12/11/78

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of PUBLIC SERVICE ELECTRIC & GAS COMPANY, et al. (Salem Nuclear Generating Station, Unit 1)	)))))))	Docket No. 50-272

LICENSEE'S RESPONSE TO INTERVENORS' FIRST SET OF INTERROGATORIES TO THE LICENSEE DATED NOVEMBER 21, 1978

Licensee hereby answers all Interrogatories served by intervenors, Alfred and Eleanor Coleman, except those Interrogatories to which Licensee has filed objections and a motion for a protective order.

Respectfully submitted,

Richard Fryling, Jr.

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Counsel for Licensee

December 11, 1978

Interrogatory 1

1. At p. 2 of the Safety Analysis, the licensee described the alternatives which were considered and "determined to be unsatisfactory" for a variety of reasons. Please describe the changes, if any, which have taken place in the status of spent fuel reprocessing and the availability of the facilities of the General Electric Company and Nuclear Fuel Services available insofar as they relate to away from reactor ("AFR") alternatives. For example, have the facilities applied for expansion of spent fuel storage? Will these facilities be available for reprocessing or AFR storage? If so, when? If not, why not?

1(a). Please explain the basis for the statement (bottom of p. 2) that "storage in the existing racks is possible, but only for a short period of time." How long? What factors and assumptions underly the time of availability, (e.g., fuel burnup, capacity factor of the unit, transshipment, etc.)?

1(b). Has the licensee considered the alternative in the intervenors' contention 9(D), "ordering the generation of spent fuel to be stopped or restricted", (e.g., operation of the unit with existing racks until an offsite AFR alternative is available.) If so, llease describe in full. If not, why not?

Answer

1. Please see answer to Question 4 of our July 31, 1978 response to NRC question. In addition, as stated in NUREG-0404 Page ES-6, "The General Electric Company's planned reprocessing plant at Morris, Illinois, has now been declared and licensed as an independent spent fuel storage installation. The initial licensed spent fuel storage capacity of about 100 MTU has been increased to about 750 MTU by installing spent fuel storage racks in its former high level waste storage pool."

1(a). Normally, a third of a reactor core is discharged into the spent fuel pool yearly. A third of a core consists of approximatey 65 fuel assemblies. The existing spent fuel racks have 264 storage locations. Assuming that a full core discharge capability is not maintained in the spent fuel pool, four yearly discharges can be made until the spent fuel storage is completely depleted.

The design average region discharge burnup is 33,000 MWG/MTU. The anticipated capacity factor is 68 percent although fuel may be discharged for operational reasons on close to an annual basis even if design values are not achieved.

1(b). Discontinuing operation of the unit has not been considered in detail in as much as this would be unacceptably costly to the licensees and its customers. In addition, the power produced would have to be replaced by fossil fueled units which would result in environmental, economic and social costs. See also NUREG-0404 page ES-7, paragraph 3.5.

Interrogatory 2

Please describe why the alternative of transshipment to other reactors is not considered a potential alternative. What reactors have been considered for transshipment?

Answer

2. Please refer to Question 2 of our July 31, 1978 submittal. Moreover the alternatives of transshipment does not offer environmental nor economic advantages. See also NUREG-0404 page 3-27 paragraph 3.2 which states, "it would be unrealistic to think that a utility with some excess storage space would prematurely fill up its pool with spent fuel from another utility."

Interrogatory 3

Please provide a full update of the licensee's plans for discharge of spent fuel, the first batch of which is planned for discharge in January, 1979 (p. 3).

Answer

The first fuel discharge into the spent fuel pool is now scheduled for April, 1979. Subsequent refuelings are planned at approximately one year intervals. See also response to interrogatory 1 (a).

Interrogatory 4

Please provide the basis and all calculations underlying the licensee's statement that "the additional cost to our customers for purchase of replacement power is estimated to be approximately $500,000 for each day the reactor is not operating."

Answer

See response to NRC Question 5(a) dated July 31, 1978.

Interrogatory 5

Has the licensee evaluated the effect, if any, of a recommendation by the Generic Environmental Impact Statement on the Handling and Storage of Spent Fuel, NUREG-0404 in favor of (AFR) facilities? If so, please describe; if not, please state the reasons.

Answer

As stated in NUREG-0404 page ES - 6, "The methods of expanding spent fuel storage capacity considered in this assessment show negligible difference in environmental impact and cost with the exception that at-reactor storage pool compact storage is least costly economically, and does not require additional transportation of spent fuel. NUREG-0404 contemplates the use of AFR storage in addition to AR storage such as being requested for Salem 1.

Interrogatory 6

What increase would occur in radiation levels in the storage water of the spent fuel pool in the event that the licensee's application is granted? (see p. 7)

6(a). What increase in radioactive materials and in radiation levels would occur in the coolant water filters? What increase would occur in the screens, traps, drains and pipes? Please provide all relevant calculations and the basis therefore.

6(b). Please explain the statement at p. 8 that "the amount of corrosion products released into the pool during any year would be the same regardless of the storage capacity of the pool," assuming increased compaction and several years of discharged fuel?

Answer

6. The spent fuel pool water clean-up system is designed to process the radioactivity released to the water from storage of (1/3 core) spent fuel assemblies on an annual basis. During this time period, this radioactivity (including corrosion and activation products) will be reduced to negligible levels by being removed from the water by the clean-up system and radioactive decay. Each subsequent discharge of spent fuel assemblies will be placed in the storage pool approximately a year after placement of the prior discharge and the clean-up system, during the next one year period, removes the activity in the water. The radioactivity being removed is predominantly from the most recently discharged fuel. The contribution from the fuel discharged in prior years is negligible. Hence, the design basis of the spent fuel pool clean-up system is unaffected by the number

of spent fuel pool storage locations, i.e., the request for additional fuel storage capacity.

6(a). Based on the response to Interrogatory 6, we do not expect an increase in radioactive materials on filters, in traps, screens, drains or pipes.

6(b). See response to Interrogatory 6.

Interrogatory 7

If the quantity of spent fuel is increased by a factor of four (4), how would the crud release rate be affected? Similarly, how would the total quantity of radioactive materials released into the spent fuel pool be affected? (see p. 9)

Answer

7. See response to Interrogatory 6.

Interrogatory 8

Please quantify the amount of corrosion products which would be present in the spent fuel pool as a result of increased compaction.

Answer

8. See response to Interrogatory 6. Our response submitted on July 31, 1978 to NRC question 11, tabulates the concentrations of principal radionuclides expected to be found in the spent fuel pool water.

Interrogatory 9

Please explain why the "resin replacement frequency will not be significantly altered by the increase in spent fuel storage capacity" (p. 9). Please describe how reliance on the "differential pressure increase" differs from the "loss of capacity to remove radioactive contaminants" in determining resin replacement (p. 8). Show calculations for predictions in each case.

Answer

9. See response to interrogatory 6. It is expected that pressure differential wil be the controlling factor in filter and resin replacement. The demineralizers are also used to remove radioactivity by the process of ion exchange. It is possible that in order to reduce exposure to plant personnel this process would lead to the replacement of resin due to high radioactivity level to increase prior to a high pressure differential indication. A water sampling procedure will also be utilized to assure that the capacity to remove radioactive contaminants is maintained in the demineralizer.

INTERROGATORY 10

What other gasses beside Kr 85 may be released from a spent fuel storage area (page 10)? Please identify and quantify.

ANSWER

Other radioactive gases that might be released from the spent fuel pool consist of radioactive xenons sucn as Xe-131m, Xe-133 and Xe-135, tritium (H-3) and radioactive iodines such as I-131 and I-133.

However, because of the half lives of those isotopes (except H-3) relative to Kr-85, the curies released of these isotopes will be substantially lower than the amount indicated for Kr-85 and the release will occur approximately during the first two months of fuel storage. Hence, increased fuel storage time will not result in any increase in releases to the environment of other radioactive gases.

The amount of gaseous tritium released from the spent fuel pool area is estimated to be 240 curies per year as a result of evaporation of water from the pool. No additional tritium release is expected as a result of increased fuel storage time since the prime sources of tritium in reactor water are ternary fission and neutron activation of other isotopes in the water. Essentially, no fission or neutron activation will take place in the fuel storage pool.

INTERROGATORY 10(b)

Why is there intended to be no separate monitoring device to measure radioactive gases released from the spent fuel storage area? Identify any impediments to such a monitoring system, if any.

ANSWER

A separate monitoring device is not necessary as all potentially radioactive station effluent, including air from above the spent fuel pool is monitored prior to discharge from vent. The concentrations of gaseous activity around the spent fuel pool are expected to be below the lower range of commercially available continuous gas monitors.

QUESTION NO. 11

Please describe how the Table 2.0-1 (p. 39) relates to the determination of expected radioactive gases released from the spent fuel storage area during the period of storage ?

ANSWER

Table 2.0-1 does not directly relate to the expected release of gases from the spent fuel storage area. The reason for including this Table was to illustrate that the total gaseous radioactive releases from Salem Unit No. i in 1976 and 1977 (first half) were very small in magnitude.

INTERROGATORY 11(a)

What is the purpose of Table 2.0-1?

ANSWER

Refer to the response to Interrogatory 11.

INTERROGATORY 11(b)

What is the lower limit of detection applicable to Table 2.0-1?

ANSWER

The lower limits of detection for those isotopes presented in Table 2.0-1 are $10^{-4} \mu$ Ci/ml for Kr-85, $10^{-12} \mu$ Ci/ml for I-31 and $10^{-6} \mu$ Ci/ml for tritium.

INTERROGATORY 11(c)

What techniques are available for measuring below this limit?

ANSWER

The amount of radioactivity estimated to be released to the environment from the spent fuel storage pool is very small and does not have any environmental significance. Furthermore, the plant effluent radiation monitoring system is capable of measuring radioactive releases that are a small fraction of the applicable Technical Specification release limit.

As a result, PSE&G has not investigated any techniques for measuring below the indicated limits of detection.

Interrogatory 12

At page 14, you describe the "B-10 loading of 0.025 gm/cm^2." Please explain this value in light of Table 3.1-1 which refers to 0.05 gm/cm^2 as minimum and Table 3.1-4 which shows 0.025 gm/cm^2 as minimum.

Answer

The February 14, 1978 PSE&G license amendment submittal delineates a change in boron loading from 0.025 to 0.020 gms B-10/cm^2 in each Boral plate.

There are two poison plates utilized between fuel assemblies in the Salem storage racks. Consequently, the total B-10 loading between two adjacent fuel assemblies is (2) x (0.02) = 0.04 gms B-10/cm^2. Therefore table 3-1-4 and page 14 are not inconsistent.

Interrogatory 12 (a)

What are the correct values and the B-10 loadings?

Answer

12 (a) See response to Interrogatory 12.

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Interrogatory 12 (b)

What is the uncertainty of the value of the expected loadings?

Answer

12 (b) The value of .02 gms B-10/cm^2 is a minimum value. To assure that the minimum value is achieved in each Boral plate, a higher average concentration of B_4C is utilized during the fabrication. Additional tests are made after fabrication to assure that the minimum boron levels are achieved.

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Interrogatory 13

Please describe why the licensee believes that the increased spent fuel compaction and storage will not affect the consequences of a spent fuel pool accident (see p. 21). Explain how the consequences of an accident would be affected by acts of sabotage.

Answer

13. Please refer to NRC questions on the same subject in our July 21, 1978 submittal and Question 15 of our October 13, 1978 submittal. As shown in the answers to these questions, the consequences of a spent fuel pool accident are not affected.

Interrogatory 14

Please describe how the fuel cask would be handled and the basis for the chosen method. Also, please describe the specific controls to be employed to assure that the cask handling does not encroach upon the pool area.

Answer

14. The cask handling crane is designed such that it is physically impossible for it to travel over the spent fuel pool.

Interrogatory 15

Please describe the basis for the statement at p. 22 that "there is no deterioration or corrosion of stainless steel in this environment." (emphasis added) Please describe the variables which affect the rate of deterioration or corrosion, if any. What assumption underly the determination that no deterioration or corrosion will occur?

Answer

15. Stainless steel of the type being used in the spent fuel racks at Salem 1 has been widely utilized in the nuclear industry. We are not aware of any corrosion of stainless steel in environments similar to the Salem 1 spent fuel pool.

Interrogatory 16

Please describe the "non-destructive testing of the cells".

Answer

16. Non-destructive testing of fuel cells is addressed in the October 31, 1978 submittal in the answer to question "Provide a description of the procedures used to insure that the fuel storage racks are leak tight."

Interrogatory 16(a)

What would the consequences of a less than 100 percent leak tightness?

Answer

16(a). The consequences of less than 100 percent leak tightness are described in Exxon Nuclear Company document XN - NS - 009 entitled "ENC Fuel Storage Racks Corrosion Program Boral - Stainless Steel - PROPRIETARY." The October 31, 1978 submittal to teh NRC documents the results of tests performed by Exxon Nuclear to characterize the effects of a fuel cell that develops a leak in the fuel storage pool environment.

These two reports demonstrate that there are no safety implications associated with less than 100% leak tightness.

Interrogatory 16(b)

Describe the basis for the 95 percent confidence level, including any calculations and methodologies.

Answer

16(b). The statistical basis for satisfying the 95 percent confidence level is discussed in the October 31, 1978 submittal to the NRC which addresses "... the procedures used to insure that the fuel storage racks are leak tight." Essentially, the helium mass spectrometer leak test program is used as an overcheck to assure the 95 percent confidence level. The leak test sampling frequency of the fuel cell was chosen to satisfy a statistical 95/95 tolerance limit, as explained in the submittal.

The leak test program is designed to qualify the entire production of fuel storage cells to the 95/95 tolerance limit, as one lot. The lot is split into several sublots for purposes of production control. Sample cells are selected at constant intervals from each sublot. Each sublot must meet a specified acceptance limit, in order to insure that the entire production meets the 95/95 tolerance limit.

The statistical methods used in this type of sampling program are explained in the paper: "On the Use of Tolerance Intervals in Acceptance Sampling by Attributes," by J. L. Jaech, Journal of Quality Technology, Volume 4, No. 2, April 1972.

Interrogatory 17

If in-pool surveillance finds problems requiring repair of spent fuel rods or racks, what are the licensee's contingency plans for removal and repair?

Answer

17. The October 31, 1978, submittal to the NRC describes the actions required to remove a fuel assembly from a swollen fuel cell.

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Interrogatory 22

The calculated K_{eff} values for ORNL Critical Lattices (p. 46) cases four and five sets forth central values which are outside the range of values for the 95 percent confidence level. Please explain how it is possible for the central value and K_{eff} to be outside the range of values provided in the table.

Answer

Please see answer to Question 12 in our May 17, 1978 submittal to the NRC.

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Before the Atomic Safety and Licensing Board

In the Matter of PUBLIC SERVICE ELECTRIC AND GAS COMPANY, et al. (Salem Nuclear Generating Station, Unit 1)	)))))))	Docket No. 50-272

CERTIFICATE OF SERVICE

I hereby certify that copies of "Licensee's Response to Intervenors' First Set of Interrogatories to the Licensee Dated November 21, 1978", in the captioned matter, have been served upon the following by deposit in the United States mail this 11th day of December, 1978:

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VERIFICATION

EDWIN A. LIDEN, being duly sworn, states that he is Project Licensing Manager - Salem Nuclear Generating Station of Public Service Electric and Gas Company; that he has been duly authorized to execute, verify and file the foregoing document, "Licensee's Response to Intervenors' First Set Of Interrogatories To The Licensee Dated November 21, 1978"; that he has read the contents of same and that the statements contained therein are true and correct to his best information, knowledge and belief.

Edwin A. Liden

EDWIN A. LIDEN

Subscribed and sworn to)
before me this 11th day)
of December, 1978.)

Annette S. Warczakowski

Notary Public

ANNETTE S. WARCZAKOWSKI
A NOTARY PUBLIC OF NEW JERSEY
My Commission Expires Jan. 4, 1982