50-443

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 10, 1997

Ms. Jane Doughty
The Seacoast Anti-Pollution League
P.O. Box 1136
Portsmouth, NH 03802

Dear Ms. Doughty:

This is in response to your letter of October 11, 1996, in which you asked several additional questions relating to my letter to you dated September 9, 1996. To avoid lengthy repetition, a number of your questions or comments contained in your letter have been paraphrased. They are shown below in italics with our responses and discussion following.

Question

> *How is it that you maintain the licensee met the ACTION requirements of Technical Specifications 3.11.2.1 and 3.3.3.10 and assured that the dose rates were decreased to the specified limits within 15 minutes? Isn't deliberately performing a containment purge, with no equipment in operation that permits one to know what the dose limits are within 15 minutes, flagrantly in violation of these Technical Specifications?*

Discussion

> Referring to my September 9, 1996, letter, my response therein to your Question 1 explained the compensatory actions the licensee must take to continue releasing radioactive gaseous effluents while the number of operable radioactive gaseous effluent monitoring instrumentation channels are less than the minimum number required to be operable. In your October 11, 1996, letter, your discussion leading to the above question attempts to link the Limiting Conditions for Operation (LCO) and associated ACTION of Technical Specifications (TSs) 3.3.3.10 and 3.11.2.1 in a manner such that if effluent monitoring instrumentation is inoperable the release of gaseous effluents monitored by that instrumentation must be suspended. This interpretation of these specifications is not correct. TS 3.3.3.10 identifies the operability requirements, including setpoints, for the monitoring instrumentation and the actions to be taken if the instrumentation is inoperable. TS 3.11.2.1 specifies dose limits for determining the setpoints for the instrumentation and the actions to be taken when the limits are exceeded.
>
> In the event dose rate limits of TS 3.11.2.1 are exceeded, the ACTION requires decreasing effluent release rates within 15 minutes to reduce the dose rates to within the limits. The requirements of this ACTION are entirely consistent with continued effluent release with inoperable monitoring instrumentation as permitted by TS 3.3.3.10 provided the grab samples (for noble gas activity) are taken and analyzed and alternate continuous sampling is performed (for iodine and particulates) as

DF01/1

specified in Table 3.3-13 ACTION 33 and 35. When Table 3.3-13 ACTION 33 and/or 35 apply and if the analysis results of grab samples or auxiliary continuous samples indicate the limits of TS 3.11.2.1 are exceeded, then TS 3.11.2.1 ACTION must be followed. However, if the analysis results do not indicate the dose limits are exceeded, then TS 3.11.2.1 ACTION does not apply.

Question

Doesn't ACTION 35, which applies to subparts b. and c. of the system require continuous collection of samples with auxiliary sampling equipment as required in the Offsite Dose Calculation Manual? Wasn't the licensee in violation of the requirements of ACTION 32 and ACTION 35?

Answer

The response provided in my September 9, 1996, letter focused mostly upon the requirements for release of radioactive noble gaseous effluents since the C-10 Radiological Monitoring System findings of November 29, 1995, which raised the issue of containment purging with plant vent wide range gas monitor (WRGM) inoperable, did not relate to the release of iodine or particulate material. My response should not have been so limited.

You are correct that ACTION 32 and 35 also apply when the plant vent wide range gas monitor is inoperable. For a period of time on November 28 - 29, 1995, a power supply failure affected certain WRGM instruments that are listed on Table 3.3-13 of the Seabrook TS. The licensee declared the entire WRGM inoperable until maintenance was completed. During the period of time the WRGM was inoperable, the licensee complied with all applicable ACTIONS listed in Table 3.3-13.

Question

In light of the C-10 findings, which indicated that things might not be as "normally" expected, wasn't the loss of the TLD at 243 degrees 1.2 miles from the plant cause for concern by the NRC, and shouldn't the NRC require the licensees or the state to provide a regular documented inventory that the TLDs are in place with no evidence of tampering?

Discussion

As discussed in NRC Inspection Report 50-443/96-05, dated June 25, 1996, NRC concluded that Seabrook Station activities were not the cause of the instrument indications observed previously by C-10's radiation monitors. As we previously stated in our September 9, 1996 response, the loss of the NRC thermoluminescent dosimeter (TLD) at station 5 did not warrant an investigation and the consequent loss of TLD measurement did not have

any significant impact on the quality of the overall data, or its use and application for radiological assessment.

In addition to the NRC's TLD network, the licensee and the State of New Hampshire maintain their own independent TLD networks around the site. The State of New Hampshire's TLDs are installed at different locations than the licensee's, but some may be colocated with NRC devices. As needed, these independent TLD networks would serve to augment the data base in the case of an emergent or off-normal condition at Seabrook Station. Relative to interpretation of the TLD measurements as reported in NUREG-0837 report for the period in question, we found nothing significant relative to any of the measurements that was out of the range of normally expected values.

A regular documented inventory is conducted for the NRC by the State of New Hampshire on a quarterly basis while exchanging the TLDs. Upon being sent to the NRC for processing, we consistently document discrepancies, such as missing dosimeters. Given the purpose and objective of the TLD network program, we are confident that our processes and controls are sufficient.

Question

Given the licensee's initial error wherein the licensed thermal power was exceeded and the long time span involved in the second incident, how is it that the NRC has confidence that the licensee correctly determined that 3418 Mwt and 3413 Mwt were the highest thermal power levels reached at any point in time during these two incidents?

Discussion

The first occurrence of overpower resulted from deliberate reactor operator actions to restore the reactor to full power following an indicated power decrease. The operator action was taken without a full understanding of the cause for the indicated power decrease. It was not until the next shift questioned the earlier power adjustment that it was determined that an instrument transmitter calibration had invalidated the normalizing constants used in the calorimetric, and that the previous shift's action to adjust the reactor to full power actually caused the power to exceed the licensed limit.

The second event occurred when the main plant computer was restarted following maintenance. The operators failed to recognize that upon restart, the main plant computer defaulted the calorimetric to the same mode that was involved in the earlier overpower event. At that time, the evaluation of the first event was not yet complete, and the incorrect normalizing constants had not yet been updated.

The detailed evaluations which afforded a full understanding of both events, including the power levels reached during the events, were not fully completed until 4 days after the second event. The NRC was involved in considerable discussions with involved personnel starting the day after the first event, and an NRC inspector reviewed the completed evaluations.

Question

Did Yankee Atomic Electric Company do t. calculations providing the basis for Seabrook Station's 100% rating being 3411 Mwt?

Discussion

The reactor design for the Seabrook Station is discussed in Chapter 4 of the Seabrook Station Updated Final Safety Analysis Report (UFSAR). Specifically, the thermal and hydraulic design is discussed in Section 4.4 of the UFSAR. The basic thermal and hydraulic design providing the basis for Seabrook's 100% rating was performed by the Westinghouse Electric Corporation.

In addition to the basic thermal and hydraulic design of the reactor, other analyses are required to demonstrate the adequacy and safety of the design of the reactor and other systems during transient and accident conditions. The large-break and small-break loss-of-coolant accident (LOCA) analyses as described in Chapter 15 of the UFSAR have been performed by Westinghouse Electric. The non-LOCA transient and accident analyses described in Chapter 15 of the UFSAR have been performed by Yankee Atomic Electric Company. Analyses relating to containment performance as described in Chapter 6 of the UFSAR have been performed by Raytheon Engineers and Constructors.

UFSAR Section 4.4.7 identifies the various reference documents which, in part, identify the methodology used to establish the reactor thermal and hydraulic design parameters. Additionally, Section 6.8.1.6.b, Appendix A of the Seabrook Station Technical Specifications identifies the approved analytical methods to be used in the establishment of core operating limits.

On November 23, 1994, the NRC issued Amendment 33 to the Seabrook operating license. This amendment authorized operation of Seabrook with an expanded axial flux difference band and authorized certain fuel design enhancements. The Yankee Atomic Electric Company provided the analytical methodology and performed the calculations to support North Atlantic's license amendment application, except for the LOCA analyses provided by Westinghouse.

The Seabrook UFSAR, Technical Specifications, and Amendment 33 are available at the Local Public Document Room located at the Exeter Public Library, Founders Park, Exeter, NH. For your convenience, I have

enclosed the reference list to UFSAR, Revision 4, Section 4.4 and Section 6.8.1.6.b (as revised through Amendment 33) of the Seabrook Technical Specifications. References bearing the designation *YAEC-* followed by a sequence of numbers and letters identify documents prepared by the Yankee Atomic Electric Company.

Question

Has the NRC verified that the sampling and analysis required by Technical Specifications 3/4.4.8 and Table 4.4-3 were carried out by the licensee, and has the NRC independently reviewed the findings?

Discussion

TS 3/4.4.8 and Table 4.4.3 require, in part, periodic sampling and analysis of the reactor coolant for gross radioactivity at least once every 72 hours and for radioiodines at least once every 14 days. More frequent sampling and analyses are required if the specified limits are exceeded. The results from the analyses of routine samples of reactor coolant, taken in conformance with TS 3/4.4.8 after the overpower events of October 19 and 26, 1995, did not exceed the specified limits and do not indicate that any fuel clad damage occurred. The licensee includes a report of the most recent reactor coolant specific activity determinations at the daily Station Director's Meeting which normally is attended by NRC personnel at the site. Additionally, the NRC resident inspectors review the results on a routine basis.

Questions

Who did the visual examinations of the eight fuel assemblies? How was the sample size of eight assemblies selected, and were the assemblies randomly selected? Does the NRC consider eight fuel assemblies an adequate sample size in these circumstances? Would visual examination detect pinhole leaks in fuel cladding?

Discussion

Updated Final Safety Analysis Report Subsection 4.2.4.6.g. states, in part, the following:

> "Visual irradiated fuel inspections will be conducted as necessary during each refueling. Selected fuel assemblies may be inspected for fuel rod failure, structural integrity, crud deposition, rod bow and other irregularities. Fuel assemblies will be selected for inspection

> based upon performance history and recommendations made by the fuel supplier.
>
> "The fuel inspection program will be expanded to include more fuel assemblies or greater detail of examination if high coolant activity is experienced during operation, irregularities are noted in fuel performance, irregularities are noted during routine inspections, or if a new fuel design is incorporated."

I indicated in my September 9, 1996, letter, the amounts of overpower in both events were not significant from the standpoint of inducing fuel cladding failure. I indicated, further, that extensive examination for fuel clad damage was unnecessary absent an increase in the gross radioactivity of the reactor coolant which is an excellent indicator of fuel clad integrity. Considering the above along with the overall excellent performance history of the nuclear steam supply system for the cycle that had just ended, and the lack of any other contraindications, the size of the sample selected by the licensee was not unreasonable. The licensee selected the assemblies to be examined based upon the position in the core and the assembly irradiation history. Licensee personnel performed the visual examinations.

The visual examinations are intended to reveal any physical manifestations of unexpected or unusual conditions. The detection of pinholes in fuel cladding would be most difficult, if possible at all.

Questions

SAPL asserts (with regard to the September 9, 1996, response to SAPL Question 5 relating to the radiological conditions encountered during the reactor head funnel guide inspection) "that the exposures <u>were</u> representative of actual radiological conditions" Why were exposures higher than expected? Why does the NRC permit operation of the plant when this inspection is incomplete?

Discussion

The basis for SAPL's assertion is not provided. If SAPL has any information that supports this assertion, it should be provided.

As stated in the September 9, 1996, response, the licensee's estimate of the radiological conditions to be encountered during the reactor head funnel guide inspection was based on dose measurements under the reactor vessel. When the funnel guide inspection was being planned, the reactor head was in place on the reactor vessel and the reactor was operating,

so the planners relied upon dose rate measurements made during a previous outage in the space <u>under</u> the reactor. The planners assumed that the exposure conditions under the reactor would be representative of the conditions workers would encounter during the inspection. Actual dose rates encountered were much higher. As was previously explained, the licensee recognized that the entire inspection would involve considerably more personnel exposure than originally expected, so further inspection was suspended.

NRC Information Notice 94-40 (May 26, 1994) and Information Notice 94-40, Supplement 1 (December 15, 1994) were issued to alert licensees of several events involving detached guide funnels underneath the reactor head. NRC Information Notices, in part, are for the purpose of alerting licensees of conditions that have been discovered at other facilities. Information Notices do not impose NRC requirements, and no specific action or written response from licensees are required. Thus, there is no restriction for continued operation of Seabrook because the licensee has not completed inspection of the reactor head guide funnels. There are no technical specification requirements applicable to this inspection.

Question

Does the NRC believe the number of samples for entrained noble gas, as identified in the response to Question 6, represents an adequate program, and does this form a solid basis for concluding that the measurements reported by C-10 can not be attributed to the release of noble gases from Seabrook?

Discussion

The conclusion reached in Inspection Report 50-443/96-05 that operating activities at Seabrook could not have caused the indications observed by the C-10 monitoring stations is based upon much more information than merely the results of the liquid samples taken and analyzed for entrained or dissolved noble gases. The conclusion, instead, is supported by information obtained from a number of diverse sources including in-plant area radiation measurements, reactor coolant sampling for gross radioactivity, in-plant process (gaseous and liquid) radiation

measurements, personnel dosimetry, operational logs, and off-site environmental monitoring. No data was uncovered from these sources for the relevant period that is in conflict with the conclusion of the inspection report.

Sincerely,

Original signed by

Albert W. De Agazio, Sr. Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-443

Enclosures: As stated

cc w/encls: See next page

Distribution
Docket File
PUBLIC
PDI-3 RF
SVarga
PMilano
OGC
ACRS
JRogge, RI

DOCUMENT NAME: G:\DEAGAZIO\SAPL1011.96
To receive a copy of this document, indicate in the box: "C" = Copy without attachment/enclosure "E" = Copy with attachment/enclosure "N" = No copy

OFFICE	PM:PDI-3	LA:PDI-1	(A)D:PDI-3		
NAME	ADeAgazio	SLittle	PMilano		
DATE	04/9/97	04/9/97	04/10/97		

OFFICIAL RECORD COPY

160049

This Enclosure contains the references to the Seabrook Station, Unit No. 1 UFSAR, Revision 4, Section 4.4.7.

ENCLOSURE 1

4.4.7 References

1. Christensen, J. A., Allio, R. J. and Biancheria, A., "Melting Point of Irradiated UO_2," WCAP-6065, February 1965.

2. Hellman, J. M. (Ed.), "Fuel Densification Experimental Results and Model for Reactor Application," WCAP-8218-P-A (Proprietary), March 1975 and WCAP-8219-A, March 1975.

3. Tong, L. S., "Boiling Crisis and Critical Heat Flux," AEC Critical Review Series, TID-25887, 1972.

4. "Evaluation of Westinghouse Request for Generic Approval", ENCLOSURE to letter from Carl Berlinger (NRC) to E. P. Rahe, Jr., (Westinghouse), "Request for Reduction in Fuel Assembly Burnup Limit for Calculation of Maximum Rod Bow Penalty", dated June 18, 1986.

5. Letter, L. A. Tremblay (Vermont Yankee Nuclear Power Corporation) to USNRC, "FROSSTEY-2 Fuel Performance Code - Vermont Yankee Response to Remaining Concerns", BVY 92-54, May 15, 1992.

6. Letter, P. Sears (USNRC) to L. A. Tremblay (Vermont Yankee Nuclear Power Corporation), "Vermont Yankee Nuclear Power Station - Safety Evaluation of FROSSTEY-2 Computer Code (TAC No. M68216)", September 24, 1992.

7. WCAP-8762-P-A, "New Westinghouse Correlation WRB-1 for Predicting Critical Heat Flux in Rod Bundles With Mixing Vane Grids", Westinghouse Electric Corporation, F. E. Motley, July 1984.

8. Tong, L. S., "Prediction of Departure from Nucleate Boiling for an Axially Non-Uniform Heat Flux Distribution," J. Nucl. Energy, 21, 241-248 (1967).

9. YAEC-1849P, "Thermal-Hydraulic Analysis Methodology Using VIPRE-01 for PWR Applications", Yankee Atomic Electric Company, F. L. Carpenito, October 1992.

10. Letter from T. C. Feigenbaum (North Atlantic Energy Service Co.) to USNRC, "Response to Request for Additional Information (TAC M86957 and TAC M86958)", March 9, 1994.

11. NP-2511-CCM Volumes 1-5, "VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores", Electric Power Research Institute.

12. Cadek, F. F., Motley, F. E. and Dominicis, D. P., "Effect of Axial Spacing on Interchannel Thermal Mixing with the R Mixing Vane Grid," WCAP-7941-P-A (Proprietary), January 1975 and WCAP-7959-A, January 1975.

13. Rowe, D. S., Angle, C. W., "Crossflow Mixing Between Parallel Flow Channels During Boiling, Part II Measurements of Flow and Enthalpy in Two Parallel Channels," BNWL-371, part 2, December 1967.

14. Rowe, D. S., Angle, C. W., "Crossflow Mixing Between Parallel Flow Channels During Boiling, Part III Effect of Spacers on Mixing Between Two Channels," BNWL-371, part 3, January 1969.

15. Gonzalez-Santalo, J. M. and Griffith, P., "Two-Phase Flow Mixing in Rod Bundle Subchannels," ASME Paper 72-WA/NE-19.

16. Motley, F. E., Wenzel A. H., Cadek, F. F., "The Effect of 17x17 Fuel Assembly Geometry on Interchannel Thermal Mixing," WCAP-8298-P-A (Proprietary), January 1975 and WCAP-8299-A, January 1975.

17. Cadek, F. F., "Interchannel Thermal Mixing Vane Grids," WCAP-7667-P-A (Proprietary), January 1975 and WCAP-7755-A, January 1975.

18. Hochreiter, L. F., "Application of the THINC-IV Program to PWR Design," WCAP-8054 (Proprietary), October 1973, and WCAP-8195, October 1973.

19. Hochreiter, L. E., Chelemer, H. and Chu, P. T., "THINC-IV An Improved Program for Thermal-Hydraulic Analysis of Rod Bundle Cores," WCAP-7956, June 1973.

20. Dittus, F. W., and Boelter, L. M. K., "Heat Transfer in Automobile Radiators of the Tubular Type," Calif. Univ. Publication In Eng., 2, No. 13, 443461 (1930).

21. Weisman, J., "Heat Transfer to Water Flowing Parallel to Tube Bundles," Nucl. Sci. Eng., 6, 78-79 (1959).

22. Thom, J. R. S., Walker, W. M., Fallon, T. A. and Reising, G. F. S., "Boiling in Sub-Cooled Water During Flow up Heated Tubes or Annuli," Prc. Instn. Mech. Engrs., 180, Pt. C, 226-46 (1955-66).

23. Hetsroni, G., "Hydraulic Tests of the San Onofre Reactor Model," WCAP-3269-8, June 1964.

24. Hetsroni, G., "Studies of the Connecticut-Yankee Hydraulic Model," NYO-3250-2, June 1965.

25. Idel'chik, I. E., "Handbook of Hydraulic Resistance," AEC-TR-6630, 1960.

26. Moody, L. F., "Friction Factors for Pipe Flow," Transaction of the American Society of Mechanical Engineers, 66, 671-684 (1944).

27. Maurer, G. W., "A Method of Predicting Steady State Boiling Vapor Fractions in Reactor Coolant Channels," WAPD-BT-19, pp. 59-70. June 1960.

28. Griffith, P., Clark, J. A. and Rohsenow W. M., "Void Volumes in Subcooled Boiling Systems," ASME Paper No. 58-HT-19.

29. Bowring, R. W., "Physical Model, Based on Bubble Detachment, and Calculation of Steam Voidage in the Subcooled Region of a Heated Channel," HPR-10, December 1962.

30. Letter from C. E. Rossi (USNRC) to J. A. Blaisdell (UGRA Executive Committee), "Acceptance for Referencing of Licensing Topical Report, EPRI NP-2511-CCM, 'VIPRE-01: A Thermal-Hydraulic Analysis Code for Reactor Cores, Volumes 1, 2, 3, and 4'", Attachment: "Safety Evaluation Report on EPRI NP-2511-CCM VIPRE-01", May, 1986.

31. Letter from A. W. De Agazio (USNRC) to T. C. Feigenbaum (NAESCO), "Acceptance for Referencing of YAEC-1849P, 'Thermal-Hydraulic Analysis Methodology Using VIPRE-01 for PWR Applications', for the Seabrook Station, Unit No. 1 (TAC M86958)", August 15, 1994.

32. Duncan, R. N., "Rabbit Capsule Irradiation of UO_2," CVTR Project, CVNA-142, June 1962.

33. Gyllander, J. A., "In-Pile Determination of the Thermal Conductivity of UO_2 in the Range 500-2500 C," AE-411, January 1971.

34. "Partial Response to Request Number 1 for Additional Information on WCAP-8691, Rev. 1," letter from E. P. Rahe, Jr., (Westinghouse), to J. R. Miller (NRC), NS-EPR-2515, dated October 9, 1981; "Remaining Response to Request Number 1" letter, from E. P. Rahe, Jr., (Westinghouse), to J. R. Miller (NRC), NS-EPR-2572, dated March 16, 1982.

35. Stora, J. P., "In-Reactor Measurements of the Integrated Thermal Conductivity of UO_2 - Effect of Porosity," Trans. ANS, 13, 137-138 (1970).

36. International Atomic Energy Agency, "Thermal Conductivity of Uranium Dioxide," Report of the Panel held in Vienna, April 1965, IAEA Technical Reports Series, No. 59, Vienna, The Agency, 1966.

37. Carter, F. D., "Inlet Orificing of Open PWR Cores," WCAP-9004, January 1969 (Proprietary) and WCAP-7836, January 1972 (Nonproprietary).

38. Novendstern, E. H. and Sandberg, R. O., "Single Phase Local Boiling and Bulk Boiling Pressure Drop Correlations," WCAP-2850 (Proprietary), April 1966 and WCAP-7916, June 1972.

39. Owens, Jr., W. L., "Two-Phase Pressure Gradient," International Developments in Heat Transfer, Part II, pp. 363-368, ASME, New York, 1961.

References 40 through 68 are not used.

69. J. A. Boure, A. E. Bergles, and L. S. Tong, "Review of Two-Phase Flow Instability," Nucl. Engr. Desi n 25 (1973) p. 165-192.

70. R. T. Lahey and F. J. Moody, "The Thermal Hydraulics of a Boiling Water Reactor," American Nuclear Society, 1977.

71. P. Saha, M. Ishii, and N. Zuber, "An Experimental Investigation of the Thermally Induced Flow Oscillations in Two-Phase Systems," J. of Heat Transfer, Nov. 1976, pp. 616-662.

72. Virgil C. Summer FSAR, Docket #50-395.

73. Byron/Braidwood FSAR, Docket #50-456.

74. South Texas FSAR, Docket #50-498.

75. S. Kakac, T. N. Veziroglu, K. Akyuzlu, O. Berkol, "Sustained and Transient Boiling Flow Instabilities in a Cross-Connected Four- Parallel-Channel Upflow System," Proc. of 5th International Heat Transfer Conference, Tokyo, Sept. 3-7, 1974.

76. H. S. Kao, C. D. Morgan, and W. B. Parker, "Prediction of Flow Oscillation in Reactor Core Channel," Trans. ANS, Vol. 16, 1973, pp. 212-213.

77. Ohtsubo A., and Uruwashi, S., "Stagnant Fluid due to Local Flow Blockage," J. Nucl. Sci. Technol., 9, No. 7, 433-434, (1972).

78. Basmer, P., Kirsh, D. and Schultheiss, G. F., "Investigation of the Flow Pattern in the Recirculation Zone Downstream of Local Coolant Blockages in Pin Bundles," Atomwirtschaft, 17, No. 8, 416-417, (1972). (In German).

79. Guimond, P. J., "Core Thermal Limit Protection Function Setpoint Methodology for Seabrook Station", YAEC-1854P, Yankee Atomic Electric Company, October 1992.

This Enclosure contains Section 6.8.1.6.b of the Seabrook Station, Unit No. 1 Technical Specifications as issued with Amendment 33 dated November 23, 1994.

ENCLOSURE 2

6.8.1.6.a. (Continued)

5. Shutdown Rod Insertion limit for Specification 3.1.3.5.

6. Control Rod Bank Insertion limits for Specification 3.1.3.6.

7. AXIAL FLUX DIFFERENCE limits for Specification 3.2.1.

8. Heat Flux Hot Channel Factor, F_{Q}^{RTP} and K(Z) for Specification 3.2.2.

9. Nuclear Enthalpy Rise Hot Channel Factor, and $F_{\Delta H}^{RTP}$ for Specification 3.2.3.

The CORE OPERATING LIMITS REPORT shall be maintained available in the Control Room.

6.8.1.6.b The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in:

1. WCAP-10266-P-A, Rev. 2 with Addenda (Proprietary) and WCAP-11524-A (Nonproprietary), "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code", August, 1986

   Methodology for Specification:
   3.2.2 - Heat Flux Hot Channel Factor

2. WCAP-10079-P-A (Proprietary) and WCAP-10080-A (Nonproprietary), "NOTRUMP: A Nodal Transient Small Break and General Network Code", August, 1985

   Methodology for Specification:
   3.2.2 - Heat Flux Hot Channel Factor

3. YAEC-1363-A, "CASMO-3G Validation," April 1988.

   YAEC-1659-A, "SIMULATE-3 Validation and Verification," September 1988.

   Methodology for Specifications:
   3.1.1.1 - SHUTDOWN MARGIN for MODES 1, 2, 3, and 4
   3.1.1.2 - SHUTDOWN MARGIN for MODE 5
   3.1.1.3 - Moderator Temperature Coefficient
   3.1.3.5 - Shutdown Rod Insertion Limit
   3.1.3.6 - Control Rod Insertion Limits
   3.2.1 - AXIAL FLUX DIFFERENCE
   3.2.2 - Heat Flux Hot Channel Factor
   3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

4. Seabrook Station Updated Final Safety Analysis Report, Section 15.4.6, "Chemical and Volume Control System Malfunction That Results in a Decrease in the Boron Concentration in the Reactor Coolant System".

   Methodology for Specifications:
   3.1.1.1 - SHUTDOWN MARGIN for MODES 1, 2, 3, and 4
   3.1.1.2 - SHUTDOWN MARGIN for MODE 5

6.8.1.6.b. (Continued)

5. YAEC-1241, "Thermal-Hydraulic Analysis of PWR Fuel Elements Using the CHIC-KIN Code", R. E. Helfrich, March 1981

   Methodology for Specification:
   3.2.1 - AXIAL FLUX DIFFERENCE
   3.2.2 - Heat Flux Hot Channel Factor
   3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

6. YAEC-1849P, "Thermal-Hydraulic Analysis Methodology Using VIPRE-01 For PWR Applications, "October 1992

   Methodology for Specification:
   2.2.1 - Limiting Safety System Settings
   3.2.1 - AXIAL FLUX DIFFERENCE
   3.2.2 - Heat Flux Hot Channel Factor
   3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

7. YAEC-1854P, "Core Thermal Limit Protection Function Setpoint Methodology For Seabrook Station, "October 1992

   Methodology for Specification:
   2.2.1 - Limiting Safety System Settings
   3.1.3.5 - Shutdown Rod Insertion Limit
   3.1.3.6 - Control Rod Insertion Limits
   3.2.1 - AXIAL FLUX DIFFERENCE
   3.2.2 - Heat Flux Hot Channel Factor
   3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

8. YAEC-1856P, "System Transient Analysis Methodology Using RETRAN for PWR Applications," December 1992

   Methodology for Specification:
   2.2.1 - Limiting Safety System Settings
   3.1.1.3 - Moderator Temperature Coefficient
   3.1.3.5 - Shutdown Rod Insertion Limit
   3.1.3.6 - Control Rod Insertion Limits
   3.2.1 - AXIAL FLUX DIFFERENCE
   3.2.2 - Heat Flux Hot Channel Factor
   3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

9. YAEC-1752, "STAR Methodology Application for PWRs, Control Rod Ejection, Main Steam Line Break," October 1990

   Methodology for Specification:
   3.1.1.3 - Moderator Temperature Coefficient
   3.1.3.5 - Shutdown Rod Insertion Limit
   3.1.3.6 - Control Rod Insertion Limits
   3.2.1 - AXIAL FLUX DIFFERENCE
   3.2.2 - Heat Flux Hot Channel Factor
   3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

6.8.1.6.b. (Continued)

10. YAEC-1855P, "Seabrook Station Unit 1 Fixed Incore Detector System Analysis," October 1992

    Methodology for Specification:
    3.2.1 - AXIAL FLUX DIFFERENCE
    3.2.2 - Heat Flux Hot Channel Factor
    3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

11. YAEC-1624P, "Maine Yankee RPS Setpoint Methodology Using Statistical Combination of Uncertainties - Volume 1 - Prevention of Fuel Centerline Melt," March 1988

    Methodology for Specification:
    3.2.1 - AXIAL FLUX DIFFERENCE
    3.2.2 - Heat Flux Hot Channel Factor
    3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

6.8.1.6.c. The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as SHUTDOWN MARGIN, and transient and accident analysis limits) of the safety analysis are met. The CORE OPERATING LIMITS REPORT for each reload cycle, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, to the NRC Document Control Desk with copies to the Regional Administrator and the Resident Inspector.

cc:

Lillian M. Cuoco, Esq.
Senior Nuclear Counsel
Northeast Utilities Service Company
P.O. Box 270
Hartford, CT 06141-0270

Mr. Peter Brann
Assistant Attorney General
State House, Station #6
Augusta, ME 04333

Resident Inspector
U.S. Nuclear Regulatory Commission
Seabrook Nuclear Power Station
P.O. Box 1149
Seabrook, NH 03874

Jane Spector
Federal Energy Regulatory Commission
825 North Capital Street, N.E.
Room 8105
Washington, DC 20426

Town of Exeter
10 Front Street
Exeter, NH 03823

Mr. George L. Iverson, Director
New Hampshire Office of Emergency Management
State Office Park South
107 Pleasant Street
Concord, NH 03301

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Office of the Attorney General
One Ashburton Place
20th Floor
Boston, MA 02108

Board of Selectmen
Town of Amesbury
Town Hall
Amesbury, MA 01913

Mr. Dan McElhinney
Federal Emergency Management Agency
Region I
J.W. McCormack P.O. &
Courthouse Building, Room 401
Boston, MA 02109

Mr. Peter LaPorte, Director
ATTN: James Muckerheide
Massachusetts Emergency Management Agency
400 Worcester Road
P.O. Box 1496
Framingham, MA 01701-0317

Jeffrey Howard, Attorney General
G. Dana Bisbee, Deputy Attorney General
33 Capitol Street
Concord, NH 03301

Mr. D. M. Goebel
Vice President-Nuclear Oversight
Northeast Utilities Service Company
P. O. Box 270
Hartford, CT 06141-0270

Mr. J. K. Thayer
Recovery Officer, Nuclear Engineering and Support
Northeast Utilities Service Company
P.O. Box 128
Waterford, CT 06385

Mr. F. C. Rothen
Vice President - Nuclear Work Services
Northeast Utilities Service Company
P.O. Box 128
Waterford, CT 06385

Mr. A. M. Callendrello
Licensing Manager - Seabrook Station
North Atlantic Energy Service Corp.
P.O. Box 300
Seabrook, NH 03874

cc:

Mr. W. A. DiProfio
Nuclear Unit Director
Seabrook Station
North Atlantic Energy Service Corporation
P.O. Box 300
Seabrook, NH 03874

Mr. Frank W. Getman, Jr.
Cocheco Falls Millworks
100 Main Street, Suite 201
Dover, NH 03820

Mr. B. D. Kenyon
President - Nuclear Group
Northeast Utilities Service Group
P.O. Box 128
Waterford, CT 06385

Mr. B. L. Drawbridge
Executive Director Services &
Senior Site Officer
North Atlantic Energy Service Corp.
Seabrook, NH 03874

Mr. R. Hallisey - Director
Department of Public Health
Commonwealth of Massachusetts
Radiation Protection Program
305 South Street - 7th Floor
Jamaica Plain, MA 02130

Commonwealth of Massachusetts
SLO Designee
A. David Rodham - Director
Attn: Mr. James B. Muckerheide
State Nuclear Engineer
Massachusetts Emergency Mgmt. Agency
400 Worcester Road Box 1496
Framingham, MA 01701-0314

Diane Tefft - Administrator
Bureau of Radiological Health
Division of Public Health Services
6 Hazen Drive
Concord, NH 03301-6527

Mr. Thomas O'Connell
Radiation Scientist
Radiation Control Program
Department of Public Health
305 South Street, 7th Floor
Concord, NH 03301-6527

Mr. Peter S. Paiton, Supervisor
Emergency Response Section
Radiological Health Bureau
Division of Public Health Services
Health and Welfare Building
6 Hazen Drive
Concord, NH 03301-6527

Mr. M. Nawoj
Chief, Technical Hazards
State Civil Defense Agency
State Office Park South
107 Pleasant Street
Concord, NH 03001-6527