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September 21, 1983
SVP-0128-83

Mr. Harold R. Denton
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Docket Nos.: STN 50-546
STN 50-547
Construction Permit Nos.:
CPPR - 170
CPPR - 171

Marble Hill Nuclear Generating Station - Units 1 and 2
Application for Licenses
NRC Request for Additional Information

Dear Mr. Denton:

On August 24, 1983, the NRC submitted to Public Service Company of Indiana, Inc. (PSI) requests for additional information to complete its evaluation of the PSI and Wabash Valley Power Association, Inc. (WVPA) application for an operating license for the Marble Hill Nuclear Generating Station. A review of these questions has been made. Several of the questions pertain to sections of the FSAR that are replicate of the Byron FSAR or are similar to Byron FSAR questions. These areas of the FSAR should have been reviewed for Byron's license application and, therefore, by the rules of replication, should not be readdressed in the Marble Hill Nuclear Generating Station's application. Advanced copies of the responses to these questions are being provided at this time. These responses will be included in the December amendment. One (1) signed original and fifteen (15) copies of this letter and attachments are provided for your use.

If you have any questions, please contact me at your convenience.

PUBLIC SERVICE COMPANY OF
INDIANA, INC. for itself and
as agent for WABASH VALLEY
POWER ASSOCIATION, INC.

S. W. Shields
Senior Vice President-
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SWS/GAS/bak

Enclosure

B001
1/1

cc: J. F. Schapker (w/o enc)
D. G. Eisenhut (w/o enc)
T. M. Novak (w/o enc)
B. J. Youngblood (w/o enc)
P. W. O'Connor (w/o enc)

QUESTION M220.1

The spent fuel pool concrete structure, listed in Table 3.2-1, should also be included in FSAR[1] Section 3.8.4 and the information regarding its design and construction provided.

RESPONSE

Although not explicitly listed, the design and construction of the replicate spent fuel pool structure is provided in subsections 3.8.4.4, 3.8.5.1.2 and applicable criteria for a seismic Category I structure.

Additional information can be provided during the structural design audit.

The Byron SER (NUREG-0876) subsection 3.8.3 concludes that "..... the design of safety-related structures other than the containment is acceptable....."

1/ Marble Hill Nuclear Generating Station Units 1 and 2, Final Safety Analysis Report.

QUESTION M220.2

The descriptive information of the internal structures is not in (3.8.3) accordance with the provisions of R. G. 1.70, "Standard Format ....." Provide sufficient information, illustrated by sketches, in FSAR Section 3.8.3, to allow the staff to perform a meaningful review of internal structures of the containment.

Reference to Sections 1.2 and 3.9 of the FSAR is not acceptable since the sketches in Section 1.2 provide only general outline of the plant. Description of the structural aspects of the internal structures should be included in FSAR Section 3.8.3.

RESPONSE

The response to this question is provided in replicate Byron Question Q130.42. This information is also provided in the structural design audit. The Byron SER (NUREG-0876) Subsection 3.8.3 concluded that " ..... the design of the containment internal structures is acceptable ...."

QUESTION M220.3

The descriptive information of Category I structures other than containment is not in accordance with the provisions of the R. G. 1.70. Provide sufficient information, illustrated by sketches in the FSAR Section 3.8.4 to enable the staff to perform a meaningful review. Referencing the FSAR Section 1.2 which illustrates the general layout of the plant is insufficient in detail for a structural review.

RESPONSE

The response to this question is provided in replicate Byron Question Q130.43. The Byron SER (NUREG-0876) subsection 3.8.3 concludes that "..... the design of safety-related structures other than the containment is acceptable ....."

QUESTION M220.6

Indicate if an investigation has been made to ensure that inclusion of additional masses of degrees of freedom would not result in more than ten percent increase in responses. Alternately, the number of degrees of freedom may be taken equal to twice the number of modes with frequencies less than 33 cps.

RESPONSE

The criteria used to ensure an adequate number of masses or degrees of freedom is described in Byron replicate Questions Q130.8 and Q130.20. The Byron SER (NUREG-0876) subsection 3.7.2 concludes that "..... the plant design is acceptable ....."

QUESTION M220.8

Provide the specific mass and frequency ratios, and the criteria used for decoupling subsystems from the primary systems or structures. Demonstrate quantitatively that these criteria are comparable to those contained in the SRP, Section 3.7.2.3 and/or justify and assess the impact of any deviation therefrom.

RESPONSE

The response to this question is provided in Byron replicate Question Q130.7 and Q130.19. The Byron SER (NUREG-0876) subsection 3.7.2 concludes that ".....the plant design is acceptable ....." A compliance analysis is provided in Appendix F page F 3.7.2-1.

QUESTION M220.9

Describe the method used to determine overturning moments of Category I structures. It is the staff's position that determination of the design moment for overturning should incorporate the following:

(a) three components of motion
(b) conservative consideration of vertical and lateral seismic forces

Indicate if the method of computation of design overturning moments complies with the above position.

RESPONSE

Discussions on overturning moments of Category I structures are provided in replicate subsections 3.8.5.3, 3.8.5.5.2.1 and Byron Question Q130.46. The Byron SER (NUREG-0876, subsection 3.8.4 concludes that "..... the design of the seismic Category I foundations is acceptable ....."

QUESTION M220.10

In Section 3.7.2.10 of the FSAR you stated that "..... each individual floor framing beam of the building was designed statically for 1.5 times the acceleration value corresponding to the fundamental frequency of the beam from the applicable wall response spectrum". The staff's position (see SRP Section 3.7.2.II, .1.a) is that to obtain an equivalent static load of a structure, equipment or component, which can be represented by a simple model, a factor of 1.5 applied to the peak acceleration of the applicable floor response spectra should be used. Justify this apparent deviation from the SRP and assess its impact on the results of the analysis.

RESPONSE

The response to this replicate question is provided in Byron Question Q130.29. The Byron SFR (NUREG-0876) subsection 3.7.2 concluded that "..... the plant design is acceptable ....."

QUESTION M220.13

The information regarding the following Category I structures has not been provided in accordance with the Regulatory Guide 1.70, Rev. 3, "Standard Format ....."

1) Tunnel which connects the refueling water storage tank with the auxiliary building

2) Main steam and auxiliary feedwater tunnel

3) Electrical ducts

4) Category I manholes

Provide the information, on the above items, with regard to their location, basic dimensions, and reinforcing.

RESPONSE

The information requested is not in accordance with the preparation guideline of the replicate Marble Hill FSAR. The base plant FSAR was written to Regulatory Guide 1.70, Rev. 2, "Standard Format ....." In turn, replicate sections of the Marble Hill FSAR were reviewed and written to R.G. 1.70, Rev. 2.

Information on item "1)" can be found in replicate subsection 3.8.4.1.3 and Figure 1.2-11; "2)" refer to replicate subsection 3.8.4.1.4 and Figures 1.2-6 & 1.2-7 ; "3)" is replicated in subsection 3.8.4.1.5.; "4)" the Marble Hill response is found in Marble Hill Question QM 220.14.

From this information, the Byron SER (NUREG-0876) subsection 3.8.3 concluded that "..... the design of safety-related structures other than containment is acceptable....."

QUESTION M420.2

Discuss in detail the design of the bypassed and inoperable status indication using detailed schematics. Include the following information in the discussion:

1. Compliance with the recommendations of R. G. 1.47.

2. The design philosophy used in the selection of equipment/systems to be monitored including auxiliary and support systems.

3. How the design of the bypassed and inoperable status indication systems comply with positions B1 through B6 of ICSB Branch Technical Position 21, and

4. The list of system automatic and manual bypasses as it pertains to the recommendations of R. G. 1.47.

5. Discuss hardware features employed to provide a consolidated, human factored display of the bypassed and inoperable status of ESF equipment (R.G. 1.47).

RESPONSE

Compliance with Regulatory Guide 1.47 is provided in replicate section A1.47. Other replicate text referenced include subsections 7.1.2.10, 7.5, 8.1.9 and replicate Byron Question Q31.15. From this information the Byron SER(NUREG-0876) subsection 7.5.2.1 concluded the "..... bypass/inoperable system meets the requirement of Regulatory Guide 1.47. On this basis, the staff concludes the system is acceptable."

QUESTION M440.5

FSAR Page F15.2.8-2 indicates that the plant has not formally met acceptance criterion b on SRP page 15.2.8-4, which is that in

> "feedwater system pipe break events ..... the assumptions as to whether offsite power is lost and the time of loss should be made conservatively. Offsite power may be lost simultaneously with the occurrence of the pipe break, the loss may occur during the accident, or offsite power may not be lost. A study should be made to determine the most conservative assumption appropriate to the plant design being reviewed. The study should take account of the effects that loss of offsite power (LOOP) has on reactor coolant and main feedwater pump trips and on the initiation of auxiliary feedwater, and the resulting modification of the sequence of events."

Page F15.2.8-2 says "the FSAR reports only two cases, one for LOOP at the beginning of the transient and the other without LOOP (but) the two cases presented bound the analysis." How does the applicant support its statement that the two cases bound the analysis?

RESPONSE

Subsection 15.2.8 of the FSAR discusses the feedwater system pipe break accident. The analysis of the effects and consequences of this accident are presented in Subsection 15.2.8.2. The Byron SER (NUREG-0876) contains no open items, confirmatory issues or license conditions concerning the feedwater line rupture.

QUESTION M440.6

SRP Section 15.4.4 deals with startup of an inactive loop or recirculation loop at an incorrect temperature. FSAR page F15.4.4-1 states that

> "at the end of the transient, when the shutdown margin is low, credit is taken for operator action but not (for) the protection system ..... Having adequate operator action time to mitigate this transient obviates the necessity for automatic mitigation."

The distinction between the "start" and the "end" of the transient is not evident in FSAR section 15.4.4. Please provide a chronological description of the transient which shows when and how the protection system functions automatically and when and how credit is taken for operator action. Also, please state what operator actions are needed and whether (further) automatic mitigation would still take place after these operator actions. Additionally, please explain what is meant by the statement that "at the end of the transient ..... the shutdown margin is low."

RESPONSE

The discussion on the startup of an inactive loop is contained in Subsection 15.4.4. The Byron SER (NUREG-0876) contains no open items, confirmatory issues or license conditions on these issues.

QUESTION M440.7

SRP Section 15.4.6, "Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant (PWR)," is under review as a result of the V. C. Summer boron dilution event, as described in the enclosed "Clarification of Review Areas and Affected Branches, SRP 15.4.6." Some information that must now be reviewed is not included in FSAR Section 15.4.6. Please provide the missing information, including for example the proposed alarm setpoints, the technique for resetting an alarm setpoint, analyses assumptions regarding control rod position ..... (relative to the clarification), and alarm times.

Enclosure: Memo from B. W. Sheron et al, April 12, 1983: "Inadvertent Boron Dilution Events," with attached "clarification" cited in Q211.7.

RESPONSE

The discussion on the Chemical and Volume control system malfunction is contained in Subsection 15.4.6. The Byron SER (NUREG-0876) contains no open items, confirmatory issues or license conditions concerning boron dilution.

QUESTION M440.9

FSAR Section 15.6.3, SGTR, page 15.6-12, paragraph a, indicates that no steam and fission product activities are released from the defective steam generator after thirty minutes. However, FSAR page Q212.154-5, September 1982, sentence 1, indicates that a single failure could make an atmospheric dump valve (ADV) inoperable. For SGTR, please identify the impact of a stuck open ADV on the entire accident sequence and on its radiological consequences.

RESPONSE

The consequences of a steam generator tube rupture accident are presented in Subsection 15.6.3.3. The Byron SER (NUREG-0876) contains no open items, confirmatory issues or license conditions on this issue.

QUESTION M440.10

Q211.154 and Q212.47 concerned Branch Technical Position RSB 5-1, "Design requirements of the residual heat removal system." The Byron response in FSAR Amendment 39, September 1982, page Q212.154-7, says

> "For some unexplained reason after reaching zero load hot standby conditions, the normal and excess letdown lines are lost. The operator must then proceed to cold shutdown via either of two methods. In the first method, all RCS depressurizations are accomplished by spraying into the pressurizer from the CVCS auxiliary spray connection..... Initially, independent of the method use to reach cold shutdown, 4 wt.% boric acid storage tanks will be used to fill the pressurizer liquid level from 25% of span to 95% of span while the plant is held at zero load hot standby conditions. This will provide sufficient boron to compensate for xenon decay at hot standby."

Please show that the 4 wt.% boric acid in the pressurizer is transferred into the reactor core region in a timely manner so as to compensate for xenon decay at hot standby and so as to compensate for reactivity changes during subsequent cooldown. Also, please show that the entire transfer of boric acid, from the boric acid tanks into the reactor core, can be done using only safety-grade equipment and satisfying the criteria of RSB 5-1 (as included in Q212.47). If this cannot be done, please provide a table identifying the non-safety grade equipment which must be used and identifying all violations of the criteria of RSB 5-1 and then justify all entries. Additionally, please identify the conditions under which the boric acid spray could violate the temperature differential limit between the spray and the pressurizer or could clog the pressurizer.

RESPONSE

The Chemical and Volume Control System is discussed in Subsection 9.3.4. The Byron SER (NUREG-0876) contains no open items, confirmatory issues or license conditions on this system.