Washington Public Power Supply System
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Docket No. 50-508

G03-83-248
March 24, 1983

Director of Nuclear Reactor Regulation
ATTN: Mr. G. W. Knighton, Chief
Licensing Branch No. 3
Division of Licensing
US Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: NUCLEAR PROJECT 3
RESPONSES TO NRC ACCEPTANCE
REVIEW QUESTIONS (February 1983)

References:
a) Letter D. G. Eisenhut to
R. L. Ferguson, dated 08/20/82
b) Letter #G03-82-830
G. D. Bouchey to
H. R. Denton, date 08/20/82
c) Letter #G03-82-1085
G. D. Bouchey to
J. D. Kerrigan, dated 11/22/82

Reference a) transmitted a set of questions generated during the NRC's acceptance review of the WNP-3 Operating License Application (reference b). Reference c) represents the initial Supply System response to these questions and provided a schedule for those cases where our evaluations were not yet complete.

This letter transmits those responses scheduled to be provided for NRC review in February. In those cases where it is considered necessary or desirable to amend the FSAR due to our responses, we have provided marked up FSAR pages which show the changes which will be included in a subsequent amendment.

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RESPONSES TO NRC ACCEPTANCE REVIEW QUESTIONS

If you require additional information or clarification. the Supply System point of contact for this matter is Mr. K. W. Cook, Licensing Project Manager (206/482-4428 ext. 5436).

Sincerely,

GD Bouchey

G. D. Bouchey, Manager
Nuclear Safety and Regulatory Programs

AJM/ss

Attachments:
1. NRC Question 410.8 (9.2.5)
2. NRC Question 460.1 (11.5.2.4.2)
3. NRC Question 471.1 (12.3.4)
4. Request for Additional Information - Enclosure 4 (item 3) Safety Related Structures. Systems and components (Q-List) controlled by the QA Program.

cc: D. J. Chin - Ebasco NYO
N. S. Reynolds - D&L
E. F. Beckett - NPI
J. A. Adams - NESCO
D. Smithpeter - BPA
A. Vietti - NRC
Ebasco - Elma
WNP-3 Files - Richland
AA Tuzes - Comb. Engr.

ATTACHMENT 1

Question No.

410.8
(9.2.5)

Section 9.2.5 does not define the number of cells per cooling tower train, nor does it contain figures 9.2.5-2a through 9.2.5-2d. Confirm the date by which you intend to supply this information.

Response

As committed to in Letter G03-82-1085 dated Octob[illegible] 1982, the atta[illegible]d Figures 9.2.5-2a through 9.2.5-2c are provided f[illegible] The [illegible] heat release curve and sensible heat removal cu[illegible] this section (replaced by Shutdown Heat Exchang[illegible] this data is covered in Subsection 6.2.1 and 5.[illegible]

During normal operation the UHS in conjunction with the CCWS heat exchanger, can reject the maximum normal heat loads while maintaining CCWS temperature at or below 95F. Table 9.2.5-1 presents expected tower performance requirements during normal conditions. Parametric performance expectations for a range of ambient temperatures with and without heat rejection to the SWS under normal and emergency conditions is presented in Table 9.2.5-2. During emergency operation the UHS provides sufficient cooling to safety-related heat loads identified in Subsection 9.2.2. In the event of the loss of one train for any reason, the redundant train can reject the maximum instantaneous heat load. Non-essential heat loads can be manually aligned as reactor decay heat affords spare UHS capacity. The appropriate controls as discussed in Section 7.3 permit the operator to monitor and select these heat load alignments.

9.2.5.3 Safety Evaluation

The UHS, with one tower operating alone is designed to meet maximum heat load following an accident. Its long term heat rejection capacity is sufficient to mitigate a postulated LOCA and return the containment pressure and temperature to ambient conditions.

The results of an analysis of the 30-day period following a design basis accident are found in Tables 9.2.5-4 and 9.2.5-5 and Figures 9.2.5-2a through
C 9.2.5-2~~X~~. This analysis has determined the total heat rejected, the sensible 2
heat rejected, the station auxiliary system heat rejected, and the decay heat release from the reactor~~X~~. Details of this analysis are provided in sections 6.2.1 & 5.4.7.

As per Branch Technical Position ASB-9-2, the decay heat curves for fission products and for heavy elements were obtained using the assumptions and uncertainties set forth in the October 1973 draft proposed ANS standard, "Decay Energy Release Rates Following A Shutdown Of Uranium-Fueled Thermal Reactors" (ANS-5), to establish the heat input due to decay of radioactive material. An equilibrium fuel cycle and an increase in the calculated heat inputs were assumed as follows:

a) For the time interval 0 to 10^3 seconds, 20 percent was added to the heat released by the fission products to account for uncertainties in their nuclear properties.

b) For the time interval 10^3 to 10^7 seconds, 10 percent was added to the heat released by the fission products to account for uncertainties in their nuclear properties.

c) For the time interval 0 to 10^7 seconds, the heat released by the heavy elements was calculated (using the best estimate of the production rate for each unit) and 10 percent was added to account for uncertainties in their nuclear properties.

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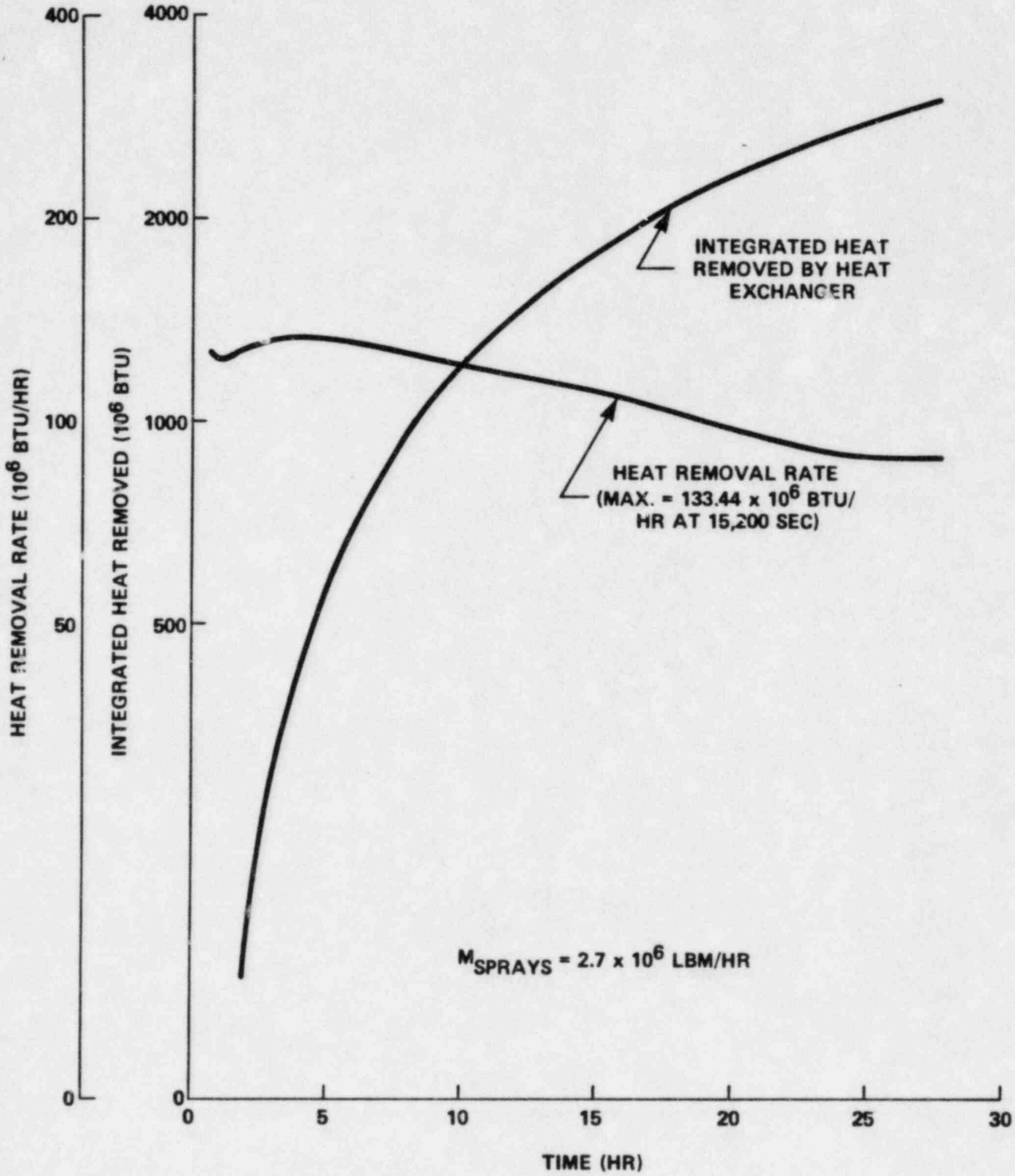
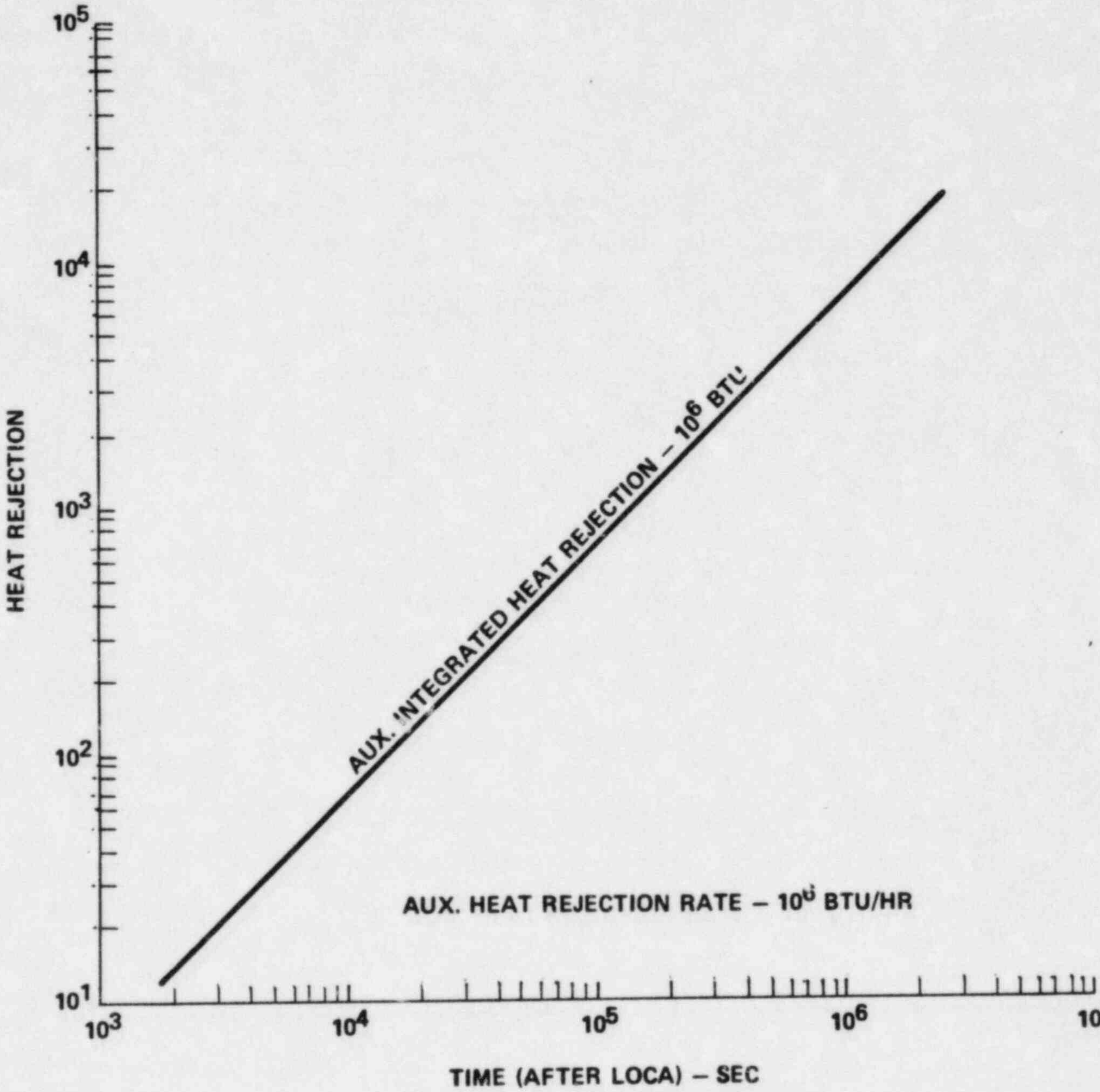
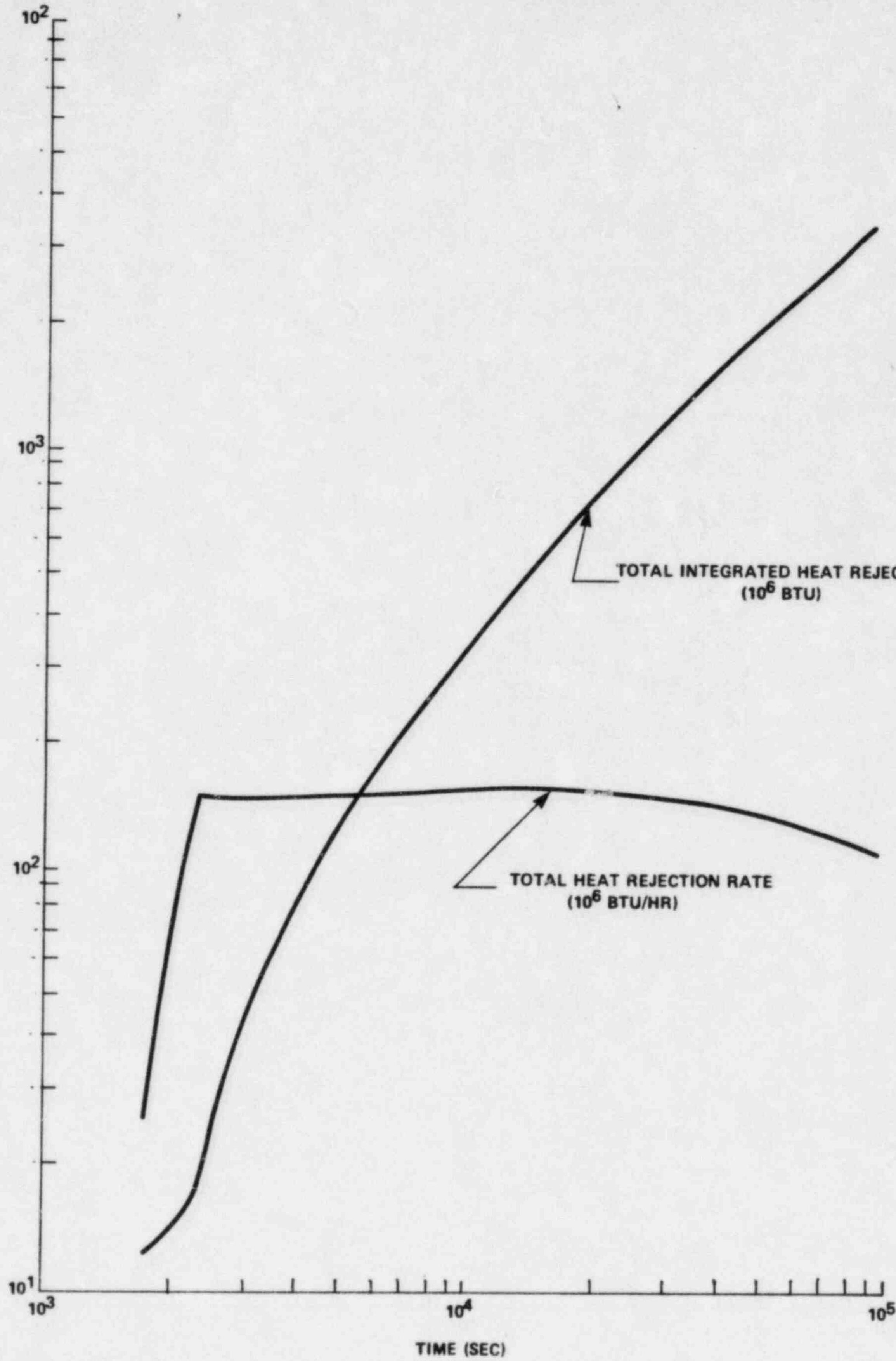


WASHINGTON PUBLIC POWER SUPPLY SYSTEM Nuclear Projects 3 & 5 FINAL SAFETY ANALYSIS REPORT	UNFOULED SHUTDOWN HEAT EXCHANGER HEAT REMOVAL (U = 392 BTU/HR - FT^2 ^{o}F) DESLS WITH MAX SI	FIGURE 9.2.5-2a



WASHINGTON PUBLIC POWER SUPPLY SYSTEM Nuclear Projects 3 & 5 FINAL SAFETY ANALYSIS REPORT	HEAT REJECTION DUE TO STATION AUXILIARY SYSTEMS	FIGURE 9.2.5-2b



WASHINGTON PUBLIC POWER SUPPLY SYSTEM Nuclear Projects 3 & 5 FINAL SAFETY ANALYSIS REPORT	TOTAL HEAT LOAD FOR COMPONENT COOLING WATER SYSTEM	FIGURE 9.2.5-2c

ATTACHMENT 2

Question No.

460.1
(11.5.2.4.2)

Supply information relating to the effluent radiation monitors for steam generator blowdown flash tank vent and steam seal gland steam condenser ventilation which the FSAR indicates as later or provide a schedule for submittal of this information.

Response

In letter GO3-82-1085 dated October 22, 1982, the Supply System committed to provide the sensitivity for the Steam Generator Blowdown Flash Tank Vent Radiation Monitor, Auxiliary Condensate Flash Tank Radiation Monitor, and the Steam Gland Seal Steam Exhaust Radiation Monitor.

This information is still unavailable. We expect to provide it by April 1983.

ATTACHMENT 3

Question No.

471.1
(12.3.4)

As per Regulatory Guide 1.70 indicate whether, and if so how, the guidance provided by Regulatory Guide 1.97 has been followed concerning area radiation and airborne radioactivity monitoring instrumentation. Reference or provide this information.

Response

The WNP-3 design for post-accident area radiation and airborne radioactivity monitoring instrumentation complies with USNRC Reg. Guide 1.97 Rev. 2 requirements with the following exceptions;

(1) Instruments monitoring radiation exposure rates inside buildings or areas which are in direct contact with primary containment, where penetrations and hatches are located (Type C variable) will not be supplied. An increase in radiation levels in these areas would be due primarily to streaming through the penetrations or to direct shine from the containment caused by elevated exposure rates inside the containment. Under these conditions, any additional increase in the radiation levels as a result of airborne radioactivity leaking from the containment would not be discriminated from streaming or direct shine by the area radiation monitors recommended by RG 1.97 Rev. 2. The four Auxiliary Building Airborne Radiation Monitors provided in the Reactor Auxiliary Building can detect airborne radioactive material leaking from the containment.

(2) In order to comply with the requirements for Area Radiation monitoring (Type E variables) in areas where access may be required to service or operate equipment important to safety, monitors have been placed at locations of WNP-3 which were identified and analyzed as vital areas in the WNP-3 TMI Shielding Study (See FSAR APP. 12A). All monitors supplied have the dynamic range recommended in RG 1.97 Rev. 2 (10^{-1} R/hr to 10^{4} R/hr) with the exception of the monitors supplied in the two Valve Operating Enclosure areas at elevation 351.00 ft of the RAB. According to the TMI Shielding Study the maximum dose rate expected in these areas after an accident is not more than 5 R/hr; therefore, the supplied monitors have a dynamic range 10^{-1} mR/hr to 10^{4} mR/hr.

The Supply System is considering the addition of an area radiation monitor for the Post Accident containment atmosphere monitoring and sampling area. The expected post accident dose rate in this area is appxoximately 0.7 R/hr and the monitor would have a range of 10^{-1} mR/hr to 10^{4} mR/hr. The staff will be informed as to the decision to add this monitor by June 1983.

ATTACHMENT 4

REQUEST FOR ADDITIONAL INFORMATION - ENCLOSURE 4

3) SAFETY-RELATED STRUCTURES, SYSTEMS AND COMPONENTS (Q-LIST) CONTROLLED BY THE QA PROGRAM

Staff requests for additional information regarding this issue have been sent to a number of OL applicants. A request from the Diablo Canyon review is provided as Enclosure 5.

Response

In our original response to this question (letter #G03-82-1085) we stated that the Supply System was in the process of developing the Q-list. We expected to have finalized this list by February 1983.

Due to the amount of time required to process this request we must extend our earlier schedule date for provision of the Q-list to June 1983.