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AUG 24 1983

Docket Nos.: STN 50-546
and STN 50-547

Mr. S. W. Shields
Senior Vice President - Nuclear Division
Public Service Company of Indiana
P. O. Box 190
New Washington, Indiana 47162

Dear Mr. Shields:

Subject: Request for Additional Information for Marble Hill

Enclosed are requests for additional information which the staff requires to complete its evaluation of your application for an operating license for Marble Hill. These requests for additional information are the result of the staff's review of the information in your FSAR. You should amend your FSAR to include the information requested in Enclosure 1.

Review of certain portions of your application has been delayed and questions concerning these portions are not included in this package. You will be advised of the schedule for transmittal of any questions resulting from this remaining review. Due to the expected number and nature of these questions, the staff expects that you will be able to provide your responses to these questions when you respond to the questions in Enclosure 1; however, if necessary, you will be given ninety days from the date of transmittal to provide your responses.

As you were advised in the letter from Darrell G. Eisenhut to you dated February 23, 1983 concerning acceptance of your application for docketing, only a single set of questions concerning your FSAR are being transmitted to you, from each review branch, for response. After the staff has reviewed your response to these questions, a draft SER will be prepared to provide a basis for a series of meetings designed to close out open items.

You should provide your response to the enclosed request for additional information no later than December 2, 1983.

OFFICE							
SURNAME							
DATE							

For further information or clarification, please contact the Licensing Project Manager, Paul W. O'Connor (301)492-4708.

Sincerely,

Original Signed By:
L. M. Gibson
for
B. J. Youngblood, Chief
Licensing Branch No. 1
Division of Licensing

Enclosures:
1. Request for Additional Information
2. Review Branch Identification Numbers

cc w/encls.: See next page

OFFICE	DL:LB#1	DL:LB#1					
SURNAME	PO'Connor/lg	BJYoungblood					
DATE	08/24/83	08/24/83					

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ENCLOSURE 1

REQUEST FOR ADDITIONAL INFORMATION
MARBLE HILL NUCLEAR GENERATING STATION, UNITS 1 AND 2
PUBLIC SERVICE COMPANY OF INDIANA
DOCKET NOS. STN 50-546/547

220.1 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.

220.2 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.

220.3 (3.8.4) 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.

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

220.4 (3.7.1.2) From the table contained in Section 3.7.1.2 of the FSAR it is impossible to determine if the frequency intervals are in compliance with the staff's position, i.e., that reduction of these intervals does not result in more than ten (10) percent change in computed spectra. Demonstrate conformance with this position of the SRP and/or justify any deviation therefrom.

220.5 (3.7.1.4) Description of the supporting media does not address all of the Category I structures. Describe the foundation material for other Category I structures, such as the refueling water storage tank and the auxiliary feedwater tunnel.

220.6 (3.7.2.1) Indicate if an investigation has been made to ensure that inclusion of additional masses or 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.

220.7 (3.7.2) Provide a summary of natural frequencies, mode shapes, modal and total responses for representative major Category I structures, in addition to those in Tables 3.7-4 and 3.7-5 such as the essential service water cooling towers and basins.

220.8 (3.7.2.3) 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.

220.9 (3.7.2.14) (B/B) 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.

220.10 (3.7.2.10) 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.

220.11 (3.7.1.3) (3.8.5) Section 3.7.2.4 of the FSAR states that the auxiliary-fuel handling building complex is partly on soil and partly on bedrock, while Table 3.7.3 indicates that these structures are entirely on the bedrock. Clarify this apparent discrepancy.

220.12 (3.7.4) Review of Section 3.7.4 of the FSAR reveals that there are several deviations from the corresponding requirements of SRP Section 3.7.4. Specifically, the seismic instrumentation is lacking the following items:

	Item	Location
1)	Triaxial Response Spectrum Recorder	Reactor Equipment Support or Reactor Support
2)	Triaxial Peak Accelerograph	Reactor Equipment or Reactor Piping
3)	Triaxial Response Spectrum Recorder	Category I Equipment Support or Category I Piping Support Outside of Containment

220.13 (3.8.4.1) 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.

220.14 (3.8.4.1) Describe the methods used as a protection of Category I manholes against tornado generated missiles.

230.1
SRP 2.5.2.1

Reference FSAR section 2.5, Table 2.5-4: Recent studies of the historic seismicity of the Eastern U. S. have resulted in significant changes in the locations and intensities of earthquakes germane to the Marble Hill site. Update FSAR table 2.5-4 to reflect these changes (cf Barstow et al, 1981). Include events of significance to the site which have been recorded by the USGS sponsored regional network operated by the St. Louis University,1980-1982, 1982.

230.2
SRP 2.5.2.2

Recent studies dealing with the concept of seismic source zones have resulted in zonations different from those proposed at the time of the construction license review (SER 1976). For instance Barstow et al, 1981 place the Marble Hill site in the Cincinnati Arch zone. Discuss the validity of these zones and the appropriateness of the SSE assumed for the site with respect to earthquake zonation. In recent safety reviews the NRC has often relied upon site specific spectra to evaluate the seismic design criteria. The reason being that site specific spectra are more in accord with the controlling earthquake size, frequency spectrum and local site conditions. For example, site specific (rock) spectra for Wolf Creek (NUREG-0881) and Perry (NUREG-0887) were developed from earthquakes with magnitudes $m_b = 5.3 \pm 0.5$, which correspond with estimates of magnitude for the 1937, Anna, Ohio earthquake. Include in the above discussion on earthquake zonation an estimate of the appropriate site specific spectrum. Compare this spectrum to the design spectrum and discuss exceedences, if any.

230.3
SRP 2.5.2.2,3,4

Reference FSAR section 2.5.2.3.9:
Studies by Braile et al, 1979, 1982a, 1982b, 1983 and Buschbach 1978, 1981, suggest a possible extension of the New Madrid faulted zone into the Wabash Valley. Evaluate these studies and discuss their impact upon the estimation of ground motion at the site. Include in your discussion the relative influence of more recent attenuation formulae and other methods of estimating ground motion from distant earthquakes than those presented in the FSAR (cf Nuttli 1981a, 1981b).

231.1
SRP 2.5.1.1

The Annual Progress Report for Fiscal Year 1981 for the NRC research project, "Geologic Studies of Faults and Terraces of Central and Western Kentucky", Contract No. NRC-G-04-81-018 states that there is stratigraphic evidence suggesting that a fault in Clark County, Kentucky, approximately 80 miles southeast of the Marble Hill site, has apparently offset approximately 10 feet of Plio-Pleistocene age Kentucky River terraces and perhaps recent alluvium. The fault is the eastern boundary of a small, northwest trending graben within the Kentucky River Fault zone, which is the northern border of the Rome Trough. Trenching of this fault is to take place in August 1983 to identify apparent tectonically controlled offset in the Plio-Pleistocene terrace deposits.

a) We request that you be informed of the project study and its objectives, and follow the trenching results,
b) evaluate the results of the study and trenching results, and
c) discuss the significance of these developments to the safety of the Marble Hill Plant operation.

231.2
SRP 2.5.1.1
FSAR 2.5.1.1.5.1

Recent studies by Braile et al (1982a, 1982b, and 1983) suggest that structures similar to the Reelfoot rift in the New Madrid area can be traced northeastward into southern Indiana and to within approximately 40 miles of the Marble Hill site. Also, Soderberg and Keller (1981) suggest that the Rough Creek graben is a buried ancient rift. Braile et al make reference to an alignment of the Rome Trough and the Rough Creek graben along the 38th parallel lineament. They state that the structures associated with these rifts could be zones of weakness and describe the regional stress field of the midcontinent area as a nearly horizontal compressive stress oriented approximately N60°E. They conclude that the presence of these ancient rifts in this stress field give rise to contemporary intraplate seismicity. In view of the proximity of the Marble Hill Station to the proposed southern Indiana rift, the Rome Trough, and the Rough Creek graben, discuss the significance to safety, if any, of the Braile et al studies. Also, discuss the potential for seismicity of the type and magnitude that is occurring in the New Madrid area and determine the safety consequences to the plant structures.

231.3
SRP 2.3.3
FSAR 2.5.1.1.5.1.1

The Area D group of faults in northern Kentucky trends northwesterly toward the site. Wilson (1974) states in Appendix 2F of the FSAR "I feel that this zone is tectonically significant and unique, not necessarily related to West Hickman-Bryan Station faults in time or geography. I feel that they are pre-Ohio Shale in age."

a) Discuss why Wilson considered Area D to be tectonically significant and unique and how this characterization may impact the site.

b) Provide a list of and perform a study of the most recent (post 1974) publications, including the Landsat Imagery coverage, and discuss any new information which will update our knowledge of the extent of the Area D faulting to the northwest, the age of faulting, the relationship of the faulting to the West Hickman-Bryan Station faulting, and whether there is any seismicity associated with any of the Area D faults.

231.4
SRP 2.5.1.2
FSAR 2.5.1.2.1.1

Appendix 2H of the FSAR includes a report of the results of a geologic mapping program started on October 24, 1977 and still in progress at the time of the FSAR preparation. Only representative photo mosaics, schematic sections, a floor map of the fuel handling building and refueling water storage tanks excavation, and columnar geologic sections were presented in this report.

a) Provide an integrated geologic map of the entire excavation shown in Figure 1 Location Map of Appendix 2H, showing details of the entire floor area and all walls.
b) Revise an appropriate figure in the FSAR to show the locations of all external pipelines, duct runs, and manholes. In addition show longitudinal sections of the bedrock geology along the different pipelines and duct runs.

231.5
SRP 2.5.1.1
FSAR 2.5.1.1.5.1

You state in the 3rd paragraph in the section on Regional Tectonic Features, "Examination of Figure 2.5-12 indicates that the distribution and orientation of faults are consistent with the theory of bending." You then give examples of faulting in the Jessamine dome, the Fairfield Basin, and the Illinois Basin. The data presented in Figure 2.5-12 does not illustrate the examples given above. Figure 2.5-11 more appropriately serves this function, so that the figure cited should be changed to the appropriate number.

REFERENCES TO Q 231 SERIES

1. Barstow, N. L. et al "An Approach to Seismic Zonation for siting Nuclear Electric Power Generating Facilities in the Eastern United States." NUREG/CR-1577, May 1981.

2. Braile, L. W. et al "An integrated geophysical and geological study of the Tectonic Framework of the 38th Parallel LIneament in the vicinity of its intersection with the extension of the New Madrid Fault Zone" NUREG/CR-1014, September 1979.

3. Braile, L. W. et al "An ancient rift complex and its relation to contemporary seismicity in the New Madrid Seismic Zone." Tectonics Vol. 1, No. 2, April 1982a.

4. Braile, L. W. et al "A tectonic study of the extension of the New Madrid Fault Zone near its intersection with the 38th parallel lineament." NUREG/CR 2741 June 1982b.

5. Braile, L. W. et al, "Technical Report on Seismic Profiling in the Wabash River Valley," USNRC-Technical Progress Report Contract No. NRC-04-80-224, June 1983.

6. Buschbach, T. C. "New Madrid Seismotectonic Study activities during fiscal year 1978." NUREG/CR-0450.

7. Buschbach, T. C. "New Madrid Seismotectonic Study" NUREG/CR-2129 September 1981.

8. Nuttli, O. W. et al "Consequences of Earthquakes in the Mississippi Valley" ASCE Pre-print 81-51. October 1981a.

9. Nuttli, O. W. "Similarities and Differences between western and eastern United States Earthquakes, and their consequences for earthquake engineering." Earthquakes and Earthquake Engineering - Eastern U. S. Ann Arbor Press, 1981b. Ed. J. E. Beavers.

10. Saint Louis University, Department of Earth and Atmospheric Sciences, "Central Mississippi Valley Earthquake Bulletins, Quarterly reports 1980-1982.

11. Saint Louis University, "A Seismological study of the Northern Extent of the New Madrid Seismic Zone, final report, June 1982. (Robert B. Hermann, principal investigator).

12. Soderberg, R. L. and Keller, G.R., "Geophysical Evidence for Deep Basin in Western Kentucky," AAPG Bulletin Vol. 65, No. 2 pp. 226-234, 1981.

13. Wilson, E. N. Written Communication, Kentucky Geological Survey, Lexington, Kentucky, 1974.

241.1 (SRP 2.5.4.3) In addition to information requested in Questions 220.5 and 220.13, provide detailed drawings (plans and cross-sections) of the foundations for the Seismic Category 1 Refueling Water Storage tank, the Auxiliary Feedwater Tunnel, the Electrical Duct Banks and the Manholes. The drawings should include the location and dimensions of the structures, the subsurface soil and rock profile and the areas of rock foundation treatment. The geologic mapping of the excavations for these structures should also be provided.

241.2 (SRP 2.5.4.10) Discuss the soil and/or rock properties used in the analysis (static and dynamic) for the structures in question 241.1. Indicate the generalized soil and/or rock profile analyzed, the gross and allowable bearing pressures (static and dynamic) and the bearing capacity factors of safety.

241.3 (SRP 2.5.4.2) Describe the laboratory testing procedures for the rock supporting the foundations of the seismic Category 1 structures. Identify standard procedures and describe and justify deviations from these standards. Provide a summary table indicating the values of the parameters (soil and rock) used in the design of the structures.

241.4 (SRP 2.5.4.3) Provide the geologic mapping of the excavations for the Category I Essential Service Water Pipelines. The mapping should indicate the location and extent of areas of rock foundation treatment.

241.5 (SRP 2.5.4.10) The Marble Hill FSAR states that the Category 1 subsurface walls have been designed to resist lateral pressures due to a granular backfill and hydrostatic pressures. Illustrate your methodology by providing lateral pressure diagrams (used in design) for a sample wall with the following loadings:

(1) Static and dynamic (earthquake) earth pressures.

(2) Static and dynamic (earthquake) water pressures.

(3) Static and dynamic surcharge loading.

Indicate if precautions were taken during construction to prevent additional lateral loadings due to effects of compaction equipment.

241.6 (SRP 2.5.4.12) Provide plans, summaries of specifications and methods of quality control for rock anchors installed in the foundation slab of the Essential Service Water Cooling (ESW) towers. What were the hydrostatic uplift pressures considered in the design of the ESW structure and what were the assumed water levels in the ESW basin?

241.7 (SRP 2.5.4.5) The properties of the Category 1 backfill materials (CCF1 and RCF2) are not presented in the FSAR. The quantities and the source of backfill materials is also not discussed. Please provide this information.

241.8 (SRP 2.5.4.5) Update Appendix 2K for field and laboratory compaction testing of the backfill materials (CCF1 and RCF2).

241.9 (SRP 2.5.4.5) Provide cross sections showing the "bash" fill relative to the rock and the Seismic Category 1 structural walls. Specifications for the bash mix used and a summary of the test results should also be provided.

241.10 (SRP 2.5.4.2) A permeability of 745 ft/year was measured (by the water pressure test) in Boring R2-4 in the interval between elevation 725 to 735 feet MSL. With respect to jointing in foundation rock, discuss the significance of this high permeability and why it "should be considered as an isolated occurrence". Was the water pressure test repeated to verify the permeability value or was there any equipment malfunction? Did rock cores extracted from this borehole reveal the existence of fractures not encountered and logged during drilling?

241.11 (SRP 2.5.4.10) For each Seismic Category 1 structure, provide the following information in tabular form:

- Total maximum bearing pressures imposed under static and under dynamic loading conditions

- Description of the foundation bearing materials (e.g., rock classification), and the allowable bearing pressures under static and under dynamic loading conditions, and the bearing capacity factors of safety.

- Estimated total and differential settlement.

- Measured total and differential settlement.

250.0 Quality Assurance Branch

260.1 Describe in more detail the criteria for determining the size of the QA organization including the inspection staff, and present the projected number of professional QA/QC personnel to be on board during the operation phase recognizing this number will vary somewhat throughout the operations phase. Identify in more detail the QA and QC organizational positions on Figure 17.A2.2 and in particular identify those positions that have quality assurance expertise in the technical areas of the nuclear facility such as mechanical, electrical (I&C), structural, and nuclear. (1A5)*

260.2 Provide a more definitive statement of the company's corporate QA policy. (2A2)

260.3 Provide a listing of QA procedures that describe how each of the 18 criteria of 10 CFR Part 50, Appendix B and the documents identified in Table 17.2-1 are to be carried out. Identify each of the criteria of 10 CFR Part 50, Appendix B and the documents in Table 17.2-1 adjacent to the procedures where these items are discussed. (2B4)

260.4 Describe or reference in clearer detail the QA program that will apply to the fire protection program.

260.5 Describe those provisions which assure that the indoctrination and training program includes the following: (2D)

a. Proficiency tests are given to those personnel performing and verifying activities affecting quality, and acceptance criteria are developed to determine if individuals are properly trained and qualified.

260.6 Describe those provisions which assure that verified computer codes are certified for use and that their use is specified. (3E4)

260.7 Describe those provisions which assure that commercial "off-the-shelf" items are acceptable. (7B4)

*This designation represents the particular items of Standard Review Plan Section 17.1 that the requests originate from.

280.1 Verify that all fire barriers have been tested and approved by an independent laboratory, in conformance with BTP CMEB 9.5-1, Section C.5.a.

280.2 Substantiate the fire resistance capability of the fire barriers used to separate safety-related areas or high hazard areas. Verify that their construction is in accordance with a particular design that has been fire tested as required by BTP CMEB 9.5-1 Section C.5.a(3), (4) and (5). Describe the design, the test method used and the acceptance criteria. Provide information for the following components (a) fire barriers, (b) fire dampers, (c) fire doors, and fire barrier penetration seals around ducts, pipes, cables, cable trays and in other openings (e.g., concrete joint seals and fillers), and special treatment for other openings. Include information substantiating the use of foil-backed Thermofiber insulation in a 3-hour rated barrier. Describe the support structure for barriers that are not floors or ceilings; describe how fire dampers that are installed in the ventilation ducts that penetrate rated fire barriers of safety-related areas; verify that all seals are of the thickness specified in the fire tests; and verify that cables and cable trays are supported in a manner similar to supporting arrangements used in the fire tests.

280.3 Describe how fire protection has been provided for structures, systems and components important to safe shutdown so that one train of systems necessary to achieve and maintain hot shutdown conditions from either the control room or emergency control station(s) is free of fire damage and that systems necessary to

achieve and maintain cold shutdown from either the control room or the emergency control station(s) can be repaired within 72 hours to comply with BTP CMEB 9.5-1 Section C.5.b(1).

Provide an analysis which shows that one redundant train of equipment structures, systems, and cables necessary for safe shutdown (for example, the remote shutdown panels in Fire Zone 11.4C-0 and the motor control centers in Fire Zone 11.4C-0), can be maintained free of fire damage by one of the following methods:

(a) Separation of cables and equipment and associated circuits of redundant trains by a fire barrier having a 3-hour rating. Structural steel forming a part of or supporting such a fire barriers should be protected to provide fire resistance equivalent to that required of the barrier;

(b) Separation of cables and equipment and associated circuits of redundant trains by a horizontal distance of more than 20 feet with no intervening combustible fire hazards. In addition, fire detectors and automatic fire suppression systems should be installed in the fire area; or

(c) Enclosure of cable and equipment and associated circuits of one redundant train in a fire barrier having a 1-hour rating. In addition, fire detectors and an automatic fire suppression system should be installed in the fire area.

280.4 Describe how the arrangement of cables and equipment in the safety valve enclosures in Fire Zone 18.3-1 complies with the provisions of BTP CMEB 9.5-1 Section C.5.b.

280.5 Describe how hydroger and other flammable gas lines which are routed through safety-related areas comply with BTP CMEB 9.5-1 Section C.5.d(5).

280.6 It is our position that Class A fire detection systems be provided to comply with BTP CMEB 9.5-1 Seciton C.6.a(2) to protect all areas of the plant which contain or present an exposure fire hazard to safety-related equipment and cables. Verify that you will comply with this position.

280.7 Describe how the primary and secondary power supplies provided for the fire detection system and for electrically operated control valves for automatic suppression systems comply with the provisions of BTP CMEB 9.5-1 Section C.6.a(6).

280.8 Verify that the fire pumps and their controllers are UL listed and installed in accordance with NFPA 20 requirements to comply with BTP CMEB 9.5-1 Section C.6.b(6).

280.9 Verify that all parts of every area which contains, or which could present a fire exposure hazard to, safety-related equipment, are within 30 feet (effective reach) of a nozzle attached to not more than 100 feet of hose from an interior hose station to comply with BTP CMEB 9.5-Section C.6.c(4). Particular attention should be given to congested plant areas where it may be difficult to deploy hose lines.

280.10 It is our position that oil collection systems should be installed on reactor coolant pumps to meet BTP CMEB 9.5-1 Section C.7.a(1)(e). Verify that you will comply with this position.

280.11 Describe the fire protection features provided in the peripheral rooms of the control room complex in lieu of automatic water suppression systems required by BTP CMEB 9.5-1 Section C.7.b.

410.4 (9.1.4) You have not indicated that the design capability for assuring safe light load handling is replicated for Marble Hill Station. You stated in FSAR Appendix F that your analysis would be complete by the third quarter of 1983. Provide this analysis and verify that the design replicates Byron/Braidwood or discuss and justify any deviation that could lead to light load drops with consequences in excess of the design basis fuel handling accident.

410.5 (9.1.5) You indicated in FSAR Appendix F that you have not yet responded to the generic letter regarding the criteria contained in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." Provide this analysis as discussed below:

Provide an analysis of the effects of dropping heavy loads other than the spent fuel cask or verify that load drops need not be postulated since the crane design meets the criteria of NUREG-0554, "Single Failure Proof Cranes." The analysis should satisfy the evaluation criteria of NUREG-0612, Section 5.1, and consider the consequences of dropping the reactor vessel head and vessel internals during preparation for or completion of fuel handling. In addition, the lower load block of both the containment building polar crane and the fuel building crane should be considered as heavy loads and an analysis of the consequences of their falling included in this evaluation. Alternatively, a comparison of heavy load handling equipment against the guidelines of NUREG-0554 should be provided. This comparison should include a point-by-point evaluation against NUREG-0554 criteria, and a failure mode and effects analysis of the crane electrical system which confirms that loss of one phase of a three phase power supply or phase reversal will not result in dropping of a load or other unacceptable load handling consequences. Also, verify that movement of spent fuel between units 1 and 2 will not follow a path that may endanger safety-related equipment such as essential buried piping or electrical cabling.

410.6 (9.2.1) (9.2.5) Verify that failure of buried nonseismic Category I pipe in a safe shutdown earthquake will not result in failure of safety-related buried piping (such as service water piping) by soil erosion. Relate your response to that provided in response to Byron/Braidwood Question 010.8.

410.7 (9.2.1) (9.2.5) Your response to question 291.3 regarding a comprehensive plan to eliminate Asiatic clams from the safety-related cooling water system is not complete. As discussed during a site visit on July 13, 1983, provide a description of the corbicula biofouling treatment program which is being formulated for makeup water from the Ohio River to the ultimate heat sink. Describe how this program will provide protection of safety-related cooling water systems from degradation due to biofouling concerns.

410.8 (9.2.5) As stated in Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants," describe the means provided for ensuring the continued operability and safety function of the ultimate heat sink beyond the 30 day design capability following a seismic event.

410.9 (9.2.5) Provide missing figure 9.2-4a (schematic illustration of the ultimate heat sink).

410.10 (9.4) Provide a description of any measures utilized for detecting and correcting dust accumulation on safety-related equipment or describe why such provisions are unnecessary in order to assure the availability of safe shutdown functions on demand.

410.11 (10.4.5) Describe the differences in the circulating water system failure flooding analysis between Marble Hill and Byron/Braidwood Stations. Verify that the resulting flooding does not result in loss of safety-related equipment or affect safe shutdown of the plant assuming a concurrent single failure in essential equipment. Relate your response to that provided in response to Byron/Braidwood Question 010.50.

420.1
(SRP 7.2)

The description in Section 7.2.1.1.2.d.4 for the reactor coolant pump breaker trip is intended to trip the reactor if any reactor coolant pump breaker opens above the P-8 power level. This reactor trip is intended to occur before pump coast-down would initiate a low flow induced reactor trip. Figure 7.2-1 sheet 5 illustrates the final system logic for the various reactor coolant pump trips. It does not show any trip logic which functions as described in Section 7.2.1.1.2.d.4. This discrepancy should be resolved. Does the Reactor Trip System include a reactor coolant pump breaker trip? (Note: This discrepancy was found in the yellow pages of the FSAR.)

420.2
(SRP 7.5)

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).

420.3
(SRP 7.1)

Some of the primary methods the Staff uses to convey information to licensees and applicants based on operating experience are Office of Inspection and Enforcement (IE) Bulletins, Circulars and Information Notices. Although only the IE Bulletins require written responses, the staff expects licensees and applicants to take appropriate action(s) on the information provided in the Circulars and Information Notices applicable to their design. Included in Attachment 1 is a list of IE Bulletins, Circulars and Information Notices that are applicable to PWRs. Provide a discussion which includes the following:

(a) Procedures for determining the applicability of the IEB, IEC and IEIN to your facility.

(b) Procedures or methods for factoring the applicable information or criteria into the Marble Hill design.

(c) Examples of specific design modifications and their implementation resulting from items (a) and (b).

420.4 (SRP 7.2 & 7.3) Discussion of RPS and ESFAS setpoint methodology per R.G. 1.105 should provide and describe the following information for NSSS and BOP safety related setpoints:

(a) Provide a breakdown of the channel uncertainties for each protective channel including drifts, calibration errors and environmental allowances.

(b) Provide the appropriate trip setpoint values, allowable values, safety analysis limits, total channel uncertainties and margins between the trip setpoints and the safety analysis limits. (For those cases where the answer is given as a percentage of span or range, give the span or range as appropriate).

(c) Identify protection channels where the Technical Specification setpoint, less the allowance for total channel error, falls within 10% of the instrument range limit or within 10% of the range between level measurement taps. For those cases specify the remaining margin to the end of the range.

(d) Identify any time limits on environmental qualification of instruments used for trip, post-accident monitoring or engineered safety features actuation. Where instruments are qualified for only a limited time justify the reasons why that time is sufficient.

(e) List those accidents that can cause harsh environments and document the environmental error allowance that is used for each reactor trip and engineered safeguards setpoint.

(f) Provide representative examples of the computations and assumptions used for the derivation of the trip setpoint and allowable value.

(g) Provide a list of the primary and secondary protection channels for each accident.

IE Bulletin No. 78-06 May 31, 1978

DEFECTIVE CUTLER-HAMMER, TYPE M RELAY WITH DC COILS

IE Bulletin No. 79-05 April 1, 1979

NUCLEAR INCIDENT @ TMI

IE Bulletin No. 79-05A April 5, 1979

IBID

IE Bulletin No. 79-06 April 11, 1979

REVIEW OF OPERATIONAL ERRORS AND SYSTEM MISALIGNMENTS IDENTIFIED DURING THE TMI INCIDENT

IE Bulletin No. 79-06A April 14, 1979

IBID

IE Bulletin No. 79-06B April 14, 1979

IBID

IE Bulletin No. 79-06A Rev. #1 April 18, 1979

SEE ABOVE (06A)

IE Bulletin No. 79-05B April 21, 1979

See 79-05A ABOVE

IE Bulletin Nos. 79-05C & 79-06C July 26, 1979

NUCLEAR INCIDENT @ TMI -SUPPLEMENT

IE Bulletin No. 79-21 August 13, 1979

TEMPERATURE EFFECTS ON LEVEL MEASUREMENT

IE Bulletin NO. 79-24 September 27, 1979

FROZEN LINES

IE Bulletin No. 79-25 November 2, 1979

FAILURE OF WESTINGHOUSE BFD RELAYS IN SAFETY-RELATED SYSTEMS

IE Bulletin No. 80-12 May 9, 1980

DECAY HEAT REMOVAL SYSTEM OPERABILITY

IE Bulletin No. 80-19 July 31, 1980

FAILURE OF MATRIX RELAY IN PROTECTIVE SYSTEMS

IE Bulletin No. 80-19 Rev. 1 August 13, 1980

FAILURE OF MATRIX RELAY IN PROTECTIVE SYSTEMS

IE Bulletin No. 80-20 July 31, 1980

FAILURE OF WESTINGHOUSE TYPE W-2 SPRING RETRUN TO NEUTRAL CONTROL SWITCH

IE Bulletin No. 80-24 November 21, 1980

PREVENTION OF DAMAGE DUE TO WATER LEAKAGE INSIDE CONTAINMENT

IE Notice No. 79-04 February 16, 1979

DEGRADATION OF ESF

IE Notice No. 79-25 October 1, 1979

REACTOR TRIPS @ TURKERY POINT UNITS 3&4

IE Notice No. 79-29 November 16, 1979

LOSS OF NONSAFETY-RELATED REACTOR COOLANT SYSTEM INSTRUMENTATION DURING OPERATION

IE Notice No. 80-10 March 7, 1980

PARTIAL LOSS OF NON-NUCLEAR INSTRUMENT SYSTEM POWER SYPPLY DURRING OPERATION

IE Notice No. 80-12 March 31, 1980

INSTRUMENT FAILURE CAUSES OPENING OF PORV AND BLOCK VALVE

IE Notice No. 80-20 May 8, 1980

LOSS OF DECAY HEAT REMOVAL CAPABILITY AT DAVIS-BESSE UNIT 1 WHILE IN REFUELING MODE

IE Notice No. 80-44 December 16,1980

ACTUATION OF ECCS IN THE RECIRCULATION MODE WHILE IN HOT SHUTDOWN

IE Notice 81-15 April 22, 1981

DEGRADATION OF AUTOMATIC ECCS ACTUATION CAPABILITY BY ISOLATION OF INSTRUMENT LINES

IE Notice 82-01 January 22, 1982

AUX FEEDWATER PUMP LOCKOUT RESULTING FROM WESTINGHOUSE W-2 SWITCH CIRCUIT MODIFICATIONS

IE Notice 82-02 January 27, 1982

WESTINGHOUSE NTSFD RELAY FAILURE FOR REACTOR PROTECTION SYSTEMS @ NUCLEAR POWER PLANTS

IE Circular No. 78-19 December 29, 1978

MANUAL OVERRIDE OF SAFETY SYSTEM ACUTATION SIGNALS

IE Circular No. 79-22 November 16, 1979

STROKE TIMES FOR POWER OPERATED RELIEF VALVES

IE Circular No. 81-11 July 24, 1981

INADEQUATE DECAY HEAT REMOVAL DURING SHUTDOWN

440.1

The Byron response to RSB Question Q212.37 was modified by Marble Hill (MH). The modification indicates that (1) changes were made in the rubbing seal materials in the number 2 seal and number 3 seal of each reactor coolant pump (RCP), and (2) "the number 1 seal is [now] designed to function satisfactorily... with a maximum ΔP of 2,300 psig" (versus 2,470 for Byron). Please (1) tell why the changes were made in the seal materials and describe the actual testing program used for these new seals materials relative to the testing program described in Byron FSAR page Q212.37-2 (first full paragraph); and (2) explain what caused the "design" ΔP to drop from 2470 to 2300 psig, describe the effects of exceeding the design ΔP, and identify the impacts the drop in design ΔP has on the transient analyses presented in Chapter 15.

440.2

FSAR Table 5.4-1 shows that the reactor coolant pump (RCP) for Marble Hill has (1) a height of 25.89 ft (1.04 ft less than for Byron); (2) a dry weight of 171,272 lb (23,928 lb less than for Byron); and (3) decreases in normal input current relative to Byron; but (4) no changes in such other RCP parameters as the moments of inertia for flywheel, motor, shaft, and impeller. The staff notes that some changes in the RCP structure must exist because of changes made by MH on FSAR page 5.4-1 (in the last sentence) and on pages Q212.37-1, 2 (per Q211.1, directly above). Please describe all changes in the RCP and their combined impact on performance and on design margin. That description should include a pictorial identification of the RCP components (as in the version of Figure 5.4-1 used in the Byron FSAR).

440.3

FSAR Section 5.4.7, "Residual Heat Removal System," page 5.4-30, line 10, indicates the discharge of suction side relief valves is collected in the recycle

holdup tank for MH whereas it is collected in the pressurizer relief tank for Byron. This change has been properly reflected in changes in FSAR Table 5.4-13, "Relief Valve Discharge to the Pressurizer Relief Tank." This change has however not been reflected in changes in FSAR Figure 5.4-4, "Residual Heat Removal System," or in FSAR Figure 6.3, "Diagram of Safety Injection," which are identical for MH and Byron.

1. Please indicate which Figures should be corrected, and whether the corrections are needed for MH or for Byron or for both.

2. Additionally, please explain the reason for the change in the collection point (made on page 5.4-30 line 10).

440.4

FSAR page F15.1.5, top, should be revised to show a commitment that II.K.3.5, "Automatic Trip of Reactor Coolant Pumps," will be resolved consistent with the guidance of Generic Letter No. 83-10c.

440.5

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?

440.6

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."

440.7

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.

440.8

FSAR Section 15.6.3, "Steam Generator Tube Rupture" (SGTR), page 15.6-12, line 3, indicates that Marble Hill uses a vacuum pump rather than the air ejector used by Byron. Please (1) describe the new equipment, (2) explain the reason for the equipment change, (3) identify the impact of the change on the availability of the condenser during an SGTR incident, and (4) identify the impact of the change on the reliability of the SGTR warning signal from the condenser radiation alarm (which is referred to on page 15.6-8, middle). Aditionally, (5) please provide the SGTR sequence of events data that should have been in Table 15.6-2 (per page Q212.9-1, last paragraph) and the SGTR radiological consequences data that should have been Tables 15.6-7, 8 (per page 15.6-41).

440.9

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.

440.10

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 mut 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 used to reach cold shutdown, 4 wt.% boric acid from the 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 presusrizer.

Clarification of Review Areas
and Affected Branches
SRP 15.4.6

Title: Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant (PWR)

I. Review Branches and Areas of Responsibility

The current version of SRP 15.4.6 states that RSB is the lead branch for the review of this SRP, states which branches may also take part in the review, if requested by RSB, and gives other guidance for other technical branches. However, as a result of recent CRGR review, it was concluded that the guidance in the SRP should be clarified. This clarification is given below. Note that the review responsibilities in the current SRP version remain the same.

Reactor Systems Branch - Lead review branch for the SRP, and coordinates technical review. In addition to the guidance contained in the current version of the SRP, RSB, with the assistance of CPB, should review the proposed alarm setpoints, and the technique for resetting the alarm setpoint due to changing reactor conditions (i.e., neutron source decay, moderator temperature change, or control rod position changes). Also, RSB and CPB will review the analyses assumptions regarding control rod position, shutdown reactivity, boron concentration (initial and critical) and the times between event initiation, alarm and criticality. CPB has the lead in the "time to criticality" analyses.

Instrumentation and Control Systems Branch

In addition to the guidance contained in the current SRP version, ICSB will review the instruments and alarms relied on to alert the operator of a boron dilution event in progress.

Core Performance Branch

In addition to the guidance contained in the current SRP version, CPB will evaluate the applicant's assumed rod configuration and verify its adequacy with respect to the requirements described in (II) below. CPB will also review the applicant's assumed initial and final (critical) boron concentrations, the initial shutdown margin and the reactivity coefficients assumed in the transient analyses. Together with RSB, CPB will review the applicant's proposed "high flux at shutdown" alarm setpoint, if assumed in the analyses, and the proposed method of ensuring adequate alarm setpoints as the reactor conditions change (i.e., neutron source decay, moderator temperature change, or control rod position changes). CPB and RSB will together review the time to criticality analyses and ensure consistent assumptions. CPB has the lead in this review.

II. SRP 15.4.6 Clarification

1. Control rod position - The SRP 15.4.6 specifies the required control rod position to be assumed in the criticality analyses for only one mode of plant condition - refueling. The SRP is silent on the other modes. This has resulted in confusion and, in some cases, inconsistent safety reviews. The following table specifies the rod positions applicants should assume until the SRP is formally revised. Note that applicants may opt to assume a more limiting initial control rod position as long as the assumed configuration is possible with respect to actual plant operations. Applicants who propose less restrictive rod positions (i.e., control rod configurations that result in a longer time to criticality) should justify their assumptions.

Mode	Rod Position
1. Power Operation	ARI-RRO*
2. Startup	ARI-RRO*
3. Hot Standby	ARI-RRO
4. Hot Shutdown	ARI-RRO
5. Cold Shutdown	ARI-RRO
6. Refueling	ARO

ARO = All Rods Out

ARI = All Rods In

ARI-RRO = All Rods In, Most Reactive rod stuck out

* The rod configuration should be assumed after the reactor trips as a results of the dilution.

2. Boron Dilution Alarms - The plant must have adequate alarms to alert the operator to the dilution event. The times given in the SRP are to be measured from the time of the alarm to the time of criticality, not from the time the dilution event starts until criticality. Two alarms should be provided (e.g. two source range detector bistables, these may be powered from different electrical sources, generally class IE- battery backed inverter power sources). The plant annunciator system is adequate for the alarms, i.e., it dosen't have to be a Class IE system. Seismic is not a design consideration nor is environmental qualification.

The alarm setpoint should be consistent with the analyses assumption and the alarm setting should be adjusted to reflect changing reactor conditions. For example, if the setpoint is twice the background count level, then as the background counts decay after shutdown, the setpoint must be lowered. This can be done either manually or automatically. If manually, applicants should commit to a procedure of resetting the alarm setpoint, based on source decay rate. Also, alarm setpoint changes should be made if the RCS temperature is changed, or if control rods are manipulated since this may also affect the neutron flux at the detectors. The alarm setpoint and the frequency to reset these alarms should, as a minimum, be included in the plant procedures.

450.1 (SRP Section 6.4)

In the evaluation of control room habitability, provide the following information:

a) A statement that no potentially hazardous toxic gas sources are stored onsite. If toxic gas sources are present, provide an accidental release analysis in accordance with Standard Review Plan Section 6.4 and Regulatory Guide 1.78 and 1.95.

b) State the type of self-contained breathing apparatus stored in the control room emergency envelope and the location and description of the nearby unlimited offsite bottled air replenishment capability.

c) State the chlorine detection time, isolation damper closure time and the travel time of chloring gas from the chlorine detector to the isolation damper.

d) Provide an estimate for the infiltration rate at the dampers or valves upstream of the control room recirculation fans during a control room isolation following a toxic gas release.

450.2 (SRP Section 6.4)

For design basis accidents identify the locations of major radiation release points relative to the control room outside air intakes. Also provide the locations for toxic gas releases relative to control room air intakes.

450.3 (SRP Section 6.4)

Provide additional information relating to the selection of the least contaminated outside air intake following a radiation release accident, including:

a) whether the selection of the proper intake for control room pressurization is manual or automatic;

b) whether the operation of the two outside makeup air filtration trains following a radiation release is in parallel or sequential;

c) whether the makeup air inlets meet applicable single failure criteria; and

d) a description of the available shielding between the control room operators and the emergency filtration units.

450.4 (SRP Section 15.6.5)

In response to Byron/Braidwood FSAR question 311.31, the applicant has computed the potential offsite radiological consequences at the Marble Hill site following a LOCA using a conservative iodine spray removal coefficient. The Marble Hill response states the dose at the Exclusion Area Boundary to be 72 Rem to the thyroid. The corresponding maximum valve for the Byron/Braidwood sites is 145 Rem to the thyroid. Because the Marble Hill FSAR is identical to the Byron/Braidwood FSAR for this accident and because the short-term atmospheric factors for the Marble Hill and Byron/Braidwood sites are quite similar, the staff does not understand why the calculations would produce results so dissimilar .

Provide either an explanation which resolves this apparent inconsistency, or all the assumptions and bases used in the cited Marble Hill analysis which are different from those used for the Byron/Braidwood response.

451.2
(SRP 2.3.3)

Describe the onsite meteorological data available in the control room, including the measurement levels and the parameters being displayed, type of display, and the type of data storage being used.

470.7 Provide a listing of the input parameters that were used in Sargent & Lundy's radiological assessment computer runs to estimate the doses from exposure to gaseous and liquid effluents from Marble Hill (see previous question 470.2 and response). In addition, provide a brief description of the format for input parameters (e.g., excerpts from the Users Manual for the computer code).

480.2 Discuss how the Marble Hill plant design satisfies the
(FSAR 6.2.1.1) concerns identified in IE Bulletin No. 80-04, "Analysis
(SRP 6.2.1.1.A) of a PWR Main Steam Line Break with Continued Feedwater Addition", dated February 8, 1980.

480.3 FSAR subsection 6.2.5.2.1 states that each hydrogen re-
(FSAR 6.2.5.2.1) combiner is designed to process between 50 and 90 scfm
(SRP 6.2.5) of gas; elsewhere in subsection 6.2.5 the recombiner flowrate is given as 90 scfm. Discuss how the recombiner flowrate will be regulated to ensure that the flowrate of containment atmosphere through either recombiner is at least 90 scfm.

480.4 Verify that the two hydrogen recombiners receive their
(FSAR 6.2.5.2.1) electrical power from different, independent Class 1E
(SRP 6.2.5) power supplies. In addition, verify that each recombiner receives power from the same supply that serves the system isolation valves.

480.5 The Marble Hill response to Byron/Braidwood Question 022.65
(FSAR 6.2.5.2.1) states that all electrical components of the hydrogen
(SRP 6.2.5) recombiner system are Class 1E, except for the hydrogen analyzer cell, the trickle heat subsystem, the thermocouple test panel, and certain space heaters. Explain why these components are not required to be Class IE.

480.6 (FSAR 6.2.5.3.1.2) (SRP 6.2.5) The Byron/Braidwood application considers the decomposition of organic materials as a source of hydrogen following a LOCA. The Marble Hill application does not consider this potential source; discuss the rationale for this difference.

492.1 A "Summary of Differences from SRP" was submitted by Public Service Company of Indiana (PSI) for Marble Hill. Although the Marble Hill loose parts monitoring system (LPMS) is not a replicate of Byron/Braidwood, Marble Hill in this submission has addressed the same questions 221.1 and 221.2 on the LPMS which were previously asked for the Byron/Braidwood plants. However, the responses are not complete. Also, PSI has taken exception to some items in Regulatory Guide 1.133. These items relate to seismic qualification, redundancy, separation, and in-containment calibration. Also Item C.5.b, Section D and some of the technical requirements are not agreed to. However, the licensee has evaluated the requirements against an early draft version of Regulatory Guide 1.133 and some of the requirements have been modified in the final version, Revision 1, May 1981. The licensee has not provided justification for these exceptions other than arguments with the Regulatory Guide 1.133 criteria. Since these criteria have been used for licensing for several years and since the cited version of Regulatory Guide 1.133 was issued with due consideration for industry comments, the justification provided is unacceptable. We will require the licensee to provide a LPMS consistent with the provisions of Regulatory Guide 1.133 as has been provided for other licensed reactors and to commit to provide prior to power operation a final design report which contains the following:

1. An evaluation of the LPMS for conformance to Regulatory Guide 1.133.

2. A description of the system hardware, operation and implementation of the loose parts detection program after start-up testing. This should also include the baseline data and alarm settings.

3. A description and evaluation of diagnositc procedures used to confirm the presence of a loose part.

4. A description of the operator training program.

A sample table of contents of the LPMS description is provided in Enclosure 1.

Finally, it should be noted that Regulatory Guides 1.133 and 1.70 state that the loose parts detection program should be described in Section 4.4.6 "Instrumentation Requirements."

492.2 In the "Summary of Differences from SRP" provided by PSI it is stated that the instrumentation for inadequate core cooling (ICC) for Marble Hill is to be a replicate of the Byron/Braidwood plant. However, the staff requires that plant-specific installations of ICC instrumentation be reviewed and approved for unambiguous indication of ICC. Therefore, the staff will require the applicant to provide documentation of its ICC system, as defined by NUREG-0737, Item II.F.2, for staff review and approval prior to issuance of an operating license. Pertinent submittals describing the generic design on the Byron/Braidwood docket may be referenced.

492.3 The applicant for Byron Station Units 1 and 2 used the improved thermal design procedure (ITDP) and was required to provide justification for uncertainties, variances, and distributions used in the ITDP. The ITDP considers statistical variations to obtain a DNB uncertainty factor. Application of this uncertainty parameter leads to a limiting DNBR value to be used for accident analysis in conjunction with input parameters at their nominal limit values. Marble Hill, as a replicate of Byron, will also be required to provide the same analysis in consideration of plant-specific uncertainty values. This should include a primary loop flow measurement uncertainty analysis incorporating plant-specific instrumentation.

492.4 The following items relate to the Technical Specifications which should include:

1. A declaration that prohibits N-1 loop operation unless it is adequately justified in plant-specific analysis.

2. Appropriate surveillance to ensure acceptable flow rates and to recognize crud buildup.

3. A discussion in the basis of the Technical Specification of any generic or plant-specific margins that have been used to offset the reduction in departure to nucleate boiling ratio (DNBR) due to rod bowing.

ATTACHMENT 1 TO Q492.1

SAMPLE TABLE OF CONTENTS
LOOSE PART DETECTION PROGRAM DESCRIPTION

630.1 Discuss the program which will provide the training to Reactor Operators and Senior Reactor Operators in the following areas:

(a) Recognition of emergency conditions.

(b) Classification of observed emergency conditions in accordance with the Emergency Classification System.

(c) Notification of emergency to off site authorities.

(d) Recommendation of protective actions to off site authorities.

(e) Direction of station staff to take protective actions.

(Ref. NUREG-0800, Sections 13.2.1.I.B.1 and 13.2.1.II.1.B)

630.2 Discuss the certifications completed pursuant to Sections 55.10(a)(6), 55.33(a)(4) and (5) of 10 CFR Part 55. Provide the title of the individual who will certify the eligibility of individuals for licensing or renewal of license.
(Ref. Enclosure 1 of H. R. Denton's March 20, 1980 letter, Section A.3)

630.3 Discuss how each of the topics as described in Sections 21 and 22 of 10 CFR Part 55 will be covered in the courses of the training program for Reactor Operators and Senior Reactor Operators.
(Ref. NUREG-0800, Section 13.2.1.I.B.1)

630.4 With respect to the simulator training program, provide the following information:

(a) The details of the program in accordance with the guidelines as specified in the Regulatory Guide 1.149.

(b) Discussion of the systematic observation or certification examination provided to demonstrate the candidate's ability.
(Ref. NUREG-0800, Sections 13.2.1.I.B.2 and 13.2.1.II.2)

(c) To what extent does the simulator training for Reactor Operators include training in differentiating kinds of instrument failures from transient or accident conditions.

(d) To the extent that some kinds of instrument failures are not simulated during training, explain whether the omission is detrimental to the safe operation of the reactors. (Ref. Licensing Board Chairman's questions for Perry Nuclear Power Plant)

630.5 Provide the details of the on site training program, including a course description and the duration of the course. The program should distinguish between classroom training and on-the-job training, before and after the initial fuel loading. (Ref. NUREG-0800, Section 13.2.1.I.B.3)

630.6 Provide the qualification of the instructors in the training program and the requalification program administered to the instructors in order to have them remain certified as instructors, as specified in Enclosure 1 of H. R. Denton's March 28, 1980 letter to all power reactor applicants and licensees. (Ref. NUREG-0800, Section 13.2.1.I.B.4)

630.7 The training program for Reactor Operators and Senior Reactor Operators should be modified to provide emphasis in heat transfer, fluid flow, thermodynamics, and reactor and plant transients. Discuss in detail how each topic as specified in Enclosure 2 of H. R. Denton's March 28, 1980 letter will be covered.

630.8 Provide detailed responses (point-by-point) for each of the following TMI Action Plan requirements:

(a) I.A.2.1 - Immediate Upgrading of Reactor Operator and Senior Reactor Operator Training and Qualifications.

(b) I.A.2.3 - Administration of Training Programs

630.9 Provide a training program for mitigating core damage as described in II.B.4 of NUREG-0737 in accordance with the guidance as specified in Enclosure 3 of H. R. Denton's letter dated March 28, 1980. Provide a listing of those individuals and their qualifications who must participate in the training program and provide a schedule for that training as related to the presently scheduled fuel load date.

630.10 With regard to the requalification training for licensed operators, provide the following modification to the programs:

(a) The content of the lecture series should be modified to include instruction in heat transfer, fluid flow, thermodynamics, plant transients and mitigation of accidents involving a degraded core as specified in Enclosure 1 of H. R. Denton's letter dated March 28, 1980.

(b) The on-the-job training program should be modified to include the control manipulations as specified in Enclosure 4 of H. R. Denton's letter dated March 28, 1980.

(c) If a non-plant-specific simulator is to be used for the training, provide the identity of the simulator which will comply with Regulatory Guide 1.149.
(Ref. NUREG-0800, Section 13.2.1.I.B.2)

In addition, discuss in detail how the requirements as described in Section 1, Sections 3.b., c, and d, and Sections 4.c, d, and e of Appendix A, 10 CFR Part 55 will be met.
(Ref. NUREG-0800, Sections 13.2.1.II.1 and 5)

630.11 Provide a detailed description of the training program for the Shift Technical Advisor in accordance with the guidance as specified in NUREG-0737, Appendix C.

630.12 With respect to the fire protection training, provide detailed discussion to show how each of the guidelines as described in the Standard Review Plan (NUREG-0800), Section 13.2.2.II.6 and Branch Technical Position CMEB 9.5.1 Sections C.3.d.(i), (k) and (l) will be met.

630.13 Provide and justify the number of persons for whom training is planned in preparation for senior operator and operator examinations prior to criticality.
(Ref. NUREG-0800, Section 13.2.1.II.4)

630.14 Provide and justify the number of non-licensed personnel for whom training is planned prior to criticality.
(Ref. NUREG-0800, Section 13.2.2.II.4)

ENCLOSURE 2

Identification Numbers To Be Used By Contributing NRR Branches For Review Questions

DIVISION	BRANCH	REVIEW AREA	IDENTIFICATION NUMBER
DL	Responsible Project Branch	Miscellaneous	100
DL	ORAB	Systems	110
DL	ORAB	Implementation	112
DE	MEB	All	210
DE	GSB	Seismology	230
DE	GSB	Geology	231
DE	SGEB	Structural Engineering	220
DE	SGEB	Geotechnical Engineering	241
DE	MTEB	Inservice Inspection	250
DE	MTEB	Component Integrity	251
DE	MTEB	Materials Application	252
DE	EQB	Environmental Qualification	270
DE	EQB	Seismic & Dynamic Load Qualification	271
DE	CEB	Fire Protection	280
DE	CEB	Chemical Technology	281
DE	EHEB	Hydrologic Engineering	240
DE	EHEB	Environmental Engineering	290
DE	SAB	Regional Impact Analysis	310
DE	SAB	Site Analysis	311
DE	SAB	Antitrust & Economic Analysis	320
DSI	ASB	All	410
DSI	ICSB	All	420
DSI	PSB	All	430
DSI	RSB	All	440
DSI	AEB	All	450
DSI	METB	Meteorology	451
DSI	METB	Effluent Treatment	460
DSI	RAB	Radiological Impact	470
DSI	RAB	Radiation Protection	471
DSI	CSB	All	480
DSI	CPB	Fuels	490
DSI	CPB	Physics	491
DSI	CPB	Thermal-Hydraulics	492

DIVISION	BRANCH	REVIEW AREA	IDENTIFICATION NUMBER
DHFS	OLB	All	610
DHFS	HFEB	All	620
DHFS	LQB	All	630
DHFS	PSRB	All	640
DST	RRAB	All	720
DST	GIB	All	730
CRBRPO	Technical Review	All	760
IE/EQA	QAB	All	260
IE/DEPER	EPB	All	810
NMSS/DS	PSLB	All	910