Advanced LWR Program
UTILITY STEERING COMMITTEE

May 8, 1991

Document Control Desk
USNRC
Washington, D.C. 20555

Subject: Project No. 669 - Request for Additional Information on Chapter 4, "Reactor System", Volume III of the Utility Requirements Document

References: (1) J. H. Wilson letter to E.E. Kintner, March 8, 1991

Gentlemen:

This is the response to the NRC staff request for additional information (Reference 1).

Please call John D. Trotter at EPRI, (415) 855-2786, if you have any questions.

Sincerely,

E. E. Kintner, Chairman
ALWR Steering Committee

cc: James H. Wilson (with attachment)
G. Bockhold
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Reference: J. H. Wilson to E. E. Kintner, dated March 8, 1991 (SRXB)

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION VOLUME III, CHAPTER 4

NRC Question

440-75 Paragraphs 3.1.3.4 and 3.1.3.5 state that the BWR reactor pressure vessel (RPV) interfaces with the reactor water cleanup system and the reactor shutdown cooling system, respectively. Chapter 3 indicates that the RWCU system also serves the purpose of the RSDC system. Please clear the confusion as to whether the RWCU and RSDC systems are physically one or two separate systems?

Response

The RWCU is a BWR system and the RSDC is a PWR system. The RWCU provides a decay heat removal function for the BWR, in addition to the cleanup function. The BWR and PWR have common requirements for the shutdown cooling function given in Volume III, Chapter 3, Paragraph. 9.2.1.

NRC Question

440-76 Paragraph 3.1.3.6 states that the RPV interfaces with the diverse reactivity control system (Chapter 3) by receiving and distributing soluble poison. Does this paragraph mean the interface with the standby liquid control system discussed in Chapter 5?

Response

The paragraph intended to say that the interface is with the standby liquid control system discussed in Chapter 5. This error will be corrected in both Volumes II and III in the next revision of the document.

Reference: J. H. Wilson to E. E. Kintner, dated March 8, 1991 (SRXB)

NRC Question

440-77 Paragraph 3.2.2.2 requires carry-under of steam in the downcomer water not to exceed an acceptable limit at rated reactor output at the normal water level. What is the EPRI's quantitative "acceptable limit" and its basis? If the designer is required to derive the acceptable limit, what are the EPRI guidelines or criteria for deriving the limit?

Response

The carry-under of steam in the downcomer water is normally set at about 1% at rated reactor output and at normal water level. It is a detail design parameter that is established by a combination of analysis and experience. The exact value, which is a function of the detail design of the core and fuel, is not a dominant parameter in core performance. A requirement will be added to limit the carry-under to 1% at rated reactors output and at normal water level.

NRC Question

440-78 Paragraphs 3.2.2.4 and 3.2.2.5 require the RPV internal design configuration, including chimneys at the reactor core exit, to provide for full natural circulation rates which shall be stable, with safety margin, in all normal operating regions of operation.

(a) Do the current approved computer codes have the analytical capabilities or limitations for the thermal hydraulics and stability analyses of the specified design configuration for normal operations and transient and accident conditions?

(b) What are the requirements regarding the experimental verification of the stability and thermal-hydraulic performance of the design configuration and the analytical code capabilities?

c) What system performance tests will be required for the first-of-a-type system design to validate thermal-hydraulic design and other issues such as stability?

Response

(a) The computer codes used have no limitations that would prohibit the development of a fully satisfactory thermal-hydraulic design of the SBWR. Relative to stability, it is acknowledged that the analytical prediction of BWR stability (SBWR or otherwise) is subject to uncertainties and, for this reason, SBWR policy is to provide defense-in-depth as described in the response to question 440.1. (This question was included in NRC RAI of February 22, 1991).

(b) Chapter 1, Section 11.2, Technology Base, requires successful proven technology be employed throughout the plant including system and component designs. The intent is to utilize the large LWR experience base. Proven technology is defined as design and analysis techniques which have been successfully demonstrated, preferably through several years of operation in existing LWRs. Chapter 1, Section 2.5.3.1 requires the designer to use best estimate models that have been benchmarked against operating plant data or appropriate test data.

(c) Chapter 1, Section 7.9, System Completion and Startup Testing, as well as Volume I Paragraph 3.3.5, require the Plant Designer to prepare a set of tests, inspections and analysis and associated acceptance criteria which will demonstrate that the plant has been constructed and will be operated in conformity with Commission regulations, the combined license, and the Atomic Energy Act. In addition, the technical basis for the completeness of the set of inspections, tests, and analysis for the specified acceptance criteria is required.

It is the responsibility of the Plant Designer to demonstrate how the requirements of (b) and (c) above will be met. The licensing process includes a requirement for definitions of Inspections, Tests, Analysis, and Acceptance Criteria (ITAAC) that need to be performed in order to verify the performance of the new design.

NRC Question

440-79 Paragraph 3.3.2.3.7 specifies the requirements on the arrangement and materials of in-core housing and internal stub tubes. Are there similar requirements for control rod drive housings and stub tubes?

Response

The requirements for the control rod drive housings and stub tubes are given in Chapter 1, Paragraph 5, and Chapter 4, Paragraphs 3.3.1.7.1 and 5.3.3.

NRC Question

440-80 Paragraph 4.2.1.2.1 requires the BWR core characteristics to allow stable operation, which is defined as the core and channel decay ratios, not exceeding 0.4 and 0.3, respectively. Because the values of the decay ratios depend on the calculational methodologies, including the codes and uncertainty treatment, the specified decay ratio limits for operational stability determination should be tied to the specific methodologies used for the decay ratio calculations. This paragraph should be expanded to clarify the stability requirement.

Response

The decay ratio limits for operational stability determination should be tied to the specific methodologies used for the decay ratio calculations. The limits specified in the Requirements Document are based on the methods and procedures currently being finalized by the BWR Owners Group (FABLE frequency domain calculations, normal and transient events, more conservative inputs, etc.). This reference already exists in the Rationale Section of Chapter 4, paragraph 4.2.1.2.1, which states that the analysis methods and procedures should be consistent with current NRC accepted techniques.

NRC Question

440-81 Paragraph 4.2.1.2.2 requires the instrumentation to be provided to detect core-wide and/or regional instability. Please specify the parameters that are required to be monitored.

Response

The parameter used to monitor for core-wide and local instabilities will be local neutron flux (LPRM readings). The logic and instrumentation is not yet final but will be based on that currently being developed by the BWR Owner's Group for operating plants.

NRC Question

440-82 Paragraph 4.2.1.3.1 required the minimum critical power ratio operating limit to be specified such that at least 99.9 percent of the fuel rods in the core will not be subject to transition boiling during the most severe identified transients. Clarification should be made on the most severe identified transients because different acceptance criteria are allowed for the ANS Conditions III and IV events.

Response

Clarification on transient and accident classification is given in Volume III, Chapter 1, Paragraph 2.3.2.3 and Table 1.2-1.

However, this paragraph applies to the most limiting identified anticipated operating occurrence and therefore ANS Conditions III and IV events are not applicable.

NRC Question

440-83 Paragraphs 4.2.1.3.3 and 7.2.1.3.1 require the fuel design to meet the requirements of 10CFR50.46 and Appendix K for postulated LOCA.

(a) Does this requirement of Appendix K mean to preclude the use of the realistic, best-estimate evaluation method under the new ECCS rules? Why? Who (i.e., EPRI or the designer) will decide whether the BE or Appendix K approach will be used for the LOCA analysis?

(b) Appendix K was originally written for existing plants. Provide a list of Appendix K features that may not be applicable to the passive ALWR designs, discuss why they do not apply and how EPRI intends to address them.

(c) If the best-estimate evaluation model will be used for LOCA calculation, what is the EPRI guidance with regard to the guidelines of Regulatory Guide 1.157, "Best-Estimate Calculation of Emergency Core Cooling System Performance," and "Code Scaling, Applicability and Uncertainty (CSAU) Evaluation Methodology" described in NUREG-1230, "Compendium of ECCS Research for Realistic LOCA Analysis"?

Response:

a) No, there is no intent to preclude the use of realistic, best-estimate evaluation methods on the passive plant. As written, the requirement could be construed to preclude their use. Accordingly, Paragraphs 4.2.1.3.3 and 7.2.1.3.1 will be revised by removing the words "and 10CFR50, Appendix K." The specific choice would be made by the plant designer. These corrections apply to both Volumes II and III.

(b) Such an evaluation of Appendix K would depend on the specifics of a particular plant design and would be done by the plant designer as part of the design-certification process. This evaluation would only be needed if the plant designer elects to use 10CFR50, Appendix K instead of best-estimate methods.

(c) The ALWR Requirements Document requires compliance with Regulatory Guide 1.157. (Table B. 1-2 of Appendix B to Chapter 1). NUREG-1230, "Compendium of ECCS Research for Realistic LOCA Analysis," summarizes and provides a general reference for the extensive research performed on LOCA/ECCS performance of LWRs. The ALWR Requirements Document takes no specific position on this document. However, it is expected that plant designers would consider the extensive research results summarized in NUREG-1230 during the development and NRC-approval phases of their efforts to establish their best-estimate analysis methodology. The ALWR Requirements Document requires (Paragraphs 2.5.1.1 and 2.5.2.2 of Chapter 1) that NRC-approved analysis methods be used for licensing analyses.

NRC Question

440-84 Paragraph 4.2.1.4.2 specifies that no preconditioning be required of the fuel for maneuvering, and indicates that improved designs of fuel and reactivity control systems, achieving this requirement have been demonstrated. What analyses are required of the designer to demonstrate that the fuel designs do not establish or affect the planned maneuvering limits?

Reference: J. H. Wilson to E. E. Kintner, dated March 8, 1991 (SRXB)

Response

The ALWR Requirements Document does not specifically required any analysis be performed; it specifies the result desired. The specific analyses needed for the plant designer to assure he meets the requirement will depend on the details of the fuel designs and fuel management schemes adopted. These are outside the scope of the ALWR Requirements Document. The intent of this requirement is to assure that previously limiting fuel characteristics (such as PCI) will not limit ALWR maneuvering. It is expected that the plant designer would include mitigating design features (e.g., fine-motion control-rod drives and barrier fuel) as part of his response to this requirement.

NRC Question

440-85 Paragraph 4.2.1.5 requires a core power-flow relationship relative to control rod position to be prepared as part of the plant design. Is the core power-flow relationship required to be independent of the fuel design as in Paragraph 4.2.1.4.2? How would this be accomplished?

Response

There is no requirement that the core power-flow relationship be independent of the fuel design. The requirement in paragraph 4.2.1.4.2 is related to time limitations in the rate of change of power imposed on early core and fuel designs. The limitation was referred to as, "preconditioning the fuel". The requirement is intended to obtain a core and fuel design that does not require such limitations.

Reference: J. H. Wilson to E. E. Kintner, dated March 8, 1991 (SRXB)

NRC Question

440-86 Paragraph 4.2.1.6.2 requires the BWR core design to allow for extended cycle operation at reduced power or with reduced feedwater temperature. Since reduced feedwater temperature will cause the plant to be more thermal-hydraulically unstable, what analysis has been done to balance the benefits of extending cycle operation against the undesirability of decreasing core stability by reducing feedwater temperature?

Response

The possibility of feedwater temperature reduction to extend cycle length will be considered in the SBWR stability calculations. Analysis procedures will be consistent with those recently developed by the BWR Owner's Group and used by operating plants. The decay ratio for the case with end-of-cycle feed water temperature reduction will be calculated and will be subject to the limits specified in Chapter 4, Paragraph 4.2.1.2.1.

NRC Question

440-87 Paragraphs 4.2.2.2 and 7.2.2.2 require the basic fuel mechanical designs be capable of peak bundle-average burnups of at least 50,000 and 60,000 MWD/MTU, respectively, for BWR and PWR fuels. The current burnup limits for the existing plants are approximately 44,000 and 55,000 MWD/MTU for BWRs and PWRs, respectively. Does EPRI have sufficient high burnup data to support fuel integrity in areas of fission gas release and cladding corrosion due to oxidation and hydriding?

Response

The minimum burnup requirements have been chosen based on current industry trends towards achieving higher burnups and the industry's assessment of what the demonstrated capabilities are expected to be at the time of an

ALWR or .er. These requirements are not based on any specific data base extending to the specified burnup levels. The burnup limits actually placed on ALWR fuel would be those demonstrated and licensable by the specific fuel vendor at the time of use.

The intent of this requirement is to assure that reactor internals and fuel assemblies are designed to accommodate high-burnup effects. For example, the reactor internals and fuel assemblies must provide sufficient space to accommodate fuel-assembly and fuel-grid growth. Fuel-rod plenum volume must be sufficient to cope with the fission-gas releases at high burnups. With regard to cladding corrosion, it is recognized that this is a major consideration at high burnups. However, both the 15% margin requirement and the 600°F hot-leg temperature limit will likely have a substantial and beneficial effect on cladding corrosion.

<u>NRC Question</u>

440-88 Paragraph 4.2.4 requires the BWR fuel channel to be designed for a lifetime of two fuel-bundle in-core residence times with the objective of reusing channels on more than one fuel assembly. What is the EPRI guidance regarding channel box bow effect on the critical power ratio for channel boxes in their second bundle life-time operation? This concern was addressed in NRC Bulletin No. 90-02, Loss of Thermal Margin Caused by Channel Box Bow.

<u>Response</u>

Paragraph 4.2.4 establishes the <u>objective</u> of a two-cycle channel lifetime, but does not <u>require</u> it. The effects of channel bow, as well as other irradiation effects such as corrosion and irradiation creep, must be properly accounted for such that core operation remains within its licensed limits. The two-cycle channel lifetime has been demonstrated as feasible for selected channels built to suitably stringent requirements and operated under a fuel

management procedure which accounts for channel irradiation effects.

NRC Question

440-89 Paragraph 7.3.1.10 requires the PWR fuel assembly to be designed so that fuel-rod spacing closure does not exceed 50 percent of the nominal spacing under operating conditions throughout life. Why is there no comparable requirements specified in Section 4.3.1 for BWR fuel design? What are the EPRI guidelines in dealing with the effects of fuel-rod bow on the specified acceptable fuel design limits?

Response

Fuel-rod bow has been a particular issue for PWR fuel; for BWRs, it has been less of a concern. Accordingly, a specific requirement for the BWR was judged to not be warranted. Fuel-rod bow is a design-specific parameter and each fuel vendor has his own fuel-rod bow model. The effects of fuel-rod bow must be included in those licensing analyses for which fuel-rod bow is an important effect. The specific models and methods of accounting for rod bow are outside the scope of the ALWR Requirements Document.

NRC Question

440-90 Paragraphs 4.3.1 and 7.3.1 specify the requirements on BWRs and PWRs respectively, regarding the material, control of hydriding and fretting corrosion of the fuel cladding. What are the requirements to deal with pellet-cladding interaction concern?

Reference: J. H. Wilson to E. E. Kintner, dated March 8, 1991 (SRXB)

Response

Pellet-cladding interaction is dealt with as a top-level requirement common to both BWRs and PWRs. Please see Paragraph 2.3.3.4 of Chapter 4. Paragraph 2.3.3.4 requires that proven design features be used to provide resistance to pellet-clad interaction.

NRC Question

440-91 Paragraph 5.2.1.4 specifies the requirement of the scram performance and design of the CRD and hydraulic scram system. Is it intended that the system shall accommodate a hafnium type, or a boron carbide type, and other types of proven control rods? The word "hafnium" is missing in the requirement.

Response

It is intended that the CRD and hydraulic system be designed to accommodate any type of control rod that has been proven by successful experience in a BWR. The word "hafnium" was inadvertently omitted after the word "accommodate". This error will be corrected at the next revision of the document.

NRC Question

440-92 Paragraph 5.2.1.8 requires the establishment of certain parameters related to the BWR CRDS, including the stroke length, and normal withdrawal and insert speeds.

(a) Clarify what is meant by "scram time (minimum requirements)"?

(b) Does the electric motor drive have only one speed for normal reactivity control and scram functions?

(c) Considering the electric motor drive is intended to provide a backup to the hydraulic scram for ATWS events, would there be a required limit on the maximum electric motor drive scram time?

(d) Should there be a requirement to take into consideration of possible severe power peaking tilt at the top of the core caused by slow reactivity insertion by the electric motor drive scram during ATWS events?

Response

(a) The "scram time (minimum requirement)" means the longest time allowed from the initiation of a scram signal for the control rod to reach various insertion percentages up to 100 percent. The numerical values are dependent on a number of detail core and fuel design considerations.

(b) The CRDS has a fast hydraulic scram insertion capability which has the scram time referred to in (a) above. In addition, there is a completely independent, single speed motor driven drive train that is used for normal reactivity control. This motor drive train also provides the independent backup to the hydraulic scram for the ATWS event.

(c) There is a minimum time that will be required to mitigate that event. As stated in the rationale of Paragraph 5.2.1.8, these features must be closely coordinated with the core and fuel design.

(d) It is not likely that the insertion of all rods at the maximum rate of the motor drive would result in severe power peaking tilt at the top of the core because the time for the rods to reach 1/4 insertion, which will take the power to virtually zero, will be quite short.

Reference: J. H. Wilson to E. E. Kintner, dated March 8, 1991 (SRXB)

NRC Question

440-93 Paragraph 6.3.1.4 indicates that the requirements of those RPV nozzles other than the reactor coolant and instrumentation nozzles remain to be established. When will they be established?

Response

The intent of the requirement is to note that additional RPV nozzles may be needed depending on the results of system design activities for systems such as RHR, HPI, etc., in particular passive plant designs. The requirement will be clarified in both Volumes II and III as follows:

"6.3.1.4 Need for Additional RPV Nozzles

In addition to ...... RPV nozzles may be required. The need for these nozzles and any requirements associated with them will be established consistent with requirements of Chapters 3, 4, and 5. In addition:"

For a particular design, the need for any additional RPV nozzles and any requirements associated with them would be expected to be defined during the preliminary design process, and before the detailed design is started.

NRC Question

440-94 Paragraph 6.3.3.1 requires a means to be provided for measuring the coolant level in the RPV during depressurized, shutdown conditions to meet the needs of various maintenance activities. However, no requirement is specified for reactor vessel level instrumentation system (RVLIS) for post accident conditions. EPRI's position for not providing RVLIS for the passive ALWRs is discussed in Paragraph 2.4.1 of Appendix B to Chapter 1. However, the new arguments were not sufficient to cause the staff to alter the NRC position requiring a RVLIS for existing PWRs, nor is there anything significantly different about the passive ALWR design which would alleviate the NRC's requirement for a RVLIS. Therefore, a RVLIS should be provided as required by NUREG-0737 Item II.F.2.

Response

Elimination of the requirement to include RVLIS in the passive PWR is a goal of the ALWR Program. This goal was established because industry experience to date indicates that a RVLIS does not provide benefits commensurate with the complexity, maintenance burden, personnel radiation exposure, and expense it adds to plant operation. Paragraph 2.4.1 of Appendix B to Chapter 1 of the ALWR Requirements Document provides a discussion of the technical considerations. The ALWR Program considers that these technical considerations warrant elimination of the RVLIS requirement for passive PWRs.

For further discussion of the RVLIS issues raised by NRC Staff, please refer to our response to question 440-31.

NRC Question

440-95 Paragraph 7.2.1.6 requires the reactor to be designed with the capability for continued operation for four hours at reduced power despite an unintended single control rod drop during power operation.

(a) At what reduced power level would the reactor continue to operate?

(b) Will the reactor be scrammed if the dropped rod is not recovered within four hours?

Response

(a) The permissible power level will depend on the details of the designs of the fuel, the control rods, and the fuel management as well as the time in life. The principal intents of the requirement are to avoid an immediate reactor scram and to provide sufficient time to recover the control rod without initiating a reactor scram, thereby reducing the chance of having to subject the plant to a shutdown/startup cycle.

(b) The specific operator action is outside the scope of the ALWR Requirements Document.