuel assembly.

b. Assess the need for a reactivity uncertainty (ΔK_{eff}) to cover possible variations of axial burnup distribution from those assumed in the criticality analyses.

Response to Question 09.01.01-7:

a) The axial burnup distributions were generated by taking the most conservative axial shapes generated from cycles 1, 2, 3, and an equilibrium cycle for the 18-month U.S. EPR core designs. For each cycle, every assembly burnup profile was binned, representing average assembly burnups of 25, 30, 40, and 50 GWd/mtU. The tolerance bands on assembly burnup were:

   +0/-5 GWd/mtU for the 25 GWd/mtU point.
   +/-5 GWd/mtU for the 30 and 40 GWd/mtU points.
   +8/-5 GWd/mtU for the 50 GWd/mtU point.

   This resulted in fuel assembly burnups up to 58 GWd/mtU, and fuel rod burnups up to 62 GWd/mtU being reflected in the criticality analysis. Since limiting criticality conditions occur at the fuel assembly ends for higher burnup assemblies, axial shapes were chosen with burnup on the fuel ends minimized. For the shapes collected in each bin, a composite shape was constructed that reflected the least burned top and bottom shapes. Additionally, the U.S. EPR fuel assemblies have axial blankets with much lower fuel enrichments on both assembly ends (typically 2-3 wt% U^{235}). The criticality analysis was performed with the assembly ends modeled at the higher fuel enrichment used in the core midplane. Additionally, the fuel cycle depletions were modeled with a 7% rod insertion for control bank D, further reducing the burnup on the assembly ends for some limiting assemblies. Limiting axial fuel and moderator temperatures from cycles 1, 2, 3, and the equilibrium cycle were also used in the generation of burned fuel isotopic data.

b) As described in the response to part (a), bounding axial burnup distributions were developed for use in the criticality analysis for the spent fuel storage racks. Because a bounding axial burnup distribution was used, there is no need for a reactivity uncertainty. Any variations in the axial burnup distribution from those assumed in the criticality analyses would result in a reduction in reactivity.

FSAR Impact:

Neither the U.S. EPR FSAR nor Holtec Technical Report UN-TR-08-001(P) will be changed as a result of this question.

Question 09.01.01-8:

Section 4.6.5 of Holtec Technical Report UN-TR-08-001(P) summarizes the Region 1 and Region 2 spent fuel storage rack design, with reference to Table 4.6.4. In comparing the rack design parameters in Table 4.6.4 against the design information provided in Section 2 of Holtec Technical Report UN-TR-08-001(P), the staff notes discrepancies in the reported widths of the poison panels for both Region 1 and Region 2 racks, and in the cell pitch of the Region 2 racks. In addition, there is insufficient fuel storage rack design information in the application for the staff to complete its review.

In order for the staff to complete its verification of the fuel storage rack data used in the criticality analysis per the guidelines of SRP 9.1.1 subsection III, provide the following information for the new fuel storage racks and the Region 1 and Region 2 spent fuel storage racks:

a. Minimum Boron-10 areal density for the Metamic™ neutron poison material.
b. Storage rack design dimensions, including manufacturing tolerances, for the parameters utilized in the criticality analyses, and
c. Justification of any differences in the values used in the criticality analyses, as presented in Table 4.6.4 of the Holtec report.

This information is required in order to ensure compliance with 10 CFR 50.68(b) and GDC 62.

Response to Question 09.01.01-8:

a) There is sufficient information in Table 4.6.4 to calculate the ^{10}B areal density from the parameters provided. The following table provides the ^{10}B areal density for nominal dimensions and minimum characteristics.

B_4C Content	Thickness	^{10}B areal density ($g\ ^{10}B/cm^2$)
Nominal	Nominal	0.0310
Minimum	Nominal	0.0300
Nominal	Minimum	0.0298
Minimum	Minimum	0.0289

b) All storage rack dimensions necessary to support the criticality safety analysis are provided in Table 4.6.4, including tolerances. The neutron absorber panel width presented in Table 2.5.1 and 2.5.2 is the correct value. The storage cell pitch for the Region 2 racks presented in Table 4.6.4 is the correct value.

c) The neutron absorber width presented in Table 4.6.4 is 0.25 inches smaller than the value in Table 2.5.1 and Table 2.5.2, and is the value used in the criticality analysis (7.25 inches). This is conservative as it reduces the amount of neutron absorber credited. The storage cell pitch for the Region 2 racks in Table 2.5.2 has been rounded up from 9.028 to 9.03 inches. The value in Table 2.5.2 is provided for informative purposes only and is not used in any analysis. The correct value of 9.028 inches is used in the criticality analysis.

FSAR Impact:

Holtec Technical Report UN-TR-08-001(P) will not be changed as a result of this question.

Question 09.01.01-9:

In order to comply with the requirements of 10 CFR 50.68(b) and GDC 62, provide a criticality analysis of the new fuel storage facility.

Response to Question 09.01.01-9:

The new fuel storage racks are identical to the Region 1 spent fuel storage racks (i.e., flux trap racks with neutron absorber). Therefore, the analysis presented for the Region 1 racks is bounding for the new fuel storage facility. Optimum moderation conditions do not occur in storage racks with neutron absorbers, as demonstrated in:

J.M. Cano, R. Caro, and J.M. Martinez Val, "Supercriticality Through Optimum Moderation in Nuclear Fuel Storage," *Nuclear Technology,* Volume 48, May 1980.

FSAR Impact:

Neither the U.S. EPR FSAR nor Holtec Technical Report UN-TR-08-001(P) will be changed as a result of this question.

Question 09.01.01-10:

The staff notes an apparent discrepancy, as follows. FSAR Section 9.1.2.2.2 states that the normal water level of spent fuel pool is 7.01 m (23 feet) above the tops of the stored fuel. Technical Specification 3.7.14 requires water to be maintained ≥ 7.01 m (≥ 23 feet) only during fuel moves as a protective measure against a fuel handling accident and for shielding during movement of spent fuel.

Resolve the apparent discrepancy between the stated spent fuel pool normal water level and the Technical Specifications spent fuel pool water level requirement.

Response to Question 09.01.01-10:

As specified in the Response to RAI 84, Question 09.01.02-11:

> "the top of the fuel assemblies seated in the spent fuel storage racks will be at elevation 33 feet 2 inches. The nominal water level in the pool is at elevation 62 feet 4 inches. This elevation corresponds to a nominal water depth of 45 feet 7 inches, and a nominal height of water above the spent fuel assemblies seated in the storage racks of approximately 29 feet."

This is well above the minimum height of 23 feet specified in U.S. EPR FSAR Tier 2, Chapter 16, Technical Specification 3.7.14 during movement of irradiated fuel in the spent fuel pool. U.S. EPR FSAR Tier 2, Section 9.1.2.2.2 was revised in the Response to RAI 84, Question 09.01.02-11 to reflect a nominal spent fuel pool water depth of 45 feet 7 inches and approximately 29 feet of water above the top of the spent fuel assemblies seated in the storage racks.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 09.01.01-11:

Technical Specification 4.3 contains the following specifications that are inconsistent with the fuel storage design as provided in Sections 2 and 4 of Holtec Technical Report UN-TR-08-001(P):

a. Technical Specification 4.3 specifies a nominal spent fuel storage rack cell pitch of 28.6512 cm (11.28 inches) as compared to values of 27.686 cm (10.9 inches) and 22.9311 cm (9.028 inches) for the Region 1 and Region 2 spent fuel storage racks, respectively per the Holtec report;

b. Technical Specification 4.3 specifies a nominal new fuel storage rack cell pitch of 28.6512 cm (11.28 inches) as compared to a value of 27.686 cm (10.9 inches) per Table 2.5.1 of the Holtec report; and

c. Technical Specification 4.3 specifies a maximum capacity of the spent fuel storage pool of 1,121 fuel assemblies as compared to 1,360 assemblies per the Holtec report.

Resolution of the above discrepancies is necessary in order for the staff to verify that the appropriate fuel storage rack design data were used in the criticality safety analyses and thereby assure compliance with 10 CFR 50.68(b) and GDC 62.

Response to Question 09.01.01-11:

The values provided in U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications 4.3.1.1c, 4.3.1.2d, and 4.3.3 are bracketed values. As addressed in U.S. EPR FSAR Tier 2, Chapter 16, Section 16.0, Introduction:

> "Brackets are used to identify information or parameters that are plant specific or are based on preliminary design information. A COL applicant that references the U.S. EPR design certification will replace preliminary information provided in brackets of the Technical Specifications and Technical Specification Bases with plant specific values."

A Reviewer's Note will be added to U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications, Section 4.3 to indicate that design details of the spent fuel storage racks will be provided by the COL applicant.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 4.3 will be revised as described in the response and indicated on the enclosed markup.

Question 09.01.01-12:

Table 4.6.1 of Holtec Technical Report UN-TR-08-001(P) identifies a maximum fuel enrichment of 5 wt% U^{235} used in the spent fuel pool criticality analyses. Technical Specification Bases B 3.7.16 states that the spent fuel racks are designed to accommodate fuel with a maximum nominal enrichment of 5 wt % U^{235} with a maximum tolerance of +/- .05 wt%. Provide the treatment of manufacturing variance in enrichment level in the analyses.

This information is required in order for the staff to complete its verification of the fuel storage rack data used in the criticality analysis per the guidelines of SRP 9.1.1 subsection III and thereby assure compliance with 10 CFR 50.68(b) and GDC 62.

Response to Question 09.01.01-12:

The maximum enrichment that can be shipped or loaded into any commercial fuel storage rack in the U.S. is currently 5.00 wt% U^{235} including the enrichment tolerance. Therefore, if the enrichment tolerance is +/-0.05 wt% U^{235}, then the maximum nominal enrichment is 4.95 wt% U^{235} unless special assay fuel is requested (enrichment tolerance of 0.017 wt% U^{235}). Regardless of the enrichment tolerance, the fuel storage racks were evaluated at the maximum allowed enrichment of 5.00 wt% U^{235}. U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications, Bases Section 3.7.16 will be changed to indicate that the maximum as-built enrichment is 5.00 wt% U^{235}.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications, Bases Section 3.7.16 will be revised as described in the response and indicated on the enclosed markup.

Question 09.01.01-13:

In order to demonstrate compliance with 10 CFR 50.68(b) and GDC 62 under credible abnormal conditions, the vertical drop of a 5 wt % U^{235} fuel assembly onto a Region 1 stored fuel assembly must be shown to result in $K_{eff} \leq 0.95$.

Section 4.8.1.5.3 of Holtec Technical Report UN-TR-08-001(P) states that a vertical drop of a fuel assembly onto a Region 1 stored fuel assembly could impart a compressive force onto the stored fuel assembly, reducing the water-to-fuel ratio, and thereby reducing reactivity.

Provide an additional explanation of the effect of the compressed stored fuel assembly on water-to-fuel ratio in order for the staff to complete its evaluation of the dropped fuel assembly event.

Response to Question 09.01.01-13:

In the event that a fuel assembly in transit is dropped onto the top of a fuel assembly in the spent fuel storage rack, there would be limited effect on reactivity for the following reasons:

1. The active fuel regions of the dropped assembly and assembly in the spent fuel storage rack would remain separated by more than 20 inches, precluding neutron coupling.
2. Any damage to the fuel assembly would be minimal and at most produce a small change in the spacing of the fuel rods within the fuel assembly.

The reactivity effect of changing the spacing of the fuel rods would be bounded by the accident that considers dropping a fuel assembly in an empty storage location.

FSAR Impact:

Holtec Technical Report UN-TR-08-001(P) will not be changed as a result of this question.

Question 09.01.01-14:

In order to demonstrate compliance with 10 CFR 50.68(b) and GDC 62 under credible abnormal conditions, the vertical drop of a fresh 5 wt % U^{235} fuel assembly either into a vacant Region 2 spent fuel storage cell or onto a Region 2 stored fuel assembly must be shown to result in $K_{eff} \leq 0.95$.

Holtec Report UN-TR-08-001(P) Section 4.8.2.8.3 states that the dropped fuel assembly could either slightly compress the stored fuel assembly or deform the storage rack baseplate, resulting in an increase in reactivity. The application states, however, that such reactivity increase would be small compared to that of a misloading of a fresh fuel assembly into the Region 2 racks and is therefore bounded by the misloading event.

Provide either additional justification to support the claim that the Region 2 fuel assembly misloading event bounds the vertical dropped fuel assembly event, or provide explicit analyses of the Region 2 vertical dropped fuel assembly event.

Response to Question 09.01.01-14:

Explicit analysis of the dropped or misplaced fresh fuel assembly into a Region 2 storage cell is provided in Section 4.8.2.8.4. The maximum required soluble boron to offset this accident is indicated in Table 4.8.5.

FSAR Impact:

Holtec Technical Report UN-TR-08-001(P) will not be changed as a result of this question.

Question 09.01.01-15:

In order to demonstrate compliance with 10 CFR 50.68(b) and GDC 62 under credible abnormal conditions, the mislocation of a fresh 5 wt % U^{235} fuel assembly in the Region 2 spent fuel pool must be shown to result in $K_{eff} \leq 0.95$. The application states in Section 4.8.2.8.5 of Holtec Technical Report UN-TR-08-001(P) that a mislocation of a fresh 5 wt % U^{235} fuel assembly outside of the Region 2 spent fuel storage racks is bounded by the mislocation analysis for the Region 1 racks where the mislocated assembly is next to a fresh assembly. This conclusion, however, is not sufficiently justified because the Region 1 racks are designed with a flux trap for added reactivity control, whereas the Region 2 racks are standard non-flux trap design.

Provide either further justification to support the assertion that the Region 1 mislocated assembly analysis bounds the Region 2 mislocated assembly condition, or provide explicit analyses of the Region 2 mislocated assembly condition.

Response to Question 09.01.01-15:

The wording in Section 4.8.2.8.5 that a mislocated fresh fuel assembly outside the Region 2 racks is bounded by the mislocation analysis for Region 1 was presented as written for the following reasons:

1. The mislocated fresh fuel assembly outside the Region 2 rack would be bounded by the misloaded fuel assembly in the Region 2 rack (i.e. placing a fresh fuel assembly in the center of the Region 2 rack, surrounded by spent fuel would produce a larger reactivity increase then placing the fuel assembly outside the rack where neutron leakage is present).
2. The mislocated fuel assembly in the Region 1 rack produces a larger reactivity increase, and subsequent higher soluble boron requirement than the misloaded fuel assembly in the Region 2 rack (see Table 4.8.2 and Table 4.8.5 to notice that the soluble boron requirement in the Region 1 rack accident is higher).

Therefore, the statement that the mislocation of a fresh fuel assembly outside the Region 2 racks is bounded by the mislocation of a fresh fuel assembly in the Region 1 racks is accurate. It may have been clearer to simply state that the mislocation of a fuel assembly outside the rack is bounded by the misloading of a fresh fuel assembly inside the rack. In either case, the bounding accident is captured in the analysis with a minimum boron content as shown in Table 4.8.2.

FSAR Impact:

Holtec Technical Report UN-TR-08-001(P) will not be changed as a result of this question.

Question 09.01.01-16:

The application, including FSAR Section 9.1.1 and Section 4.0 of Holtec Technical Report UN-TR-08-001(P), “Criticality Evaluation,” does not adequately address criticality safety during fuel handling evolutions.

In order to comply with the requirements of 10 CFR 50.68(b) and GDC 62, provide a criticality analysis of the fuel handling system, including normal and credible abnormal configurations during new fuel handling, refueling operations, spent fuel storage operations, and spent fuel transfers to dry cask storage.

Response to Question 09.01.01-16:

The criticality analysis for the new and spent fuel storage racks considered the range of applicable fuel handling evolutions. These evolutions included a horizontal fuel assembly drop, a vertical assembly drop into a fuel cell, a misloaded fresh fuel assembly, and a mislocated fresh fuel assembly. The fuel assembly elevator and transfer canal is limited to a single assembly. Misplacing a fresh assembly next to these would not produce a reactivity increase that would begin to approach the reactivity effect of deborating the entire pool, which is the bounding reactivity accident. Fuel loading into a shipping cask for transport or dry storage is not possible for at least 10 years following plant operation and fuel discharge. Fuel movement into fuel shipping and transportation containers has not been evaluated because the design features of these containers will be significantly different in the year 2036 than those containers currently available. Fuel movement will be evaluated prior to movement into those containers.

The analysis and results presented in Section 4.0 of UN-TR-08-001(P), "Criticality Evaluation" covers the spent fuel and new fuel storage racks designed by Holtec International. New fuel handling and refueling operations are not covered in this document. Spent fuel transfers to dry cask storage would be covered by an approved Part 72 Certificate of Compliance and corresponding storage cask FSAR.

FSAR Impact:

Neither the U.S. EPR FSAR nor Holtec Technical Report UN-TR-08-001(P) will be changed as a result of this question.

Question 09.01.01-17:

In the staff's safety evaluation of Holtec International Report HI-2022871, "Use of Metamic™ in Fuel Pool Applications," dated June 17, 2003, the staff requires implementation of a coupon sampling program to ensure consistent performance of the Metamic™ material. Section 3.4 of Holtec Technical Report UN-TR-08-001(P) describes an in-service surveillance program similar to the coupon sampling program prescribed in the staff's earlier safety evaluation; however, certain provisions of the staff's safety evaluation are not explicitly addressed in Holtec Technical Report UN-TR-08-001(P), e.g., technique for measuring initial B_4C content of the coupons, simulation of scratches on the coupons, effects of any fluid movement and temperature fluctuations of the pool water on the coupons.

In order to comply with the staff's safety evaluation of Holtec International Report HI-2022871, "Use of Metamic™ in Fuel Pool Applications," dated June 17, 2003, the applicant's description of the Metamic™ in-service surveillance program must include a discussion of all the aspects of the in-service surveillance program listed in the subject safety evaluation (Reference: "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Holtec International Report HI-2022871 Regarding Use of Metamic in Fuel Pool Applications," Facility Operating License Nos. DPR-51 and NPF-6, Entergy Operations, Inc., Docket No. 50-313 and 50-368, USNRC, June 2003). The description of the Metamic™ in-service surveillance program may be supplied by the COL applicant, in which case a COL Information Item should be added to the DC FSAR.

Response to Question 09.01.01-17:

The surveillance program described in Section 3.4 of the Holtec report is essentially the same as similar surveillance programs previously approved by the NRC for multiple spent fuel pool uses of Metamic. As for the specific provisions questioned in the RAI, specific responses are as follows:

1. Technique for measuring initial B_4C content of the coupons - The initial B_4C content is calculated from material batching records generated during manufacturing of the Metamic. The weight of B_4C is divided by the total constituent materials weight to obtain a B_4C weight percentage.
2. Simulation of scratches on the coupons - The coupons are excised pieces of the panels themselves, so any surface markings resulting from the material manufacturing process would automatically be present on the coupons. Additionally, coupons are stamped with a serial number, which provides nucleation sites for corrosion. Finally, the coupons are mounted on a "tree" in a manner that creates a crevice between the coupon and a washer, which provides a site for potential crevice corrosion.
3. Effects of any fluid movement and temperature fluctuations of the pool water on the coupons - The coupons are installed in the cell of a rack in the pool. As a result, they would be subjected to the same pool temperature fluctuations as the Metamic installed in the racks themselves. Because the Metamic in the racks is encased in stainless steel while the coupons are exposed, the coupons would conservatively be exposed to higher velocities (although erosion of Metamic is not possible, given it's monolithic state, so the velocity field is largely irrelevant).

FSAR Impact:

Holtec Technical Report UN-TR-08-001(P) will not be changed as a result of this question.

Question 09.05.01-68:

Response to RAI 20 Question 09.05.01-20 states that “An alternative approach to fixed, self-contained eight-hour rated battery powered lighting units is taken for illuminating the MCR and RSS in support of post-fire safe shutdown. Both locations are illuminated by the special emergency lighting system. The special emergency lighting system receives power from redundant emergency diesel generator-backed uninterruptible power supplies, thus providing continuous illumination. Adequate lighting is available in the MCR or RSS, as necessary, to facilitate post-fire safe shutdown of the plant.” U.S. EPR FSAR needs to clarify that in the event of a fire that adversely affects special emergency lighting equipment, cables, or power supplies for fire areas outside of the MCR or RSS that adequate special emergency lighting is available in the MCR or RSS, as necessary, to facilitate post-fire safe shutdown of the plant.

Response to Question 09.05.01-68:

U.S. EPR FSAR Tier 2, Section 9.5.1.2.1 will be revised to include the requested information. The last sentence of the paragraph that describes the alternative approach to fixed, self-contained eight-hour rated battery powered lighting units will be replaced with the following:

> “In the event a fire outside the MCR or RSS adversely affects special emergency lighting equipment, cables or power supplies, adequate special emergency lighting is available in the MCR or RSS, as necessary, to facilitate post-fire safe shutdown of the plant.”

FSAR Impact:

U.S. EPR FSAR Tier 2, Section 9.5.1.2.1 will be revised as described in the response and indicated on the enclosed markup.

U.S. EPR Final Safety Analysis Report Markups

Communications

For the purposes of fire fighting and operational post-fire safe shutdown activities, the U.S. EPR plant relies on the portable wireless communication system described in Section 9.5.2. The system is multi-channeled and is capable of interfacing with the public address and digital telephone systems. Use of the portable wireless communication system does not interfere with the communications capabilities of the plant security force. Fixed components of the portable wireless communication system are protected as necessary from fire damage to provide seamless communication capability in vital plant areas with the exception of radio sensitive locations. The capability of the fire brigade or operations personnel to communicate using the portable wireless communication system from within radio sensitive locations is not desired to preclude potential spurious operation of equipment.

Emergency Lighting

Section 9.5.3 contains design information for the U.S. EPR lighting system.

Portable hand-held, eight-hour rated lights are provided for use by the fire brigafe in accordance with RG 1.189, Rev. 1, Section 4.1.6.2b. The egress route from the MCR to the RSS is illuminated by independent fixed, self-contained eight-hour rated battery powered lighting units. Other post-fire safe shutdown activities performed by operators outside the MCR and RSS are supported by independent fixed, self-contained eight-hour rated battery lighting units at the task locations and in access and egress routes.

An alternative approach to fixed, self-contained eight-hour rated battery powered lighting units is taken for illuminating the MCR and RSS in support of post-fire safe shutdown. Both locations are illuminated by the special emergency lighting system. The special emergency lighting system receives power from redundant emergency diesel generator backed uninterruptible power supplies, thus providing continuous illumination. 09.05.01-68 In the event a fire outside the MCR or RSS adversely affects special emergency lighting equipment, cables, or power supplies, adequate special emergency lighting is available in the MCR or RSS, as necessary, to facilitate post-fire safe shutdown of the plant.

Ventilation System Design Considerations

The design of the heating, ventilation and air conditioning (HVAC) systems are in accordance with SRP 9.5.1 (Reference 37) and RG 1.189. Safety-related HVAC systems are also designed in accordance with NFPA 90A (Reference 16). The HVAC design provides reasonable assurance that smoke, hot gases, or fire suppression agents (e.g., gaseous suppression agents) will not migrate into other fire areas and adversely affect safe shutdown capabilities, including operator actions.

4.0 DESIGN FEATURES

4.1 Site Location

[A COL Applicant that references the U.S. EPR design certification will provide site-specific information for Section 4.1, Site Location.]

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 241 fuel assemblies. Each assembly shall consist of a matrix of fuel rods clad with a zirconium based alloy with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

4.2.2 Control Rod Assemblies

09.01.01-11

The reactor core shall contain 89 control rod assemblies. The control material shall be silver indium cadmium as approved by the NRC.

---REVIEWER'S NOTE---
The design of the spent fuel storage racks is to be provided by the COL applicant. The required boron concentration will be provided as a part of the spent fuel rack design.

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in FSAR Section 9.1; and

c. A nominal [11.28] inch center to center distance between fuel assemblies placed in the spent fuel storage racks.

B 3.7 PLANT SYSTEMS

B 3.7.16 Spent Fuel Storage

---Reviewer's Note---
The design of the spent fuel storage racks is to be provided by the COL~~A~~ applicant. The required spent fuel storage configuration will be provided as a part of the spent fuel rack design.

BASES

BACKGROUND	The spent fuel storage facility as described in Ref. 1 has a capacity of at least [1020] fuel assemblies. The spent fuel storage racks are designed to accommodate fuel with a maximum ~~nominal~~ enrichment of 5.0 wt% U-235 ~~(maximum tolerance of +/- 0.05 wt%)~~. The spent fuel storage racks are designed for unrestricted spent fuel assembly storage. 09.01.01-12
APPLICABLE SAFETY ANALYSES	----------------------------------REVIEWER'S NOTE-------------------------------- The design of the spent fuel storage racks is the responsibility of the COL applicant. A COL applicant that references the U.S. EPR design certification will demonstrate that the design satisfies the criticality analysis requirements for the spent fuel storage racks. -- The criticality analysis shows that the fuel remains subcritical under all credible abnormal conditions. The design shows acceptable prevention of an increase in effective multiplication factor (k-effective) beyond safe limits based on the guidelines in Ref. 2. The configuration of fuel assemblies in the fuel storage pool satisfies Criterion 2 of 10 CFR 50.36(~~d~~c)(2)(ii).
LCO	09.01.01-12 Unrestricted storage of fuel assemblies within the spent fuel pool is allowed provided that the maximum ~~nominal~~ U-235 enrichment is equal to or less than 5.00 weight percent. This ensures the k-effective of the spent fuel pool will always remain less than 0.95 assuming the pool is flooded with [unborated water].
APPLICABILITY	This LCO applies whenever any fuel assembly is stored in the spent fuel pool.