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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

before the

ATOMIC SAFETY AND LICENSING BOARD

In the Matter of PUBLIC SERVICE COMPANY OF NEW HAMPSHIRE, et al. (Seabrook Station, Units 1 & 2)	Docket Nos. 50-443 OL 50-444 OL

APPLICANTS' ANSWERS TO
"NECNP FIRST SET OF INTERROGATORIES
AND REQUEST FOR DOCUMENTS
TO APPLICANTS ON CONTENTIONS
I.D.1, I.D.2, I.D.3, I.D.4, I.F,
I.G, I.I, I.L, I.M, I.N, AND I.U."

Pursuant to 10 CFR § 2.740b, the Applicants hereby respond to the "NECNP First Set of Interrogatories and Request for Documents to Applicants on Contentions I.D.1, I.D.2, I.D.3, I.D.4, I.F, I.G, I.I., I.L, I.M, I.N, and I.U ," served on them by mail on October 29, 1982.

D503

GENERAL MATTERS

1. Definitions. The Applicants object to so much of the definition "identify" as is set forth in paragraph 2 under "Instructions for Use" as purports to require the Applicants to set forth the content of documents identified, inasmuch as, since NECNP has requested (or may request) production of identified documents for inspection and copying, an attempt to summarize or duplicate the contents of the document in an answer to an interrogatory is a patently unreasonable and bootless waste of time and effort.

The Applicants object to so much of the same definition as calls for "the present custodian . . . of any and all copies of the document" on the grounds that most of the documents to be identified are published documents and all have been widely circulated, with the result that this request would be unreasonably burdensome and probably impossible to respond to and that the information called for is not relevant to any admitted contention within the meaning of 10 CFR § 2.740(b)(1).

The Applicants object to the definitions contained in paragraph 3 and 4 under the heading "Instructions for Use" on the grounds that these appear to be incomprehensible and not to be related to any of the interrogatories actually propounded.

2. Production of Documents. The Applicants will make the documents for which production is called for available for inspection and copying at one or more appropriate places at a time to be mutually agreed upon by counsel for NECNP and the Applicants.

Please note that all responses to Requests for Additional Information (RAI's) are on file in the Public Document Room.

SPECIFIC INTERROGATORIES

Interrogatory No. I.D.1-1

Question:

1. What is the Applicants' position with respect to NECNP Contention I.D.1.? State all facts and opinions and identify and provide access to all documents on which that position is based.

Answer:

PSNH intends to test the reactor vessel welds during pre-service and inservice examinations using the requirements of ASME Section XI as specified in the PSI/ISI program and per the requirements of Regulatory Guide 1.150 as amended in Recommended Changes to Regulatory Guide 1.150 prepared by the "Ad Hoc Committee of Electric Utility Industry" and presented to the U.S. Nuclear Regulatory Commission in August 1982.

The specific implementation of the Regulatory Guide will be detailed in the "Reactor Vessel Examination Plan" which is in the course of preparation by PSNH's PSI contractor.

PSNH, therefore, contends that it does comply with GDC 1 with respect to pre-service and inservice examinations.

Interrogatory No. I.D.1-2

Question:

2. Identify all individuals whom Applicants expect to call as witnesses with respect to NECNP Contention I.D.1., and identify all documents on which Applicants expect to rely at the hearing with respect to this contention.

Answer:

Applicants have not yet determined which, if any, witnesses they will call nor have they identified the documents on which their testimony will be based on for the above contention.

Interrogatory No. I.D.1-3

Question:

3. List all provisions of Reg. Guide 1.150 which Applicants have implemented or intend to implement.

a. Describe in detail Applicants' program for implementing those provisions.

Answer:

See Recommended Changes to Regulatory Guide 1.150 prepared by the "Ad Hoc Committee of the Electric Utility Industry" presented to the USNRC in August 1982.

Interrogatory No. I.D.1-4

Question:

4. List all of the provisions of Reg. Guide 1.150 which the Applicants have not implemented or do not intend to implement.

a. Of those items, which do Applicants consider to be not viable within the current state of technology? State the basis for this conclusion.

b. State any other reasons supporting Applicants' decision not to comply with Reg. Guide 1.150.

Answer:

See Recommended Changes to Regulatory Guide 1.150 prepared by the "Ad Hoc Committee of the Electric Utility Industry" presented to the USNRC in August 1982.

Interrogatory No. I.D.1-5

Question:

5. Where Applicants do not meet the requirements to Reg. Guide 1.150, describe all alternative measures which have been taken or will be taken to monitor the integrity of reactor vessel welds, under both preservice and inservice conditions.

a. Is it Applicants' position that the implementation of these measures provides an assurance of safety which is equivalent to that obtained by compliance with Reg. Guide 1.150? State the reasons supporting your conclusion.

Answer:

PSNH intends to meet the requirements to the extent and in the manner set forth in their response to Interrogatory 1.D.1-1. There will be no alternate measures required.

a. Yes. The measures described in Recommended Changes to Regulatory Guide 1.150 satisfy the intent of the Regulatory Guide and, further, since the original guide was thought to be unusable by a large portion of the industry, the PSNH position to use this Recommended Change enhances the assurance of safety over the original Regulatory Guide.

Interrogatory No. I.D.1-6

Question:

6. Do Applicants believe they must comply with any or all provisions of Reg. Guide 1.150 in order to satisfy General Design Criterion 1 of Appendix A to 10 CFR Part 50?

a. For each provision Applicants believe is required, state the reasons supporting this conclusion.

b. For each provision Applicants believe is not required, state the reasons supporting this conclusion.

Answer:

No. The Applicants believe that compliance with the provisions of Regulatory Guide 1.150, as amended as set forth in Recommended Changes to Regulatory Guide 1.150 is sufficient to demonstrate satisfaction of GDC 1. The Applicants have not determined whether compliance with either the amended or unamended provisions of Reg. Guide 1.150 is the only way in which GDC 1 may be satisfied.

a. & b. The format of Section 3 of Recommended Changes to Regulatory Guide 1.150 provides the original and alternate positions on the guide. Section 4 provides the justification for the changes.

Interrogatory No. I.D.1-7

Question:

7. Describe and provide the test results for all tests performed to date on reactor vessel welds.

Answer:

No PSI or ISI has been performed on the vessel to date. Other examinations performed on the vessel are documented in the manufacturer's data package.

Interrogatory No. I.D.1-8

Question:

8. Identify and provide access to any and all documents relied upon or referred to in preparing the response to the interrogatories regarding Contention I.D.1.

Answer:

1. Recommended Changes to Regulatory Guide 1.150 prepared by the "Ad Hoc Committee of the Electric Utility Industry" and presented to the USNRC in August 1982.
2. USNRC Regulatory Guide 1.150 - June 1981.
3. SB FSAR.
4. The manufacturers data package referred to in response to Interrogatory I.D.1-7.

Interrogatory No. I.D.2-1

Question:

1. What is the Applicants' position with respect to NECNP Contention I.D.2.? State all facts and opinions and identify and provide access to all documents on which that position is based.

Answer:

The Applicants disagree with the contention that the proposed testing of protection systems and actuation devices fails to meet the requirements of GDC 21. NUREG-0737, Task II.D.1, Performance Testing of

Boiling-Water Reactor and Pressurized-Water Reactor Relief and Safety Valves, does not apply to periodic testing of protection systems and actuation devices.

See FSAR 1.8 (discussion relating to conformance with Regulatory Guides 1.22 and 1.118), Sections 7.1.2.5, 7.1.2.11, 7.2.2.2(c), 7.2.3. and 7.3.2.2(e) for a discussion of the testing program and justification for not testing the actuated equipment listed on FSAR page 1.8-9. Note that page 1.8-9 was revised in Amendment 45 and only contains 11 items that are not tested at power.

Interrogatory No. I.D.2-2

Question:

2. Identify all individuals whom Applicants expect to call as witnesses with respect to NECNP Contention I.D.2., and identify all documents on which Applicants expect to rely at the hearing with respect to this contention.

Answer:

Applicants have not yet determined which, if any, witnesses they will call nor have they identified the documents on which their testimony will be based on for the above contention.

Interrogatory No. I.D.2-3

Question:

3. For each of the twelve functions described at FSAR 1.8-9 which are not to be tested at full power, state the reasons the function is not to be tested at power, and provide the justification in each instance for the failure to test at full power. Identify and provide access to all documents relied on in support of these positions.

Answer:

See FSAR Section 7.1.2.5.

Interrogatory No. I.D.2-4

Question:

4. Define testing at the "device or system level" as Applicants use the term. Identify other levels at which protection functions can be tested and state whether any of the twelve functions are tested at these levels.

Answer:

"Device and/or system level" refers to the classification of the equipment being tested. Items 1 through 5 are considered system level equipment; Items 6 through 11 are considered device level equipment.

See the response to Interrogatory 1 for testing being performed.

Interrogatory No. I.D.2-5

Question:

5. For each of the twelve functions, state whether it is the "actuation device" and/or "actuated

equipment" (as the terms are used in Reg. Guide 1.22) which is (are) not tested.

Answer:

In Items 1 through 5 the actuation device/system is not tested; in Items 6 through 11 the actuated equipment is not tested.

Interrogatory No. I.D.2-6

Question:

6. For each of the twelve protection functions, state whether any of the alternative means of testing actuation functions described in Reg. Guide 1.22 Section C., paragraph 3 will be applied. For those functions where the alternative measures are not taken, describe the reasons and state the justification therefor, and identify any and all documents relied upon in making that determination.

Answer:

See the response to Interrogatory 1 for testing being performed.

Interrogatory No. I.D.2-7

Question:

7. For each of the twelve protection functions, state whether the Applicants consider the reasons for not testing them at full power to be compelling, and state the justification for that position.

Answer:

See the response to Interrogatory 1 for justification for not testing certain functions.

Interrogatory No. I.D.2-8

Question:

8. State whether Applicants believe that Task II.D.1. of NUREG-0737 requires testing at power of any of the twelve functions listed at FSAR 1.8-9.

a. If so, state which of those functions must be tested and describe Applicants' reasons for not complying with the requirement. Identify and provide access to all documents relied on in support of this determination.

b. If not, state the reasons supporting Applicants' conclusion that Task II.D.1. does not apply. Identify and provide access to all documents relied on in support of this determination.

Answer:

No.

a. N/A.

b. See the response to Interrogatory 1.

Interrogatory No. I.D.2-9

Question:

9. State the reasons for Applicants' conclusion that "There is no practicable system design that would permit operation of the equipment without adversely affecting the safety or operability of the plant." FSAR at 1.8-9. Identify and provide access to all documents supporting that conclusion.

a. Define the term "practicable" as used by Applicants.

b. Describe the adverse effects of testing each function.

Answer:

See the response to Interrogatory 1 for justification for not testing certain functions.

a. The Applicants have not used the term "practicable" in any specialized fashion.

b. See the response . . . .

Interrogatory No. I.D.2-10

Question:

10. State the reasons for Applicants' conclusion that "The probability that the protection system will fail to initiate the operation of the equipment is, and can be maintained, acceptably low without testing the equipment during reactor operation. FSAR at 1.8-10. Identify and provide access to all documents relied on in reaching this conclusion.

a. Describe Applicants' definition of acceptably low probability.

Answer:

See FSAR Section 7.1.2.5.

Interrogatory No. I.D.2-11

Question:

11. Describe any means other than testing at full power by which Applicants have tested or evaluated or intend to test or evaluate the reliability of the twelve safety functions listed at FSAR 1.8-9.

a. Do Applicants consider that these measures provide an assurance that protection systems will operate when called upon? State the justification for this conclusion and provide access to all documents relied upon.

b. Do Applicants consider any such alternative measures to provide an assurance of safety equivalent to that achieved through compliance with Reg. Guide 1.22? Describe the reasons supporting this determination and identify and provide access to all documents relied upon.

Answer:

See FSAR Section 7.1.2.5.

a. Yes, see FSAR Section 7.1.2.5.

b. The alternative testing methods described in FSAR Section 7.1.2.5 are in conformance with the guidance provided in Regulatory Guide 1.22, see FSAR Section 7.1.2.5.

Interrogatory No. I.D.2-12

Question:

12. Do Applicants believe that compliance with General Design Criteria 20 and 21 requires satisfaction of any and all of the terms of Reg. Guide 1.22?

a. For each term Applicants believe is required, state the reasons supporting this conclusion.

b. For each term Applicants believe is not required, state the reasons supporting this conclusion.

Answer:

The Applicants believe that compliance with the guidance of Regulatory Guide 1.22 is sufficient to demonstrate satisfaction of GDC 21 and 22. The Applicants have not determined whether compliance with this guidance is the only way in which such satisfaction may be demonstrated.

a. Regulatory Guide 1.22, in its entirety, has been used for guidance on testing of actuation functions for over 10 years and is accepted by the NRC and the nuclear industry as an adequate method for testing these functions.

b. N/A.

Interrogatory No. I.D.2-13

Question:

13. Describe and provide the results of all tests performed to date on protection system actuation functions.

Answer:

No testing has been performed as Seabrook is under construction.

Interrogatory No. I.D.2-14

Question:

14. Identify and provide access to all documents referred to or relied upon in preparing the response to the Interrogatories on NECNP Contention I.D.2.

Answer:

The documents referred to or relied on in preparing the response to the interrogatories regarding Contention I.D.2 are identified in the various responses. These documents will be available for inspection and copying at one or more appropriate places at a time to be mutually agreed upon by counsel for NECNP and the Applicants.

Interrogatory No. I.D.3-1

Question:

1. What is the Applicants' position with respect to NECNP Contention I.D.3.? State all facts and opinions and identify and provide access to all documents on which that position is based.

Answer:

GDC 21 deals with the reliability and testability of protection systems. No portion of the Leakage Detection System initiates automatic control and for that reason no such portion is properly classified as a protection system as defined by GDC 20.

The Leakage Detection System is implemented using the guidelines of Regulatory Guide 1.45, including provisions for testing, which describes acceptable methods of compliance with GDC 30. See FSAR Section 5.2.5 and response to RAI 420.55.

Interrogatory No. I.D.3-2

Question:

2. Identify all individuals whom Applicants expect to call as witnesses with respect to NECNP Contention I.D.3., and identify all documents on which the Applicants expect to rely at the hearing with respect to this contention.

Answer:

Applicants have not yet determined which, if any, witnesses they will call nor have they identified the documents on which their testimony will be based on for the above contention.

Interrogatory No. I.D.3-3

Question:

3. Identify the location and function of all components of Applicants' reactor coolant pressure boundary leakage detection system. Identify all instruments used for leakage detection, and the parameters which they measure.

Answer:

See FSAR Section 5.2.5.

Interrogatory No. I.D.3-4

Question:

4. Identify all provisions of Reg. Guide 1.45 with which Applicants comply, and explain how compliance is achieved.

Answer:

The Leakage Detection System is implemented in accordance with the guidance of Regulatory Guide 1.45, and is in general conformance with each of the items identified in the regulatory position (Section C) of the guide.

a. Item 1 is discussed in FSAR Subsection 5.2.5.1.

b. Item 2 is discussed in FSAR Subsections 5.2.5.2, 5.2.5.3 and 5.2.5.5.

c. Item 3 is discussed in FSAR Subsection 5.2.5.3.6.

d. Item 4 is discussed in FSAR Subsection 5.2.5.4.

e. Item 5 is discussed in FSAR Subsection 5.2.5.5 and Table 5.2-8.

f. Item 6 is discussed in FSAR Subsection 5.2.5.6.

g. Item 7 is discussed in FSAR Subsection 5.2.5.7.

h. Item 8 is discussed in FSAR Subsection 5.2.5.8.

i. Item 9 is discussed in FSAR Subsection 5.2.5.9 and in Chapter 16.

(Please note that FSAR 5.2.5.6 is under consideration for revision or supplementation.)

Interrogatory No. I.D.3-5

Question:

5. Identify all provisions of Reg. Guide 1.45 with which Applicants do not comply. Explain your reasons for noncompliance. Identify any alternative measures employed by Applicants to satisfy General Design Criterion 30 of Appendix A to 10 CFR Part 50.

Answer:

As per the discussion for Interrogatory I.D.3-4 above, the Leakage Detection System conforms to the guidelines of Regulatory Guide 1.45.

Interrogatory No. I.D.3-6

Question:

6. Identify which components of the reactor coolant pressure boundary leakage detection system must be tested or calibrated under IEEE-279-71. Of these, identify components which must be tested and/or calibrated at power.

Answer:

Those systems which must be periodically tested and their respective testing interval are detailed in the plant Technical Specifications. FSAR Chapter 16, Section 3/4.4.7.1.

Interrogatory No. I.D.3-7

Question:

7. Identify all provisions of IEEE-279-71 with which Applicants' reactor coolant boundary leakage detection system complies, and explain how compliance is achieved.

Answer:

IEEE 279-1971, Requirement 4.10 is complied with as discussed in FSAR Section 5.2.5.8.

Interrogatory No. I.D.3-8

Question:

8. Identify all provisions of IEEE-279-71 with which Applicants' reactor coolant pressure boundary leakage detection system does not comply. Explain your reasons for noncompliance. Do Applicants believe that in spite of any noncompliance with IEEE-279-71, the design of the reactor coolant pressure boundary leakage detection system complies with General Design Criterion 30? Explain the reasons for your answer.

Answer:

As has been previously indicated, no portion of the Leakage Detection System is classified as part of a protection system thus compliance to the criteria of IEEE Standard 279-1971 other than that for Paragraph 4.10, as indicated in Regulatory Guide 1.45, is not dictated.

Interrogatory No. I.D.3-9

Question:

9. Do Applicants believe that compliance with Reg. Guide 1.22 requires testing of the reactor coolant pressure boundary leakage detection system at power? Explain the reasons for your answer.

Answer:

No. Regulatory Guide 1.22 deals with the testing of protection systems. Since no portion of the Leakage Detection System is part of a protection system, conformance to the guidelines of Regulatory Guide 1.22 is not required.

Interrogatory No. I.D.3-10

Question:

10. Describe any and all components of the leakage detection system which are to be tested at power. Describe all components of the leakage detection system which are not to be tested at power. Explain for each of these the reasons for not testing them at power, and identify any alternative means of testing them.

Answer:

The testing of the various constituents of the Leakage Detection System is described in FSAR

Subsection 5.2.5.8. Those systems which require periodic calibration and functional testing and their respective testing interval are detailed in the plan Technical Specifications, FSAR Chapter 16, Section 3/4.4.7.1.

Interrogatory No. I.D.3-11

Question:

11. Explain the following statement made at FSAR page 1.8-17:

> The response time of containment airborne particulate radioactivity monitoring devices to detect a one gpm leakage rate may vary from the given requirement, depending on equilibrium conditions in containment at the inception of the leak.

In particular, describe the parameters of the response time; the time requirement and its source; and the causes and effects of changes in equilibrium conditions in the containment at the inception of a leak.

Answer:

The ability to detect a leakage rate of 1 gpm across the reactor coolant pressure boundary via the use of the containment airborne particulate radioactivity monitor is recommended by Regulatory Guide 1.45.

The response time of this method of monitoring is dependent on a large number of parameters, and is generally determined for the worst case or near worst case situation by optimizing these various parameters. Among the important variables are the primary coolant source terms, the length of power operation, the containment conditions (i.e., use of atmospheric cleanup systems), and the operational parameters of the Detection System. An analysis to obtain the response time would use:

a) A minimized value for the reactor coolant source terms,

b) An equilibrium value for the activity already in the containment atmosphere (this is termed the background), and

c) Fixed values for the leak rate, detector specifications and other pertinent parameters.

Under these conditions, the response time obtained may be considered a maximum. Any change of these parameters would result in a faster response time.

Interrogatory No. I.D.3-12

Question:

12. Identify and provide access to any and all documents referred to or relied on in preparing the response to the interrogatories regarding Contention I.D.3.

Answer:

The documents referred to or relied on in preparing the response to the interrogatories regarding Contention I.D.3 are identified in the various responses. These documents will be available for inspection and copying at one or more appropriate places at a time to be mutually agreed upon by counsel for NECNP and the Applicants.

Interrogatory No. I.D.4-1

Question:

1. What is the Applicants' position with respect to NECNP Contention I.D.4.? State all facts and opinions and identify and provide access to all documents which support that position.

Answer:

The Applicants disagree with the contention in that compliance with IEEE 338-1975, as endorsed by Regulatory Guide 1.118, Revision 1, is sufficient to demonstrate satisfaction of GDC 21. These documents formed the basis for the design of Seabrook at the construction permit stage and are considered adequate for operation of Seabrook.

The Standards and Regulatory Guides are guidance documents that describe one acceptable method for performing periodic testing. These documents will be used when preparing the Seabrook periodic testing program. The procedures that implement the testing program will be available for review three months prior to fuel loading.

Interrogatory No. I.D.4-2

Question:

2. Identify all individuals whom Applicants expect to call as witnesses with respect to NECNP Contention I.D.4., and identify all documents on which Applicants expect to rely at the hearing respecting this contention.

Answer:

Applicants have not yet determined which, if any, witnesses they will call nor have they identified the documents on which their testimony will be based on for the above contention.

Interrogatory No. I.D.4-3

Question:

3. Identify and describe each and every system which Applicants believe must comply with GDC 18, GDC 21, 10 CFR § 50.55a and Criterion XI of Appendix B to 10 CFR Part 50.

Answer:

FSAR Appendix 3H lists the safety-related electrical equipment (Class 1E) that must comply with GDC 18, GDC 21, 10 CFR § 50.55a or Criterion XI of

Appendix B to 10 CFR § 50. The FSAR describes the system that these equipments are part of.

Interrogatory No. I.D.4-4

Question:

4. Describe the BOP electric power and safety system testing.

Answer:

The Preoperational Test Program is described in detail in the FSAR, Chapter 14.

Operational testing requirements are dictated by Section 3/4 of Technical Specification. Procedures to be used to satisfy these requirements are presently being prepared and will be available for review 60 days prior to core load.

Interrogatory No. I.D.4-5

Question:

5. Describe NSSS electric power and safety system testing.

Answer:

The Preoperational Test Program is described in detail in the FSAR, Chapter 14.

Operational testing requirements are dictated by Section 3/4 of Technical Specification. Procedures to be used to satisfy these requirements are presently being prepared and will be available for review 60 days prior to core load.

Interrogatory No. I.D.4-6

Question:

6. Identify the standards with which Applicants have complied in order to provide acceptable methods for the periodic testing of electric power and protection systems, to satisfy GDC 18 and 21, 10 CFR § 50.55a, and Criterion XI of Appendix B to 10 CFR Part 50. With regard to each such standard identified above, state clearly where that standard can be found, whether in IEEE-338-1975, IEEE-338-1977, Regulatory Guide 1.118, or in any other publication or code.

Answer:

Protection systems periodic testing is discussed in Chapter 7 of the FSAR as well as the Technical Specifications. The standards are listed in Table 7.1-1, Chapter 7 and conformance to listed standards is discussed in Section 7.1.

The Electrical Power System testing is discussed in Chapter 8 of the FSAR as well as the Technical Specifications. The IEEE Standards are listed in Section 8.1.5.2 of Chapter 8.

Interrogatory No. I.D.4-7

Question:

7. Identify all differences between IEEE Std. 338-1975 and IEEE Std. 338-1977.

Answer:

Our response to RAI 430.3 provides a comparison of the differences between IEEE Standard 338-1975 and IEEE Standard 338-1977.

Interrogatory No. I.D.4-8

Question:

8. Identify all ways in which Applicants meet the design and operational criteria for performance of periodic testing of safety systems. State whether or not Applicants have provided an alternative method for complying with the NRC regulations and requirements cited in Interrogatory No. 6 above, and describe in detail any such alternative methods.

Answer:

The design of the safety systems (as defined by GDC 21) is discussed in FSAR Sections 7.1, 7.2, 7.3, 8.1 and 8.3 and meet the requirement of IEEE 279-1971. Additionally, conformance to IEEE 338-1975 is discussed in FSAR Sections 7.1.2.11 and 8.1.5.2.

Periodic testing of Engineered Safeguards Features Actuation System and Reactor Trip System is described in FSAR Subsections 7.2.2 and 7.2.3 and complies with Regulatory Guide 1.22 as described in Section 1.8.

Interrogatory No. I.D.4-9

Question:

9. Identify the NSS supplier. Explain all instances in which the NSSS supplier will not guarantee that it will comply with the recommendations contained in IEEE-338-1975, or any superseding IEEE Standard.

Answer:

The NSSS is supplied by Westinghouse Electric Corporation. Compliance with IEEE 338-1975 is discussed in Section 7.1.2.11 and 8.1.5.7. Refer to response to Question 7 for a reference to the differences between IEEE 338-1975 and 338-1977.

Interrogatory No. I.D.4-10

Question:

10. Identify all systems which Applicants define as safety systems and therefore subject to periodic testing as required by GDC 21.

Answer:

GDC 21 addresses "Protection Systems" which is limited to 1) Reactor Trip System, 2) Engineered Safeguards Features Actuation System. Periodic testing is expanded to include those systems identified in Technical Specifications, Section 3/4 (see Answer 4).

Interrogatory No. I.D.4-11

Question:

11. State whether or not each such system will be tested under conditions that simulate the performance required of the system in the event of a design basis event. If the answer to the question above is no, explain why not.

Answer:

They will be.

Interrogatory No. I.D.4-12

Question:

12. Identify all cases in which Applicants intend to simulate design basis accident conditions for testing purposes. Describe the accident conditions in full, state why they were chosen, and describe how Applicants intend to simulate those conditions, with specific reference to all differences between simulated and actual accident conditions.

Answer:

Environmental conditions such as seismic events, radiation fields, temperature and moisture conditions are addressed in design qualification tests, and are not simulated for periodic testing. Qualification tests are discussed in detail in Section 3.11 of the FSAR. System conditions are simulated by injecting test signals into the process sensors. Refer to the response to Interrogatory I.D.4-4 for details of test procedures.

Interrogatory No. I.D.4-13

Question:

13. Describe how, if at all, the design of the safety systems provide[s] the capability for periodic testing simulating the performance required of the system in the event of a design basis event.

Answer:

Compliance of the system's design to IEEE 279-1971 and IEEE 308-1971 is discussed in detail in Chapters 7 and 8 of the FSAR. The referenced chapters also address capability for testing of Reactor Trip System and Engineered Safeguards Features Actuation System.

Interrogatory No. I.D.4-14

Question:

14. Describe how, if at all, the safety systems were designed so as to lessen, to the greatest degree possible, the impact of testing on plant availability and operation.

Answer:

See response to Interrogatory I.D.4-13.

Interrogatory No. I.D.4-15

Question:

15. Describe how, if at all, the test equipment will not cause a loss of independence between redundant channels or load groups.

Answer:

System design prohibits the testing of more than one channel at a time, in that a tripped condition will result if more than one channel is in the test mode. (See FSAR Section 7.2.2.2.) This design meets the requirements of IEEE 279-1971.

Interrogatory No. I.D.4-16

Question:

16. Describe how, if at all, there is sufficient redundancy within each safety system to provide redundancy even when degraded by a single random failure.

Answer:

Chapter 7 of the FSAR, Section 7.2 describes in detail redundances within the safety systems and the design criteria for the systems.

Interrogatory No. I.D.4-17

Question:

17. Describe how testing may be carried out in all instances when Applicants cannot practicably initiate protective action.

Answer:

See response to Interrogatory I.D.4-4.

Interrogatory No. I.D.4-18

Question:

18. Describe all categories of tests to be used by Applicants, as, for example, instrument checks or functional tests. Identify the systems to which these different categories of tests will be applied.

Answer:

Technical Specifications, Section 3/4, dictate the type of test and the frequency to be performed. Tests to be used to satisfy these requirements are presently being prepared and will be available for review 60 days prior to core load. These tests will meet the requirements of IEEE 338-1975, Sections 6.3.1 and 6.3.2.

Interrogatory No. I.D.4-19

Question:

19. Describe, for each system to be tested, the specific test procedure developed for that system, and the manner in which it meets the standard criteria contained in IEEE St. 338-1977, 6.4.

Answer:

The applicant has not committed to the 1977 issue of IEEE 338. The minimum requirement for procedures is IEEE 338-1975, Section 6.4. However, during the preparation of test procedures, the 1977 revision of

IEEE 338 will be reviewed, and complied with if it is reasonably achievable.

Test procedures are presently being prepared and will be available for review 60 days prior to core load.

Interrogatory No. I.D.4-20

Question:

20. Identify the test intervals and whether or not Applicants anticipate that the test intervals will change over the life of the two Seabrook plants. Describe the method by which Applicants calculated these test intervals.

Answer:

Test intervals are specified in the Seabrook Technical Specifications which is mandated by the NRC in the form of Standard Technical Specifications (Nuclear Regulatory Guide 452). The test intervals may not be lengthened without an approved license change. They may, however, be modified in the conservative direction if the Applicant feels that any protection function appears marginal, or requires more frequent adjustment. We anticipate no change in test intervals at this time.

Interrogatory No. I.D.4-21

Question:

21. Describe all test procedures according to which the periodic tests will be conducted. Identify and provide access to all documents containing or summarizing these test procedures.

Answer:

See response to Interrogatory I.D.4-4.

Interrogatory No. I.D.4-22

Question:

22. Explain why the NSSS supplier considers status, annunciating, display and monitoring functions to be control functions. Define a "reasonability check" and how these checks will be conducted for the above so-called "control functions."

Answer

These systems do not perform active safety functions. They are, therefore, considered "Control - Functions." Reasonability checks are defined in IEEE 378-1975, Section 6.3.1. See response to Interrogatory I.D.4-4.

Interrogatory No. I.D.4-23

Question:

23. Explain why response time testing will not be performed for control functions operated from protection system sensors or nuclear instrumentation systems detectors.

Answer:

A. Applicant's response to RAI 420.17 addressed this question and stated in part, "Response time testing will be performed only on those channels having a limiting response time established and credited in the safety analysis." In no case in the safety analysis is credit taken for the response time of a control system.

B. Nuclear instrumentation sensors are exempt from time response testing since "their worst case response time is not a significant faction of the total overall system response (i.e., less than 5%)." This exception is permitted by IEEE 338-1975.

Capacitance tests will be performed on detector-cable assemblies to detect changes in detector response characteristics.

Interrogatory No. I.D.4-24

Question:

24. Explain why the "expected environmental and mechanical configuration of the actual installation" will not be duplicated for the testing of sensors which must be removed to do response time testing. Describe any alternative method Applicants intend to adopt to ensure that sensor time response will be adequately tested.

Answer:

Response time tests of process sensors will be performed in situ. See response to RAI 420.17 for discussion of proposed response time testing program. Environmental conditions such as seismic events, radiation fields, temperature and moisture conditions are not simulated during testing. These conditions are addressed in design qualification tests which are discussed in detail in FSAR, Section 3.11.

Interrogatory No. I.D.4-25

Question:

25. Explain why the standard NSSS supplier scope protection system does not include design provisions to permit in situ testing of processor Nuclear Instrumentation System sensors.

Answer:

Process and nuclear instrumentation sensors are tested in situ, except where the state-of-the-art limits this practice. At this time, the only sensors that are removed for calibration are temperature sensors. It must be noted that the "Instrument Checks" required by Section 3/4 of the Seabrook Technical Specification are always performed in situ for all sensors.

Interrogatory No. I.D.4-26

Question:

26. State all instances in which actuated equipment will be simultaneously tested with associated protection system equipment, and state all instances where overlap testing will be substituted for such simultaneous tests.

Answer:

Refer to FSAR 1.8 for discussion of compliance with Regulatory Guide 1.22, Revision 0, "Periodic Testing of Protection System Actuation Features."

Interrogatory No. I.D.4-27

Question:

27. Describe all substitute tests Applicants propose to conduct when perturbing the monitored variable will not be practicable.

Answer:

The Applicants do not consider perturbing of the monitored variable to be a practicable method of performing periodic tests. Tests are performed by injecting a test signal (i.e., pressure, temperature, etc.) at the input of the process sensors. See response to Interrogatory I.D.4-4.

Interrogatory No. I.D.4-28

Question:

28. Describe how, if at all, the design of safety systems provides means to prevent the expansion of any bypass condition to redundant channels or load groups during testing operations.

Answer:

Chapter 7 of the FSAR Section 7.1.2.5, "Conformance to Regulatory Guide 1.22," and Section 7.1.2.6, "Conformance to Regulatory Guide 1.47," describe the design to prevent the expansion of any bypass conditions.

Interrogatory No. I.D.4-29

Question:

29. Describe how, if at all, the design of the particular systems permit redundant components to be tested independently where redundant components are used within a single channel or load group.

Answer:

Safety systems (as defined by GDC 21) design does not utilize redundant components within a single channel.

Interrogatory No. I.D.4-30

Question:

30. Describe Applicants' proposed testing relating to neutron detectors and how, if at all, such testing will confirm neutron detector response time

characteristics and at the same time avoid undue radiation exposure of plant personnel.

Answer:

In conjunction with the detector-cable capacitance tests outlined in the answer to Interrogatory I.D.4-23, the neutron detectors will be introduced to a neutron source, and neutron response verified. The results of these two tests will establish the base-line data for all future detector-cable capacitance tests. The performance of the neutron response test will be under the constant supervision of Health Physics personnel. Additionally, procedures to be used in the performance of this test will receive an "As Low As Reasonably Achievable" (ALARA) review prior to approval.

Interrogatory No. I.D.4-31

Question:

31. Describe whether temporary jumper wires Applicants propose to use meet all requirements of IEEE Std. 338-1977.

Answer:

The use of temporary jumper during the performance of testing is discussed in the Applicant's response to RAI 420.17.

Interrogatory No. I.D.4-32

Question:

32. Identify and provide access to any and all documents referred to or relied on in preparing the response to the interrogatories regarding contention I.D.4.

Answer:

The documents referred to or relied on in preparing the response to the interrogatories regarding Contention I.D.4 are identified in the various responses. These documents will be available for inspection and copying at one or more appropriate places at a time to be mutually agreed upon by counsel for NECNP and the Applicants.

Interrogatory No. I.F-1

Question:

1. What is the Applicants' position with respect to NECNP Contention I.F.? State all facts and opinions and identify and provide access to all documents on which that position is based.

Answer:

The Seabrook Station complies with Regulatory Guide 1.9 and with IEEE 323-1974. FSAR Sections 1.8, 8.1 and 8.3 describe compliance with Regulatory Guide 1.9. FSAR Sections 3.11 and 8.1 describe compliance with IEEE 323-1974.

Interrogatory No. I.F-2

Question:

2. Identify all individuals whom Applicants expect to call as witnesses with respect to NECNP Contention I.F. and identify all documents on which the Applicants expect to rely at the hearing with respect to this contention.

Answer:

Applicants have not yet determined which, if any, witnesses they will call nor have they identified the documents on which their testimony will be based on for the above contention.

Interrogatory No. I.F-3

Question:

3. State with particularity in what ways, if any, the diesel generator design fails to comply with Reg. Guide 1.9. State the justification for noncompliance.

a. For those areas of nonconformance, describe any alternative measures which Applicants have employed to meet General Design Criterion 17 of Appendix A to 10 CFR Part 50.

Answer:

Compliance with Regulatory Guide 1.9 is discussed in FSAR Sections 1.8, 8.1 and 8.3. Justification for any apparent areas of non-compliance is provided in these sections of the FSAR.

a. The design complies with all the requirements of General Design Criterion 17.

Interrogatory No. I.F-4

Question:

4. Describe the tests conducted by Applicants to environmentally qualify the diesel generator units at Seabrook. Describe the results of all such tests and identify all documents containing the results.

Answer:

The diesel generator equipment qualification file contains a description of all tests conducted to qualify the diesel generator units. The results of the tests are provided in the equipment qualification file presently at UE&C's Philadelphia offices.

Interrogatory No. I.F-5

Question:

5. Describe all tests the Applicants have not yet conducted but intend to conduct to environmentally qualify the diesel generator units at Seabrook.

Answer:

No additional qualification tests are envisioned.

Interrogatory No. I.F-6

Question:

6. Describe the type, model, and capacity of the diesel generator units at Seabrook.

Answer:

The Seabrook diesel generator sets are Colt-Pielstick PC2 diesel generator units rated at 6083 kW continuous. Additional details are provided in FSAR Section 8.3.

Interrogatory No. I.F-7

Question:

7. State whether or not Applicants have met the requirements of IEEE-323-1974.

a. If the answer to Interrogatory No. 7 is no, identify each and every aspect of noncompliance.

b. If the answer to Interrogatory No. 7 is no, describe any alternative method by which Applicants intend to comply with the requirements of GDC 17. State the justification for each.

Answer:

The Seabrook Station meets the requirements of IEEE Standard 323-1974.

Interrogatory No. I.F-8

Question:

8. Do Applicants comply with IEEE-323-1977 in every respect?

a. Identify any and all aspects of noncompliance.

b. For items of noncompliance, describe any alternative method by which Applicants intend to comply with the requirements of GDC 17. State the justification for each.

Answer:

IEEE Standard 323-1977 does not exist.

Interrogatory No. I.F-9

Question:

9. Describe all differences between IEEE-323-1974 and IEEE-323-1977.

a. Do Applicants believe that compliance with IEEE 323-1977 provides an equivalent assurance of safety as compliance with IEEE 323-1974? State the reasons for your answer.

Answer:

IEEE Standard 323-1977 does not exist.

Interrogatory No. I.F-10

Question:

10. Identify and provide access to any and all documents referred to or relied on in preparing the response to the interrogatories regarding Contention I.F.

Answer:

The documents referred to or relied on in preparing the response to the interrogatories regarding Contention I.F are identified in the various responses. These documents will be available for inspection and copying at one or more appropriate places at a time to be mutually agreed upon by counsel for NECNP and the Applicants.

Interrogatory No. I.G-1

Question:

1. What is the Applicants' position with respect to NECNP Contention I.G.? State all facts and opinions and identify and provide access to all documents which support that position.

Answer:

The Applicants disagree with the contention as the Seabrook wide range pressure transmitters (RC-PT-403 and 405) are located outside the containment and are not subject to the high energy line break environment that caused the inaccuracies addressed by IE Information Notice No. 82-11. Drawings 9763-M-506635 and 506636 show the location of subject transmitters.

Interrogatory No. I.G-2

Question:

2. Identify all individuals whom Applicants expect to call as witnesses with respect to NECNP Contention I.G. and identify all documents on which Applicants expect to rely respecting this contention.

Answer:

Applicants have not yet determined which, if any, witnesses they will call nor have they identified the documents on which their testimony will be based on for the above contention.

Interrogatory No. I.G-3

Question:

3. Identify and describe the function and/or use of the wide range pressure instruments supplied to Applicants by Westinghouse and used at the Seabrook plants.

Answer:

The wide-range pressure transmitters (RC-PT-403 and 405) are used to:

- Verify vessel NDT criteria,
- Maintain primary inventory subcooled (particularly with loss of off-site power),
- Establish correct conditions for RHR operation,
- Determine whether RCP operation should be continued,
- Determine whether high head safety injection should be terminated or reinitiated,
- Provide pressure input the power-operated relief valve controls for normal and low pressure operation, and
- Provide pressure interlocks for RC-V-22, 23, 87 and 88.

Interrogatory No. I.G-4

Question:

4. Describe and provide copies of any test results conducted by Applicants on any such instrument identified in Interrogatory No. 3 above.

Answer:

The qualification of these transmitters is discussed in FSAR Section 3.11.

Interrogatory No. I.G-5

Question:

5. Identify any remedial or corrective measures taken by Westinghouse and/or Applicants to remedy problems identified in IE Information Notice No. 82-11 (April 9, 1982). For each such corrective measure identified, state the expected remedial or corrective effect and whether or not such measures have been tested.

Answer:

No action is required for Seabrook.

Interrogatory No. I.G-6

Question:

6. For any instrument identified in Interrogatory No. 3 above, state all tests, analyses or evaluations Applicants have conducted or intend to conduct on that instrument. Identify and provide copies of all results of such tests, analyses or evaluations.

Answer:

The qualification of the transmitters is discussed in FSAR Section 3.11.

Testing of the installed equipment will be performed as part of the Preoperational Test Program discussed in FSAR Chapter 14 and as part of the Periodic Surveillance Test Program to meet Criteria XI of 10 CFR 50, Appendix B. Copies of the applicable test procedures will be available three months before fuel loading.

Interrogatory No. I.G-7

Question:

7. Identify the function of the wide range pressure instruments in preventing an accident and in mitigating the consequences of any possible accident. State how, if at all, the deficiencies identified in IE Information Notice No. 82-11 will prevent the instruments from functioning properly in preventing an accident or mitigating the consequences of any possible accident.

Answer:

The functions of the wide-range pressure transmitters are discussed in the response to Interrogatory I.G-3. The deficiencies identified in IE Information Notice No. 82-11 do not apply to Seabrook.

Interrogatory No. I.G-8

Question:

8. Identify and provide access to any and all documents referred to or relied on in preparing the response to the interrogatories regarding Contention I.G.

Answer:

The documents referred to or relied on in preparing the response to the interrogatories regarding Contention I.G are identified in the various responses.

These documents will be available for inspection and copying at one or more appropriate places at a time to be mutually agreed upon by counsel for NECNP and the Applicants.

Interrogatory No. I.I-1

Question:

1. What is Applicants' position with respect to NECNP Contention I.I.? State all facts and opinions and identify and provide copies of all documents on which that position is based.

Answer:

It is the Applicants' position that one method (path) of achieving and maintaining a cold shutdown condition will be identified and environmentally qualified as required by IE Bulletin 79-01B, Supplement 3.

Seabrook has made a verbal commitment to the NRC Staff at various meetings that a path of achieving and maintaining a cold shutdown condition will be provided. Identification of required equipment and review of its environmental qualification is presently underway. Formal submittal of this information to the NRC will be made in the near future.

Interrogatory No. I.I-2

Question:

2. Identify all individuals whom Applicants expect to call as witnesses with respect to NECNP Contention I.I. and identify all documents on which Applicants expect to rely at the hearing with respect to this contention.

Answer:

Applicants have not yet determined which, if any, witnesses they will call nor have they identified the documents on which their testimony will be based on for the above contention.

Interrogatory No. I.I-3

Question:

3. Identify and describe all systems required for safe shutdown as required by section 7.4.1. of the Standard Review Plan. Provide the results of Applicants' analysis, as requested by section 7.4.2. of the standard format, and demonstrate how the requirements of the general design criteria and IEEE Std. 279-1971 are satisfied and the extent to which the recommendations of applicable regulatory guides are satisfied, including but not limited to Reg. Guide 1.139, and all applicable NRC regulations, including but not limited to GDC 3, 4, 15, 17, 18, 20-25, 34, 35, and 44 of Appendix B to 10 CFR Part 50.

Answer:

Due to recent design changes implemented to provide a qualified method (path) to achieve and maintain cold shutdown, FSAR 7.4 is presently being rewritten. When completed, the revised Section 7.4 will be submitted to the NRC as an Amendment to the FSAR. This revision will address the concerns raised by NECNP in this interrogatory. It is expected that this revision will be submitted in the near future.

Interrogatory No. I.I-4

Question:

4. Identify and justify any exceptions Applicants claim with regard to their failure to satisfy the applicable regulatory guides, or applicable NRC regulations.

Answer:

It is not anticipated that there will be any exceptions to applicable Regulatory Guides or applicable NRC regulations other than the exception identified in the response to Interrogatory I.I-14.

Interrogatory No. I.I-5

Question:

5. Identify and describe all safety-related structures, systems and components relied upon to remain functional during and following design basis events to assure the capability to shut down the reactor and maintain it in a safe cold shutdown condition.

Answer:

In addition to the systems identified in the response to NECNP Contention I.B.1, Interrogatory 19, additional systems and descriptions of same can be found in FSAR Sections 6.2, 6.3, 6.4, 6.5 and 6.8. These systems are contained within the structures listed in FSAR Table 3.2-1, Sheet 1 of 5.

Interrogatory No. I.I-6

Question:

6. State whether or not all structures, systems and components identified above were environmentally qualified. If any of the structures, systems and components were not environmentally qualified, justify and/or explain why not.

Answer:

The systems and components identified in the response to Interrogatory 5 are all environmentally qualified. The structures which house this equipment are all Seismic Category I structures.

Interrogatory No. I.I-7

Question:

7. Identify and describe all structures, systems and components important to safety which compose the shutdown system, and state whether any of the structures, systems and components are environmentally qualified. If any such structure, system or component is not environmentally qualified, justify and/or explain why not.

Answer:

In addition to the equipment and structures identified in the response to Interrogatory 5, the Reactor Trip System as identified in FSAR Section 7.2

can be considered important to safety. The Reactor Trip System identified above is environmentally qualified.

Interrogatory No. I.I-8

Question:

8. Identify any "limited operator actions" which will be able to restore operability of non-safety grade components or systems in the event Applicants cannot rely on safety-grade systems to take the plant to cold shutdown.

Justify and explain for each such action listed above, how the action will substitute for the use of safety-grade components or systems.

Answer:

The Applicants do not rely on non-safety-grade components or systems to take the plant to cold shutdown.

Interrogatory No. I.I-9

Question:

9. Define a safe hot standby condition for the Seabrook plants. Define a cold shutdown condition for the Seabrook plants.

Answer:

The definitions of hot standby and cold shutdown conditions are identified in Table 1.1 of the

Definition section of the Seabrook Technical Specifications.

Interrogatory No. I.I-10

Question:

10. Justify and explain why Applicants believe it is not necessary to be able to achieve cold shutdown conditions within 36 hours using only safety-grade components or systems.

Answer:

It is the Applicants' position that the Seabrook units will have the capability to achieve cold shutdown conditions within 36 hours using only safety-grade components or systems.

Interrogatory No. I.I-11

Question:

11. Justify and explain why Applicants do not believe they need to identify and environmentally qualify one path to cold shutdown.

Answer:

It is the Applicants' position that one method (path) to achieve and maintain the cold shutdown condition will be identified and environmentally qualified.

Interrogatory No. I.I-12

Question:

12. Explain whether and how long the plant can be maintained in a safe hot standby or hot shutdown condition using the auxiliary feedwater system or residual heat removal system.

Answer:

The Seabrook design permits plant operation at hot shutdown for at least 4 hours followed by cooldown to the conditions permitting operation of the RHR System. Once RHR System operation has been established, the plant can remain in the hot or cold shutdown condition indefinitely.

Interrogatory No. I.I-13

Question:

13. Identify all equipment and operating procedures at Seabrook which bring the plants to a safe hot standby condition and to a safe hot shutdown condition.

Answer:

The essential systems necessary to bring the plant to safe hot standby or safe hot shutdown condition have been identified in the response to NECNP Contention I.B.1, Interrogatory 19.

Interrogatory No. I.I-14

Question:

14. State whether Applicants satisfy any exemption to Regulatory Guide 1.139. If not, explain or justify why Applicants believe they do not need to satisfy any exemption.

If Applicants satisfy any exemption, describe in detail how they satisfy that exemption.

Answer:

The Applicants will satisfy the regulatory positions of Regulatory Guide 1.139, Revision 0, May 1978 with the exception of Item C.2.b.4. Due to operational requirements and equipment location, the frequency of leak testing will be in accordance with ASME Section XI, IWV-3420.

Interrogatory No. I.I-15

Question:

15. Identify and describe the residual heat removal system for the Seabrook plants. Describe the auxiliary feedwater system at the Seabrook plants.

Answer:

The Residual Heat Removal System is described in FSAR Section 5.4.7. The Startup Feedwater System is described in FSAR Section 10.4.12 and Emergency Feedwater System is described in FSAR Section 6.8.

Interrogatory No. I.I-16

Question:

16. Explain how, if at all, Applicants can demonstrate that the plant can be maintained for a longer period of time than 36 hours in a safe hot standby or hot shutdown condition with adequate heat removal via the auxiliary feedwater system or residual heat removal system.

Answer:

The limiting parameter for maintaining a plant in the hot standby or hot shutdown condition using the Emergency Feedwater (EFW) System is the availability of makeup feedwater for the steam generators. The Seabrook design provides a sufficient dedicated safety-grade supply of EFW makeup water to meet the requirements of Regulatory Guide 1.139.

Should the plant decide to remain in the hot standby or hot shutdown condition for an extended period of time, many other sources of makeup water are available such as the condenser hotwells, the demineralized water storage tank and makeup from the water treatment plant.

If the plant is in the hot shutdown condition utilizing the Residual Heat Removal System, steam generator feedwater makeup is not necessary and the plant could remain in this condition indefinitely.

Interrogatory No. I.I-17

Question:

17. Answer each of the questions referred or addressed to Applicants by the NRC Staff in their letters of April 21, 26 and 28, 1982, and identify and provide copies of all documents relating to these answers.

Answer:

The Applicants are unaware of an NRC staff letter dated April 21, 1982 and believe the correct reference should be an April 22, 1982 letter. Responses to all items identified in the April 22, 1982 letter were provided in a letter from PSNH to the NRC (SBN-300) dated July 27, 1982.

Relative to the April 26, 1982 letter, partial response was provided in a PSNH letter to the NRC (SBN-329) dated September 21, 1982. The remaining responses are presently being prepared and will be submitted in the near future.

Responses to the April 28, 1982 letter are presently being prepared. A submittal with these responses will be sent to the NRC in the near future.

Interrogatory No. I.I-18

Question:

18. Identify any selective upgrading of equipment and operating procedures for the shutdown system which Applicants have identified and consider to provide increased operating flexibility and/or increased margins of safety under abnormal plant conditions.

Answer:

In meeting the commitment to provide a qualified method (path) to achieve and maintain the cold shutdown condition, it was found necessary to modify the original design and in some cases provide qualified equipment where the original equipment had not previously required qualification.

The areas where upgrading of equipment was found necessary included the operators and controls for the steam generator atmospheric relief valves and portions of the Emergency Feedwater System. Descriptions of these changes can be found in PSNH letters to the NRC, SBN-300 and SBN-321, dated July 27, 1982 and September 7, 1982, respectively.

Interrogatory No. I.I-19

Question:

19. Identify all equipment necessary to achieve and maintain cold shutdown and their redundant counterparts.

Answer:

Systems and components essential for achieving and maintaining the cold shutdown condition are identified in the response to NECNP Contention I.B.1, Interrogatory 19. Those systems identified all contain redundant equipment necessary to perform the system's function.

Interrogatory No. I.I-20

Question:

20. Identify and provide copies of all test results for testing of all components, systems, or structures necessary to bring the Seabrook plants to a safe shutdown.

Answer:

Test results documenting the qualification of equipment essential to bring the Seabrook plants to a safe shutdown condition are presently being maintained in the documentation files at UER&C's Philadelphia offices. These documents will be available for inspection and copying at one or more appropriate places at a time to be mutually agreed upon by counsel for NECNP and the Applicants.

In addition to the qualification test reports, system and equipment testing will be performed as part of the Initial Test Program outlined in FSAR Section 14.2.

Interrogatory No. I.I-21

Question:

21. Identify all aspects of Reg. Guide 1.139 with which Applicants comply and explain how compliance is achieved.

Answer:

In responding to a Request for Additional Information (RAI), the Applicants will provide a detailed response as to how each item of Branch Technical Position RSB 5-1 will be met. RSB 5-1 contains all the requirements of Regulatory Guide 1.139 and is presently being utilized by the NRC as their licensing basis as identified in NUREG-0800, USNRC Standard Review Plan. The response to this request, RAI 440.133, is presently being prepared and will be submitted to the NRC in the near future.

Interrogatory No. I.I-22

Question:

22. Identify all aspects of Reg. Guide 1.139 with which Applicants do not comply. Explain Applicants' justification for each item of noncompliance. Describe any alternative means employed by Applicants to satisfy General Design Criteria 19 and 34 of Appendix A to 10 CFR Part 50.

Answer:

See response to Interrgatory 14.

Interrogatory No. I.I-23

Question:

23. Have Applicants complied with all the provisions of IE Bulletin No. 79-01B Rev. 3? If not, state which provisions are not complied with, and explain and justify the noncompliance.

Answer:

As indicated in the answer A.3 in Supplement No. 2 to IE Bulletin 79-01B, Seabrook is required to be in compliance with NUREG-0588. Seabrook is in compliance with NUREG-0588.

Interrogatory No. I.I-24

Question:

24. Identify and provide access to any and all documents referred to or relied on in preparing the response to the interrogatories regarding Contention I.I.

Answer:

The documents referred to or relied on in preparing the response to the interrogatories regarding Contention I.I are identified in the various responses. These documents will be available for inspection and copying at one or more appropriate places at a time to be mutually agreed upon by counsel for NECNP and the Applicants.

Interrogatory No. I.L-1

Question:

1. What is the Applicants' position with respect to NECNP Contention I.L.? State all facts and opinions and identify and provide access to all documents on which that position is based.

Answer:

The Applicants disagree with the contention in that qualified valve position limit switches and open/closed indication lights are provided for the Power-Operated Relief Valves (PORVs), RC-PCV-456A and B. Our response to RAI 420.05 describes the indication provided.

Interrogatory No. I.L-2

Question:

2. Identify all individuals whom Applicants expect to call as witnesses with respect to NECNP Contention I.L., and identify all documents on which the Applicants expect to rely at the hearing with respect to this contention.

Answer:

Applicants have not yet determined which, if any, witnesses they will call nor have they identified the documents on which their testimony will be based on for the above contention.

Interrogatory No. I.L-3

Question:

3. Identify the location and function of all pressure-operated relief valves in the reactor coolant pressure boundary, including valves that provide isolation for the system.

Answer:

The ASME Section III safety valves (RC-115, 116 and 117) are the only pressure actuated relief valves in the reactor coolant pressure boundary. These valves are located in the containment, are provided for overpressure protection of the Reactor Coolant System as discussed in FSAR Section 5.2.2, and do not have any isolation valves.

Interrogatory No. I.L-4

Question:

4. Describe the methods employed by Applicants for detecting flow in each of the PORV valves identified in response to Interrogatory No. 3.

Answer:

Flow detection is not provided for each of the PORVs. Direct valve position indication is provided.

Interrogatory No. I.L-5

Question:

5. Describe in detail Applicants' acoustic accelerometer system. Include a discussion of the sensitivity of the apparatus to sound, and of how it

can distinguish the sound of flowing liquid from other sounds.

Answer:

An Accoustic Accelerometer System for detecting flow through the Power-Operated Relief Valves (PORVs) is not provided.

Interrogatory No. I.L-6

Question:

6. Is it Applicants' position that the system or systems employed for detecting PORV flow complies with NUREG-0737, Item II.D.3.? State the reasons for your answer.

Answer:

Yes.

II.D.3 criteria are met by:

1. The operator is provided with an unambiguous open-closed PORV position indication.

2. The PORV position is indicated in the Control Room on the main control board, Section CF, directly above the PORV control switches.

3. The PORV position limit switches are safety grade.

4. The PORV position limit switches are environmentally and seismically qualified.

5. The Human Factors Control Room Design Review will include review of the PORV position indication.

Interrogatory No. I.L-7

Question:

7. Identify and provide access to any and all documents referred to or relied on in preparing the response to the interrogatories regarding Contention I.L.

Answer:

The documents referred to or relied on in preparing the response to the interrogatories regarding Contention I.L are identified in the various responses. These documents will be available for inspection and copying at one or more appropriate places at a time to be mutually agreed upon by counsel for NECNP and the Applicants.

Interrogatory No. I.M-1

Question:

1. What is the Applicants' position with respect to NECNP Contention I.M.? State all facts and opinions and identify and provide access to all documents on which that position is based.

Answer:

NECNP Contention I.M states that the Applicant's Fire Protection System does not meet the requirements of GDC 3 as implemented by the Commission in CLI-80-21 with respect to a list of items. The part of CLI-80-21 which deals with fire protection appears to have several statements which could be interpreted to apply to the implementation of GDC 3. The first is a re-endorsement of a previous statement that:

> "The Commission endorses the staff's position that no one level of defense-in-depth can be made invulnerable. Strengthening of one of the levels can compensate in some measure for reduced safety margins in the others."

The Applicants concur with that statement and have designed its fire protection program using the statement as guidance.

A second Commission request is directed to the NRC staff, and requests that fire protection research be continued, that a schedule be developed, and that periodic updates be given to the Commission.

As it is directed at the NRC staff, it has nothing to do with Seabrook and we have no position on it.

The third section deals with the development and promulgation of a proposed rule concerning fire protection, its Appendix R, and its application to nuclear power plants docketed for Construction Permit prior to July 1, 1976. It states that the combination of the requirements contained in that document combined with the guidance contained in Appendix A to BTP 9.5-1 define the essential elements for an acceptable fire protection program. It further states that similar acceptable guidance is provided in BTP 9.5-1 for nuclear power plants docketed for a Construction Permit after July 1, 1976.

Seabrook was docketed for a Construction Permit on July 9, 1973. Therefore, the progress of the proposed rule was followed with great interest because of the potential affect on the fire protection program. The revised Section 50.48 as finally published states that each operating nuclear power plant shall have a fire protection plan that satisfies Criterion 3 of Appendix A to 10 CFR Part 50. It then states general criteria for such a plan. Seabrook Station is in the process of developing a plan which complies with these criteria. This plan is addressed in the documents titled, "Seabrook Station Fire Protection Program, Evaluation and Comparison to BTP APCSB 9.5-1, Appendix A, Revision 1," and in the Seabrook Station FSAR.

The final published revision to Appendix R to 10 CFR Part 50 makes no mention of Construction Permit docket dates. Instead, it is titled "Appendix R - Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979." Therefore, it does not apply to Seabrook. Nevertheless, in compliance with an NRC request, we compared our fire protection program with Appendix R. The results of that review can be found in a document titled "Comparison to Appendix R to 10 CFR 50, Fire Protection Program for Operating Nuclear Plants," attached as Exhibit 1 to SBN-248.

As noted in the above Commission position, Appendix A to BTB APCSB 9.5-1 contains guidance to be used in the development of an acceptable fire protection program. Seabrook has used that guidance in the development of its fire protection.

The fourth section addresses the schedule for implementation of Fire Protection System modifications on operating nuclear power plants. It has nothing to do with Seabrook, and we have no position on it.

The balance of the document deals with Commission positions on the methods of analysis used by the staff during its reviews of the risk associated with allowing certain nuclear power plants to continue operation while Fire Protection System modifications were being completed. This does not apply to Seabrook, and we have no position on it.

The documents upon which our position is based are:

1. CLI-80-21,
2. 10 CFR 50.48 and Appendix R to 10 CFR Part 50, Appendix A to BTP 9.5-1,
3. BTP APCSB 9.5-1, Appendix A,
4. Revision 1 of Seabrook Station Fire Protection Program, Evaluation and Comparison to BTP APCSB 9.5-1, Appendix A,
5. Seabrook Station Fire Protection of Safe Shutdown Capability (10 CFR 50, Appendix R),
6. Comparison to Appendix R to 10 CFR 50; Fire Protection Program for Operating Nuclear Power Plants.

Documents 1, 2 and 3 are NRC documents. Documents 4, 5 and 6 can be found at the Offices of United Engineers and Contractors, Philadelphia.

Interrogatory No. I.M-2

Question:

2. Identify all individuals whom Applicants expect to call as witnesses with respect to NECNP Contention I.M., and identify all documents on which the Applicants expect to rely at the hearing with respect to this contention.

Answer:

Applicants have not yet determined which, if any, witnesses they will call nor have they identified the documents on which their testimony will be based on for the above contention.

Interrogatory No. I.M-3

Question:

3. Is it Applicants' position that under the Commission's order in CLI-80-21, 11 NRC 707, 718 (1980), and Commission regulations, the revision of 10 CFR 50.48 and Appendix R to 10 CFR Part 50, published November 19, 1980 at 45 FR 76602 and Appendix A to Branch Technical Position 9.5-1 of the Standard Review Plan constitute Commission requirements for compliance by the Seabrook plant with General Design Criterion 3 of Appendix A to Part 50? Please state your reasons if you answer this question in the negative.

Answer:

It is Seabrook's position that the revision of 10 CFR 50.48, paragraphs (a) and (e) constitutes a Commission requirement for compliance by the Seabrook plant with GDC 3 of Appendix A to Part 50.

It is Seabrook's position that the revision of 10 CFR 50.48, paragraphs (b), (c) and (d) do not constitute a commission requirement for compliance by the Seabrook plant.

The reasons for this position are that 10 CFR 50.48(b), (c) and (d) reference the application and schedule for application of Appendix R to Part 50 to nuclear power plants licensed to operate prior to January 1, 1979. As Seabrook was not licensed to operate prior to January 1979, these paragraphs do not apply.

It is Seabrook's position that, under the Commission's order in CLI-80-21 and Commission regulations, Appendix R to 10 CFR Part 50, published November 19, 1980 does not constitute a Commission requirement for compliance by the Seabrook plant with GDC 3 of Appendix A to Part 50. The reasons for this position are:

1. The Commission's statement in CLI-80-21 dealt with a proposed rule and Appendix R, which they felt at that time would be tied to a Construction Permit docket date.

2. The final published Appendix R was revised to apply only to plants operating before January 1, 1979.

It is Seabrook's position that Appendix A to BTP APCSB 9.5-1 does not constitute a Commission requirement for compliance by the Seabrook plant with GDC 3 of Appendix A to Part 50. It supplies guidance only. The reasons for this position are:

1. CLI-80-21 refers to the guidance contained in Appendix A to BTP 9.5-1 as opposed to the requirements of a rule.
2. The cover letters sent out with Appendix A to BTP APCSB 9.5-1 refer to the guidelines in the position itself. They also state further that the Appendix presents "certain acceptable alternatives" to those guidelines in the BTP.

It is clear to us that Appendix A to BTP APCSB 9.5-1 is a set of guidelines, not a requirement.

<u>Interrogatory No. I.M-4</u>

<u>Question</u>:

4. Identify all provisions of 10 CFR 50.48 and Appendix R with which Applicants are not in compliance, and explain the reasons and justifications for noncompliance. For each of these, describe what

measures must be taken to bring them into compliance, and state why they have not been implemented.

Answer:

In its response to Contention I.M, Interrogatory 3, Seabrook stated its position that the only two paragraphs of 10 CFR 50.48 that apply to us are paragraphs (a) and (e). Paragraph (a) requires that an operating nuclear power plant have a fire protection plan that addresses many items. Following the normal course of events, this plan is currently under development. As previously committed to the NRC staff, the plan will be in place and in agreement with these guidelines by the time the plant is operating.

In its response to Contention I.M, Interrogatories 1 and 3, Seabrook stated its position that Appendix R does not apply to Seabrook. Nevertheless, we did compare our fire protection program with Appendix R. The results of that review are in Reference 6 under Contention I.M, Interrogatory 1. The document identifies those sections of Appendix R with which we do not comply. There are no measures that must be taken to bring them into compliance, as Appendix R does not apply to Seabrook Station. We are, however, working with the NRC staff to develop a fire protection program which addresses the concerns expressed in Appendix R to Part 50 as well as the guidelines of Appendix A to BTP APCSB 9.5-1.

Interrogatory No. I.M-5

Question:

5. Identify all provisions of Appendix A to BTP 9.5-1 with which Applicants are not in compliance, and explain the reasons and justifications for noncompliance. For each of these, describe what measures must be taken to bring them into compliance, and state why they have not been implemented.

Answer:

The document titled "Seabrook Station Fire Protection Program, Evaluation and Comparison to BTP APCSB 9.5-1, Appendix A" identifies and provides justification for those guidelines in BTP APCSB 9.5-1 which the Seabrook fire protection program does not meet. As these are guidelines and there are acceptable alternatives, there are no measures that must be taken to bring out program into compliance.

Interrogatory No. I.M-6

Question:

6. Provide access to any and all documents generated in preparation for or as a result of the meeting on fire protection between PSNH and the NRC Staff on March 10, 1982, relating to a comparison of PSNH's fire protection system and 10 CFR Part 50, Appendix R.

Answer:

The documents are composed of correspondence, internal review documents, procedures for review, notes of meetings, computer printouts, results of analyses, drawings and schematics. They are in files of United Engineers and Constructors, Philadelphia.

Interrogatory No. I.M-7

Question:

7. Provide access to the revision of the "Seabrook Station Fire Protection System Evaluation and Comparison to Branch Technical Position 9.5-1, Appendix A", described in a letter dated May 15, 1981, from John DeVincentis to Robert L. Tedesco.

Answer:

This document is available for review at the Philadelphia offices of UE&C.

Interrogatory No. I.M-8

Question:

8. Identify and provide access to any and all documents referred to or relied on in preparing the response to the interrogatories regarding Contention I.M.

Answer:

The documents referred to or relied on are:

1. CLI-80-21,

2. 10 CFR 50.48 and Appendix R to Part 50,

3. NECNP Contention I.M,

4. Seabrook Station Fire Protection Program Evaluation and Comparison to BTP APCSB 9.5-1, Appendix A (Revision 1, April 1982),

5. Seabrook Station Fire Protection and Safe Shutdown Capability (10 CFR 50, Appendix R),

6. BTP APCSB 9.5-1, Appendix A.

7. Comparison to Appendix R to 10 CFR 50; Fire Protection Program for Operating Nuclear Power Plants (Exhibit 1 to SBN-248, letter from J. DeVincentis for F. J. Meraglia, dated April 1, 1980).

Documents 4, 5 and 7 can be found at the Offices of UE&C, Philadelphia.

Interrogatory No. I.N-1

Question:

1. What is Applicants' position with respect to NECNP Contention I.N.? State all facts and opinions and identify and provide access to all documents which support that position.

Answer:

The original design of the Solid Waste Handling System utilized urea formaldehyde (UF) for the solidification or immobilization of liquid and resin waste streams produced during normal reactor operation including anticipated operational occurrences. However, over the past several years questions have been raised concerning the capability of UF to consistently produce solidified waste forms which could meet burial ground disposal criteria on the amount of free water allowed with the solidified waste. In January 1982, the Barnwell Disposal Site, South Carolina Radioactive Material License No. 097, Amendment No. 34 removed UF from their list of acceptable solidification media. In addition, the State of Washington Radioactive Materials License No. WN-I019-2, Amendment No. 15 (October 1982) issued to U.S. Ecology for the Richland, Washington disposal site also deleted UF from their list of allowable solidification media. In forseeing the possibility approximately two years ago that UF would not be a viable solidification binder when Seabrook Station became operational, it was decided to replace the original UF equipment.

A revised equipment specification (UE&C No. 9763-006-267-3) has been prepared and issued for the purpose of evaluating potentially new solidification systems, including possible volume reduction equipment for minimizing the volume of radioactive waste requiring off-site disposal. At this time the selection of replacement equipment for the original UF system design has not been made and details concerning its design not available. However, it is the intent of the Applicant to have available for use at the time of Unit 1 fuel load an installed Solid Waste Disposal System meeting all applicable regulatory requirements, or a contracted mobile solidification system for the proper handling and packaging of solid waste. Design details concerning the final Solid Waste Disposal System will be forwarded to the NRC when available.

Interrogatory No. I.N-2

Question:

2. Identify all individuals whom Applicants expect to call as witnesses with respect to NECNP Contention I.N., and identify all documents on which Applicants expect to rely at the hearing respecting this contention.

Answer:

Applicants have not yet determined which, if any, witnesses they will call nor have they identified the documents on which their testimony will be based on for the above contention.

Interrogatory No. I.N-3

Question:

3. Describe the current design of Applicants' solid waste management system. For each component of that system, describe the approximate state of completion of construction and testing, or the expected date of completion.

Answer:

As indicated in response to Interrogatory I.N-1, the original Solid Waste Management System (urea formaldehyde solidification) will be replaced by a new irplant system design, or contracted Mobile Waste Handling System which will meet all applicable regulations. The originally selected UF system has not been installed in the Waste Processing Building. The areas in the Waste Processing Building which were originally intended for the placement of the UF System have been left open in order to allow flexibility in design when a replacement system is selected.

Interrogatory No. I.N-4

Question:

4. Explain whether the original waste solidification equipment purchased for the Seabrook Station has been replaced or substitution equipment is being planned. State all reasons for the replacement or substitution of the original waste solidification equipment.

Answer:

The current status and reason for the decision to replace the original waste solidification equipment purchased for Seabrook Station is given in response to Interrogatory I.N-1.

Interrogatory No. I.N-5

Question:

5. Explain any special problems which might be caused by the sharing of one solid waste management system by the two Seabrook units.

Answer:

No special problems have been identified concerning the sharing of a common Solid Waste Management System for the handling of radioactive solid waste produced during normal reactor operations, including anticipated operational occurrences, from both Seabrook units.

Interrogatory No. I.N-6

Question:

6. Describe what if any components of the solid waste management system are nuclear safety-grade equipment. For those components of the system which are not nuclear safety-grade, explain why there is no need to provide nuclear safety-grade components

Answer:

Though a replacement Solid Waste Management System for Seabrook Station has not as yet been selected, it is not intended that any component of the replacement system be classified as nuclear safety-grade equipment since that classification is not required by any regulation or design standard due to the limited reactor safety impact associated with these systems.

Interrogatory No. I.N-7

Question:

7. Explain how Applicants have calculated the annual volume of solid waste anticipated. Explain how this estimated amount compares with the amount produced by other nuclear plants of a size comparable to Seabrook.

Answer:

The information pertaining to the annual volume of solid waste anticipated was provided in response to Seabrook RAI 460.31 which was transmitted to the NRC in a letter (SBN-254) dated April 9, 1982.

Interrogatory No. I.N-8

Question:

8. Identify all figures, descriptions, and calculations which are as of yet not completed by Applicants for insertion in the portion of the FSAR relating to the solid waste management system.

Answer:

When a replacement Solid Waste Management System is selected, Section 11.4 of the Seabrook FSAR will be rewritten in its entirety. Revised system descriptions, including operational characteristics, calculations of expected final waste volumes requiring off-site disposal, as well as new solid waste management P&I diagrams will be included in Section 11.4.

Interrogatory No. I.N-9

Question:

9. Describe how, if at all, the solid waste management system has been designed to handle radioactive solid waste produced under accident conditions.

Answer:

Since the final selection and design of a new Solid Waste Management System has not been made, assessments of its capabilities to handle waste produced under various accident conditions has not been made. It should be noted that the design basis for the Solid Waste System does not include requirements to consider waste produced beyond normal reactor operations including anticipated operational occurrences.

Interrogatory No. I.N-10

Question:

10. Describe the radwaste system and all components which Applicants believe must comply with GDC 60, and Criterion 1 and 2 of Appendix A to 10 CFR Part 50.

Answer:

As described in the response to Interrogatory I.N-1, components of a new Solid Waste Management System have not been selected at this time. However, GDC 60 will be complied with for solid waste handling before fuel load of Unit 1.

Criteria 1 and 2 of Appendix A to 10 CFR Part 50 are concerned with structures, systems and components important to safety and that they are designed, fabricated, erected and tested to quality standards commensurate with the importance of the safety functions to be performed. Seabrook's commitment for conformance with the appropriate portions of Regulatory Guide 1.143 for Solid Radwaste Systems provide assurance that Criteria 1 and 2 will be adequately met with respect to solid radwaste management.

Interrogatory No. I.N-11

Question:

11. Describe the level of radiation which Applicants anticipate operating and maintenance personnel will be exposed to from the radioactive waste management system, and how Applicants determined that level is as low as is reasonably achievable.

Answer:

Present plans require that the replacement solidification and associated process equipment that would be permanently installed at Seabrook be remotely operated to ensure that radiation exposures are kept as low as reasonably achievable. Operating personnel exposures during normal operation is expected to be less than 2.5 mrem/hour due to equipment layout and shielding requirements.

In keeping with the guidance of Section 4.1 of Regulatory Guide 1.143, the design of the replacement Waste Management Systems will require that appropriate provisions be included to minimize radiation exposures to maintenance personnel during periods of system maintenance and repair.

Interrogatory No. I.N-12

Question:

12. Describe the level of radiation which Applicants anticipate the general public will be exposed to from the radioactive waste management system, and how the design and construction of the system provides assurance that such radiation exposures are as low as is reasonably achievable.

Answer:

Due to the location of the shielded area within the Waste Processing Building (WPB) set aside for the Solid Radwaste System components and the distance between the WPB and the site boundary (approximately 3000 feet), it is not anticipated that the general public will be exposed to any radiation above a small fraction of the annual dose objectives of ALARA as detailed in Appendix I to 10 CFR Part 50 for liquid and gaseous effluents.

Interrogatory No. I.N-13

Question:

13. How, if at all, have Applicants designed the solid waste management system to standards which enhance system reliability, operability, and availability?

Answer:

The design of the replacement Solid Waste Management System will conform with the appropriate portions of the ANSI/ANS 55.1 (1979) standard, "Solid Radioactive Waste Processing Systems for Light-Water Reactor Plants." This will provide assurance that the installed Solid Waste System will meet performance objectives concerning reliability, operability and availability.

Interrogatory No. I.N-14

Question:

14. Describe the solid waste management systems at McGuire, Comanche Peak, and Byron/Braidwood.

Describe all similarities and all differences between the systems at the three plants listed above, and the system at the Seabrook station.

Answer:

We have no knowledge of the specific designs of the Solid Waste Management Systems at McGuire, Comanche Peak and the Byron/Braidwood Stations.

Interrogatory No. I.N-15

Question:

15. State whether or not the solid radwaste system will be designed and tested to all requirements set forth in the codes and standards contained in Table 1 supplementing the regulatory positions 3.1.2 and 4 of Reg. Guide 1.143.

Answer:

It is anticipated that the replacement Solid Radwaste Systems will be designed and tested as appropriate in accordance with the requirements set forth in Table 1 (Equipment Codes) to Regulatory Guide 1.143.

Interrogatory No. I.N-16

Question:

16. State whether or not the foundations and adjacent walls of structures that house the solid radwaste system are designed to the seismic criteria given in regulatory position 5 of Reg. Guide 1.143.

Answer:

Foundations and adjacent walls of structures that house the Radwaste System are in compliance with the requirements stated in Position 5 of Regulatory Guide 1.143, "Seismic Design for Radwaste Management Systems and Structures Housing Radwaste Management Systems."

Interrogatory No. I.N-17

Question:

17. Explain how, if at all, radioactive waste management structures, systems and components are designed to control leakage and to facilitate access, operation, inspection, testing and maintenance in order to maintain radiation exposures to operating and maintenance personnel as low as is reasonably achievable.

Answer:

It is planned that the design of any inplant replacement Solid Waste System will include design details to ensure that the Criteria of 4.1 of Regulatory Guide 1.143 are implemented.

Interrogatory No. I.N-18

Question:

18. Explain Applicants' quality assurance procedures for their radioactive waste management system.

Answer:

With respect to the Solid Waste Management System for future installation at Seabrook Station, the following list of UE&C quality assurance procedures will provide assurance that the intent of Section 6 of Regulatory Guide 1.143 is met.

9763-QAS-2	Quality Assurance Administrative and Systems Requirements
9763-WS-2	Welding and Nondestructive Examination for Pressure Components and Power Piping
9763-MPS-2	Material and Processing Requirements for Non-Nuclear Components

Interrogatory No. I.N-19

Question:

19. State whether or not pressure-retaining components of process systems use welded construction to the maximum practicable extent.

Answer:

It is our intent to require that pressure-retaining components that would be part of the replacement Solid Rewaste System use welded construction to the maximum practicable extent in conformance with Section 4.3 of Regulatory Guide 1.143.

Interrogatory No. I.N-20

Question:

20. State which piping systems are hydrostatically tested in accordance with applicable ASME or ANSI codes.

Answer:

In accordance with Section 4.4 of Regulatory Guide 1.143, it will be required that all appropriate piping systems that are part of the Solid Radwaste System be tested in accordance with applicable ASME or ANSI codes. Specific identification of these piping systems cannot be made until a replacement Solid Radwaste System is selected.

Interrogatory No. I.N-21

Question:

21. Describe any testing provisions incorporated into the radioactive waste management system to enable periodic evaluation of the operability and required functional performance of active components of the system.

Answer:

It is our intent to require, where appropriate, that the design of the replacement inplant Solid Radwaste System incorporate testing provisions to enable periodic evaluation of the operability and required functional performance of the systems active components in accordance with Section 4.5 of Regulatory Guide 1.143. Specific details of testing provisions cannot be made until the final design of the Solid Waste System is available.

Interrogatory No. I.N-22

Question:

22. Identify and provide access to any and all documents referred to or relied on in preparing the response to the interrogatories regarding Contention I.N.

Answer:

The following documents where referred to or relied on in preparing responses to interrogatories regarding Contention I.N:

1. UE&C Specification No. 9763-006-267-3, "Radwaste Solidification and Handling Equipment."
2. UE&C Specification No. 9763-QAS-2, "Quality Assurance Administrative and Systems Requirements."

3. UE&C Specification No. 9763-WS-2, "Welding and Nondestructive Examination for Pressure Components and Power Piping."

4. UE&C Specification No. 9763-MPS-1, "Material and Processing Requirements for Non-Nuclear Components."

5. Seabrook Letter (SNB-253) to F. J. Maraglia, USNRC, dated April 9, 1982: "Responses to 460 Series RAI: (ETSB)." (See response to RAI 460.31.)

Interrogatory No. I.U-1

Question:

1. What is the Applicants' position with respect to NECNP Contention I.U.? State all facts and opinions and identify and provide access to all documents on which that pos tion is based.

Answer:

General Design Criterion 4 of Appendix A of 10 CFR Part 50 states, "Structures, systems and components important to safety shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping and discharing fluids that may result from equipment failures . . ." (Reference 1). Regulatory Guide 1.115 states that the NRC staff considers a hazard rate of 10^{-7} per year from a single event to be an acceptable risk rate (The risk rate in this context consists of the frequency of events which lead to off-site radiological consequences in excess of 10 CFR Part 100 exposure guidelines.), and that one way to ensure that this goal is achieved is to have the probability of damage of all essential systems located in the direct zone be 10^{-3} (Reference 2). This value of P_2 P_3, coupled with a probability of a turbine producing missiles of $P_1 = 10^{-4}$/year, yields $P_4 = P_1 P_2 P_3 = 10^{-7}$/year, the desired level.

Although we concur that the overall objective of a 10^{-7}/year risk rate is reasonable, we believe that it is very difficult to produce a realistic analytical model which is free from uncertainty and possible non-conservative modeling. This type of situation occurs frequently in the evaluation of external hazards and the NRC position of this point is given in SRO 2.2.3 (Reference 3). It is stated that a risk rate "of approximately 10^{-6}/year is acceptable if, when combined with reasonable qualitative arguments, the realistic probability can be shown to be lower."

We contend that the analytical model and the specific assumptions used in the evaluation are conservative, do in fact yield a risk rate on the close order of 10^{-6}/year or less. Hence, the intent of USNRC Regulatory Guide 1.115 and SRP 3.5.1.3 (Reference 4) is satisfied.

Although portions of the containment of one unit may be struck by low trajectory missiles ejected from the turbine located at the other unit, the majority of such missiles will not enter the containment or cause secondary missiles to enter the containment. Hence, the probability of unacceptable damage to the containment from low trajectory missiles is less than 10^{-7} per year.

References for responses to interrogatories regarding this contention are included as part of the response to Interrogatory I.U-17.

Interrogatory No. I.U-2

Question:

2. Identify all individuals whom Applicants expect to call as witnesses with respect to NECNP Contention I.U., and identify all documents on which the Applicants expect to rely at the hearing with respect to this contention.

Answer:

Applicants have not yet determined which, if any, witnesses they will call nor have they identified the documents on which their testimony will be based on for the above contention.

Interrogatory No. I.U-3

Question:

3. Describe your analysis of the total probability of unacceptable damage to safety-related systems, components, and structures from low-trajectory missiles, including all formulas employed for this analysis. (Footnote omitted.)

Answer:

In this context, the term "safety-related" refers to essential systems as defined in Reference (4):

". . . those structures, systems and components necessary to ensure:

- The integrity of the reactor coolant pressure boundary.
- The capability to prevent accidents that could result in potential off-site exposures that are comparable to the exposure guidelines of 10 CFR Part 100, 'Reactor Site Criteria.'
- The capability to shutdown the reactor and maintain it in a cold shutdown condition.

The details of the analytical model are presented in FSAR Section 3.5.1.3, Pages 3.5-7 to 3.5-17.

Interrogatory No. I.U-4

Question:

4. Describe alternative analyses to determine the total probability of unacceptable damage to safety-related systems, components, and structures from low-trajectory missiles, of which you are aware but did not use, including alternative values or formulas for P_1 or the rate of turbine failures in events per year; P_2 or the conditional probability that a missile will strike a specified target; P_3 or the conditional probability that a missile will cause damage to the target that may lead to unacceptable consequences, if the target is hit; and P_4 or the probability in events per year of occurrence of unacceptable damage from a turbine failure.

Answer:

A large number of techniques have been developed to evaluate the effects of turbine missile generation. References 4, 6, 7, 8 and 9 provide descriptions of a few of these methods. These range from simple hand calculations to complex Monte Carlo codes. The present method performs an averaging procedure over plausible initial conditions as described in the response to Question 3 and was selected because its computational errors are very small.

Alternate formulae or assumptions are described below:

P_1 - The values of P1 used in this analysis are the values recommended by SRP 3.5.1.3:

$P1 = 6 \times 10^{-5}$/year normal overspeed

$P1 = 4 \times 10^{-5}$/year runaway overspeed

These values are based upon work of S. Bush. In 1978, Bush performed a reassessment of this evaluation (Reference 10) which tended to confirm an expected value of about 10^{-4}/year with an error factor of about 3. In Reference 10, work performed by Reference 11 is cited. This work, performed by the turbine manufacturer (GE) yields:

$P_1 = 8.7 \times 10^{-9}$/year normal overspeed

$P_1 = 5.0 \times 10^{-9}$/year runaway overspeed

If these estimates are fairly accurate, turbine missile production would not be a design basis event. The turbine manufacturers point out that the data base for turbine missiles - perhaps 7 relevant failures in 10^5 years of turbine operation - is far too small for a realistic assessment and that some of these failures are not really relevant to their design and fabrication processes.

<u>P_2</u> - The assessment of P_2 depends upon certain assumptions in the initial conditions and missile

dynamics. The assumption of no air resistance is conservative. Other assumptions are:

Uniform energy distribution for missiles - Reference 8 considers three sets of energy distributions - uniform in energy, uniform in velocity and a single ejection energy. A single ejection velocity is not physically plausible and a distribution uniform in energy produces higher estimates of PSD (Probability of Strike-and-Damage, equivalent to P_2 P_3 in the present case) than the uniform velocity distribution case.

Gaussion distribution for azimuthal angle - Reference 8 demonstrates the effect of using a Gaussion distribution in azimuthal angle, uniform distribution in polar angle as compared with the model used in the present analysis (uniform in solid angle). Reference 8 suggested standard deviations in the range of 1/3 to 1 times the maximu azimuthal angle. Use of = 1/3 (25°) for the end discs would tend to reduce the probability of a containment strike by about a factor of 20. The probability of impact on the Unit 2 control building would decrease by a factor of about 2.8 and the probability of an impact on the Unit 1 control building would drop by a factor of 9. The use of = 1 (25°) would not greatly change the results of the analysis.

It can be seen that use of reasonable alternate assumptions could result in estimates of P_2 lower by a factor of 10 or 20.

$\underline{P_3}$ - The modified NDRC equation (Reference 5) is used to calculate the effects of missile impacts on concrete structures. The prediction depth provided by this equation was found to be in good agreement with a series of tests conducted by A. Stephenson (Reference 11) but produced poor predictions of scabbing. That is, the formula uniformly over-estimated the thickness of concrete required to prevent scabbing. Also, the amount of penetration (and backface effects) were found to be less with a missile impact at oblique angles. The number of these tests was limited and may not be applicable to irregularly chopped disc fragments (Reference 2) but do provide some indication that the use of the modified NDRC equation to predict spall is conservative. Finally, the assumption that secondary missiles entering a compartment containing safety-related equipment constitutes unacceptable damage should not be confused with the assumption that this will produce unacceptable consequences. In many such cases, none of the safety objectives given in the response to question 3 will be violated.

$\underline{P_4}$ - The evaluation of P_4 depends upon the parameters P_1, P_2 and P_3 which have been estimated using a conservative set of assumptions and analytical techniques. Accordingly, P_4 provides a conservative (i.e., upper bound) estimate of the probability of unacceptable damage due to low trajectory turbine missile strike.

Interrogatory No. I.U-5

Question:

5. Define "unacceptable consequence" and "unacceptable damage" as those two terms are used in your analysis.

Answer:

The term "unacceptable consequences" refers to damage to safety-related equipment (as defined in Question 3) which

1. Causes off-site radiological exposures comparable to 10 CFR Part 100 limits.
2. Prevents the reactor from being placed and maintained in a cold shutdown condition.

3. Impairs the integrity of the reactor coolant pressure boundary.

"Unacceptable damage" consists of damage to safety-related structures or equipment which may (or may not) lead to unacceptable consequences.

Interrogatory No. I.U-6

Question:

6. Describe the categories of turbine failure you considered in your analysis. If you did not consider stress corrosion as a possible cause of turbine failure, explain why you did not.

Answer:

For the purpose of this analysis, two classes of failures were postulated, per the recommendations of Standard Review Plan 3.5.1.3. These are normal (design) overspeed and runaway (destructive) overspeed. Probability estimates for these conditions were obtained from work performed by S. Bush. Historical failures caused by stress corrosion were included in the data base which was used to estimate the probability of turbine missiles produced by a normal overspeed condition. Hence, stress corrosion failures were included implicitly in the analysis.

Interrogatory No. I.U-7

Question:

7. Identify all safety-related systems, components, and structures which could be directly or indirectly damaged by turbine missiles.

Answer:

Safety-related systems are located inside the following Category 1 structures:

1. Containment,
2. Control Building,
3. Diesel Generator Building,
4. Condensate Storage Tank,
5. Emergency Feedwater Pump Building,
6. Equipment Vaults,
7. Service Water Pump House,
8. Primary Auxiliary Building, and
9. Fueld Storage Building.

Of these structures, numbers 1-7 are in the direct zone; all structures are subject to high trajectory missile strikes. Reference 13 contains a list of all safety-related equipment inside Category 1 structures and FSAR Section 3.2 (Reference 14) contains a breakdown by system.

Interrogatory No. I.U-8

Question:

8. Identify all systems for which you considered or provided protection from turbine missiles. Identify for each and every system identified above the form of protection provided. Identify all possible risks to those systems from turbine missiles, including direct effects and indirect effects.

Answer:

Protection of safety-related components, systems and structures is provided for the plant as a whole by certain design and layout procedures including:

1. Locating safety-related structures outside the direct zone (-5^0 to 5^0) of interior turbine discs.
2. External missile barriers which preclude penetration of all but the most energetic missiles.
3. Redundancy of key safety-related systems (Note - credit is taken for redundancy only in those cases in which it can be demonstrated that at least one safety train will not be damaged by primary or secondary missiles).
4. A plant layout which ensures that most low trajectory missile strike will impact safety-

related barriers obliquely, thus, reducing the probability of primary missile penetration and secondary missile production. This layout is also such that missiles impacting the face of one safety-related structure will be deflected away from the other safety-related structures.

With the exception of plant layout, this protection was provided to ensure design adequacy against other external plant hazards such as tornado wind loading, missiles, etc.

The systems protected by the above methods have been defined in the response to Question 7.

Interrogatory No. I.U-9

Question:

9. State the probability of damage (in events per year) by low-trajectory turbine missiles for design overspeed failures. Describe the ways in which the design overspeed failure rate may be reduced, including improvements in turbine design, etc. Have any of these methods to reduce the design overspeed failure rate been employed at Seabrook? If so, describe them.

Answer:

The probability of unacceptable damage (as defined in the response to Question 5) due to low trajectory missile strikes by a turbine failure at design overspeed is estimated as:

$P_4 = 4.2 \times 10^{-7}$/year Unit 1

$P_4 = 5.2 \times 10^{-7}$/year Unit 2

$P_4 = 1.8 \times 10^{-9}$/year Service Water Pump House

These values are upper bound estimates of the probability of unacceptable consequences. Realistic values would be somewhat lower.

The turbine generator and its overspeed protection system for Seabrook Units were designed and fabricated in accordance with current criteria and procedures. For additional discussion, see response to Question 13.

Interrogatory No. I.U-10

Question:

10. State the probability of damage (events per year) by low-trajectory turbine missiles due to destructive overspeed failures. Describe the ways in which the destructive overspeed failure rate may be reduced. Have any of these methods to reduce the overspeed failure rate been employed at Seabrook? If so, describe them and how they were employed.

Answer:

The probability of unacceptable damage (as defined in response to Question 5) due to low trajectory missile strikes by a turbine failure at destructive overspeed is estimated as:

$P_4 = 3.3 \times 10^{-7}$/year Unit 1

$P_4 = 4.1 \times 10^{-7}$/year Unit 2

$P_4 = 1.2 \times 10^{-9}$/year Service Water Pump House

These values are upper bound estimates of the probability of unacceptable consequences. Realistic values could be somewhat lower.

The overspeed failure probability will be reduced by the overspeed protection systems furnished for this plant. The turbine is furnished with both automatic and manual systems to trip the turbine before overspeed occurs. Refer to response to Question 13.

Interrogatory No. I.U-11

Question:

11. For the following methods of protecting essential systems, components and structures from low-trajectory missiles, state whether or not they have been employed at Seabrook, and why, if they were considered and not employed, they were rejected:

a. Exclusion of essential systems from the low-trajectory hazard zone;

b. Placement of the turbines far enough from the essential system that an acceptable probability of the turbine missile striking the system has been calculated;

c. Placement of essential systems;

d. Separation of redundant equipment;

e. Strike, damage or other analyses of turbine valve reliability;

f. Barriers.

Answer:

The layout and design of the safety-related structures at Seabrook were set by limitations of space, economics and consideration of the effects of turbine missiles. Because the probability of unacceptable consequences resulting from a turbine missile strike is very low for the given configuration, no serious consideration was given to adding additional protection. Specific comments on the protection schemes listed above follows:

a. Safety-related structures lie outside the direct zone of the interior discs and three of the end discs. This placement results in a major reduction in strike probability over the older "in-line" configuration. Space limitations and economics precluded placement of all safety-related structures outside the direct zone.

b. Increasing the distance between the turbine and safety-related structure to reduce the overall probability of a missile strike is not possible with the Seabrook site and would result in a major increase in costs of it could be employed. Because the missiles contributing to P_4 are very energetic, the drop-off of P_4 with distance is relatively slow (i.e., P_4 is approximately proportioned to the inverse square of the distance).

c. With the present configuration, a considerable portion of the safety-related structures lies in the geometric shadow of other structures, thus limiting the area over which a missile strike can take place. Placement of safety-systems within buildings will not reduce the probability of unacceptable damage since no credit is taken for the conditional probability that missiles (primary or secondary) entering a structure will inflict unacceptable consequences.

d. Separation of redundant equipment is required for protection against other hazards, especially fire. This separation will actually result in additional protection against turbine missiles but no credit is taken for redundancy unless it is not possible for a turbine missile or secondary missile to damage both trains. In general, this will occur only if a missile barrier separates trains.

e. The Applicants have not performed a detailed turbine valve analysis. Physical separation and redundancy of the turbine valves assures reliability. Turbine valves are not safety-class components.

f. The barriers used to protect safety-related structures from tornado missiles play a major role in protecting these structures from turbine missiles. They prevent unacceptable damage from occurring because of impacts by low energy missiles. The most energetic missiles can, on the other hand, produce secondary missiles in substantial thicknesses of concrete. For example, the modified NDRC equations predict that an 8200 lb. missile traveling at 650 ft./sec. will perforate over 100 inches of concrete and will cause scabbing in a barrier of thickness equal to 130 inches. Protection against such missiles is not predictable. Not withstanding the uncertainty involved in use of the modified NDRC equations, it is desirable to provide sufficient barrier thickness to limit the estimated value of P4 to an acceptable range as was done in the present case.

Interrogatory No. I.U-12

Question:

12. How if at all does the placement of two nuclear units on the Seabrook site affect the total probability of unacceptable damage to safety-related systems, components and structures from low-trajectory missiles?

Answer:

As can be seen from Figure 3.5-1 and Table 3.5-9, the bulk of the probability of unacceptable damage to safety-related structures at Unit 1 is due to low trajectory missiles originating from the turbine at Unit 2 and vice versa.

The overall effect of a turbine failure at one unit plus a safety system failure at the second unit will generally be less severe than an event consisting of a turbine failure and safety system failure both occurring at the same time.

Interrogatory No. I.U-13

Question:

13. Describe the turbine overspeed protection system, including in this description an analysis of its redundance, diversity, component reliability and testing procedures.

Answer:

Turbine Overspeed Protection System: The electrohydraulic control system for the turbine generator provides protection against shaft overspeed with two essentially separate and redundant systems. These systems offer a high degree of protection due to the fail-safe design and testing provisions.

The two separate basic systems protecting the turbine against overspeed are:

a. The Normal Overspeed Protection System, and

b. The Emergency Overspeed Protection System.

The Normal Overspeed Protection System uses two speed signals generated in the turbine front standard. For slow speed changes it implements its control through proportional position signals to the control and intercept valves in the main and crossaround steam lines, respectively.

To keep the turbine shaft speed below the setpoint of the Emergency Overspeed Protection System during rapid speed changes under normal operation of the turbine, including load reductions, the Normal Overspeed Protection System is equipped with a power-load unbalance sensing system. The performance of the Power-Load Unbalance System is superior to alternate acceleration sensing devices.

The Power-Load Unbalance System receives as input two signals; one is indicative of generator electrical power output and the other is indicative of the mechanical power produced by the turbine. These two signals are continuously compared. When the turbine mechanical power exceeds the generator electrical power by a fixed amount, it is an indication of imminent speed rise. Detection of such a difference by the Power-Load Unbalance System will initiate immediate closing of turbine steam valves. Once the steam valves have been closed, the Proportional Positioning System will again take over control of the speed. The Power-Load Unbalance System will reset automatically once the initiating condition has disappeared.

The Normal Overspeed Protection System consisting of a combination of proportional position for control of slow speed changes and power-load unbalance for fast closing has permitted full load rejection on the largest turbines built to date without the turbine shaft speed rising to the point of activating the Emergency Overspeed Protection System.

The Emergency Overspeed Protectin System is a mechanical hydraulic system. An overspeed trip device is mounted at the from end of the turbine shaft. At the overspeed setpoint, the trip device will actuate a trip mechanism. This mechanism converts the trip signal to a hydraulic signal to the steam valve actuators. This hydraulic signal is termed the Emergency Trip Signal (ETC) and is connected to the fast closing devices on the actuators of main stop valves and crossaround stop valves which will cause fast closing of the valves.

This set point of the emergency overspeed is normally 1% of rated speed above the overspeed reached by the turbine following full load rejection controlled by the Normal Overspeed Protection System. Typical values for an 1800 rpm nuclear turbine are: normal overspeed, 109% of rated speed; setpoint range for the emergency overspeed, 110-111% of rated speed.

The Emergency Overspeed Protection System is entirely mechanical hydraulic. The system takes its energy for actuation from the turbine shaft; the trip and reset mechanism has energy for tripping stored in springs. The only power supply required is a hydraulic pressure. It is inherent in the turbine design that failure of the hydraulic pressure will cause a turbine trip, which is a safe failure mode.

The Normal Overspeed Protection and the Emergency Overspeed Protection Systems constitutes two automatic, and essentially redundant and independent systems. The only exception to both redundancy and independence is the common hydraulic pressure, with the design being such that loss of hydraulic pressure causes all valves to close.

The Overspeed Protection System is designed to rest all components whose failure would significantly increase the probability of a serious overspeed incident while the turbine is carrying load. The test intervals are selected on the basis of operating experience to be practical for the operators, provide a worthwhile improvement in reliability and not wear out devices by too frequent testing.

In order to test a protection system "on-line," without major disturbance of the turbine load, it is necessary either to isolate certain portions for test, or provide parallel devices. To provide isolation of components, lock-out equipment is incorporated. This equipment is designed such that its purpose is defeated. Also, it must not seriously affect the operating reliability by causing unwarranted protective action.

Interrogatory No. I.U-14

Question:

14. Describe any analysis you have conducted about the probability in events per year of turbine failures due to stress corrosion.

Answer:

No analyses have been conducted on the probability of turbine failure induced by stress corrosion. See the response to Question 6 for more details on this point.

Interrogatory No. I.U-15

Question:

15. Describe how if at all you have taken into account turbine failures due to stress corrosion in calculating the total probability of unacceptable damage to safety-related structures from low-trajectory missiles.

Answer:

Turbine failures due to stress corrosion enter into the calculation of P_4, the probability of unacceptable damage, through the factor P_1, the probability of turbine failure. See Question 6 for details on the treatment of P_1.

Interrogatory No. I.U-16

Question:

16. Explain how you have demonstrated that you meet GDC 4 of Appendix A to 10 CFR Part 50, as implemented by Reg. Guide 1.115, or have devised an adequate alternative method other than the one described in Reg. Guide 1.115 to meet GDC 4.

Answer:

The relevant portions of GDC 4 are quoted in the response to Question 1. This GDC contains no quantitative guidance; U.S. NRC Regulatory Guide 1.115 provides quantitative guidance of two types. It is stated that:

a. The overall hazard rate from low trajectory turbine missiles should be less than the NRC objective (10^{-7}/year) for damage to all essential systems in the direct zone for a single event.

b. One method of ensuring this objective is met is to demonstrate that the probability of unacceptable damage to essential systems in the direct zone is 10^{-3}/turbine failure.

In problems of this sort, it is difficult to obtain a realistic estimate that the goal of 10^{-7} event per year has been achieved. Thus, it is generally permissible to use a criterion of about 10^{-6} events per year if it can be demonstrated that the analysis is conservative (that is, qualitative arguments can be used to demonstrate that the actual probably is lower than the estimate).

FSAR Section 3.5.1.3 does show that the analytical methods used yield a probability of unacceptable damage of less than or about 10^{-6}/year. Arguments concerning the conservative value of the model are given in response to Interrogatory I.U-4 but highlights will be summarized below:

$\underline{P_1}$ The values prescribed by SRP 3.5.1.3 were used in the analysis, but are three orders of magnitude above the estimates provided by the turbine manufacturer.

$\underline{P_2}$ The values of P_2 is based upon conservative probability distributions for missile energy and emission angles. If air resistance were taken into account, the estimates for P72 would be reduced slightly.

$\underline{P_3}$ The major portion of conservatism lies in the estimate of P_3. Firstly, the damage criterion used in most cases is spall of an external missile barrier as predicted by the modified NDRC equations. This model is likely to be quite conservative. A second conservative assumption is that spall fragments entering a building necessarily results in damage to safety systems. Finally, because of redundancy of function, damage to safety systems. Finally, because of redundancy of function, damage to several of the safety-related systems does not necessarily lead to off-site consequences in excess of 10 CFR 100 limits nor does it necessarily prevent placing and maintaining the plant in a cold shutdown condition.

Since each of the parameters P_1, P_2 and P_3 is estimated by using conservative procedures, it is clear that P_4 is a conservative estimate of the yearly probability of unacceptable consequences resulting from a turbine failure.

Interrogatory No. I.U-17

Question:

17. Identify and provide access to any and all documents referred to or relied on in preparing the response to the interrogatories regarding Contention I.U.

Answer:

A list of references is provided below.

1. General Design Criterion 4, Appendix A to 10 CFR 50.

2. USNRC Regulatory Guide 1.115, "Protection Against Low Trajectory Turbine Missiles," Revision 1, July 1977.

3. USNRC Standard Review Plan 2.2.3, "Evaluations of Potential Accidents," NUREG-0800.

4. USNRC Standard Review Plan 3.4.1.3, "Turbine Missiles," NUREG-0800.

5. Kennedy, R. P., "A Review of Procedures for the Analysis and Design of Concrete Structures to Resist Missile Impact Effects," Holmes and Narver, September 1975.

6. Semanderes, S. N., "Methods of Determining the Probability of a Turbine Missile Hitting a Particular Plant Region," WCAP-7861, February 1972.

7. Bhattacharryya, A. K., and Chaudhuri, S. K., "The Probability of a Turbine Missile Hitting a Particular Region of a Nuclear Power Plant," Nuclear Technology, Volume 28, February 1976.

8. Swan, S. W. and Meleis, M., "A Parametric Study of the Basic Functions Determining Turbine Missile Damage," Transactions of American Nuclear Society, June 1975. (Reference Uncertain)

9. Johnson, B., et al., "Calculation of Strike Probabilities and Energy Ranges Including the Effect of Missile-Barrier Interactions, Transactions of American Nuclear Society, June 1975. (Reference Uncertain)

10. Bush, S. H., "A Reassessment of Turbine Generator Failure Probability," Nuclear Safety, Volume 19, No. 6, November-December 1978.

11. Downs, J. E., "Hypothetical Turbine Missiles-Probability of Occurrence," General Electric Memo Report, March 15, 1975.

12. Stephenson, A. E., "Full Scale Tornado Missile Impact Tests," EPRI NP-440, Project 399, Final Report, July 1977.

13. Public Service Company of New Hampshire, Seabrook Station - Units 1 and 2, Seismic Qualification Review Team (SQRT) Equipment List.

14. FSAR Section 3.2, "Classification of Structures, Components and Systems."

Signatures

As to Answers:

I, Wendell P. Johnson, being first duly sworn, do depose and say that the foregoing answers are true, except insofar as they are based on information that is available to the Applicants but not within my personal knowledge, as to which I, based on such information, believe them to be true.

Wendell P. Johnson
Wendell P. Johnson

Sworn to before me this
21st day of November, 1982:

Notary Public
My Commission expires: ______________
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CERTIFICATE OF SERVICE

I, Robert K. Gad III, one of the attorneys for the Applicants herein, hereby certify that on November 22, 1982 I made service of the within "Applicants' Answer to 'NECNP First Set of Interrogatories and Request for Documents to Applicants on Contentions I.D.1, I.D.2, I.D.3, I.D.4, I.F, I.G, I.I, I.L, I.M, I.N, and I.U,'" by mailing copies thereof, postage prepaid, to:

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