PSNH PUBLIC SERVICE
Company of New Hampshire

SEABROOK STATION
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T.F. B 7.1.2

United States Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. Frank J. Miraglia, Chief
Licensing Branch #3
Division of Licensing

References: (a) Construction Permits CPPR-135 and CPPR-136, Docket Nos. 50-443 and 50-444
(b) USNRC Letter, dated February 12, 1982, "Request for Additional Information," F. J. Miraglia to W. C. Tallman

Subject: Responses to 471 Series RAIs; (Radiological Assessment Branch)

Dear Sir:

We have enclosed responses to the subject RAIs, which you forwarded in Reference (b).

Very truly yours,

YANKEE ATOMIC ELECTRIC COMPANY

John DeVincentis
Project Manager

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Enclosure

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Question 471.1
(FSAR Section 12.1)

This section states that Regulatory Guides 8.8 and 8.10 will be used as guidelines. Identify the individuals responsible for deciding which, if any, provisions will not be followed; provide examples of provisions which may not be followed, and the criteria for deciding not to follow them.

Response to 471.1

FSAR Section 12.1.1.1 states that "the Station Superintendent (since changed to Station Manager) has the overall responsibility and authority for implementing the ALARA philosophy. He delegates this responsibility and authority to the Health Physics Supervisor," (since changed to Health Physics Department Supervisor). This statement places the decision making authority with respect to ALARA concerns in the hands of the Station Manager. It also indicates that the Health Physics Department Supervisor is the Station Manager's key representative in overseeing and coordinating the functional aspects of ALARA philosophy.

FSAR Section 12.1.1 indicates that "what is 'reasonably achievable', for exposure reduction, is a judgement which all Seabrook Station management personnel will be required to make." This commitment reflects the policy of Seabrook Station to include pertinent management personnel in decisions relevant to their respective areas of concern.

Seabrook Station, in principle, accepts the provisions and intent of Regulatory Guides 8.8, Revision 3 and 8.10, Revision 1. Actual implementation of the provisions is conditional upon each application. Questionable situations or conditions are assessed individually with due regard for determining what is "reasonably achievable". Criteria for determining what is "reasonably achievable" involves "an assessment of the state of technology and the economics of improvements in relation to all of the benefits from these improvements" as specified in FSAR Section 12.1.1.

Specifically, such assessments include: effects on plant and/or personnel safety; effects on plant operability/availability; significance of exposure reduction versus implementation and maintenance exposure; technical feasibility; and a cost/benefit analysis as necessary.

Thus, it is the responsibility of all Seabrook managers and supervisors to make decisions and take action on ALARA related matters within their purview. ALARA decisions having potentially significant and/or widespread impact are evaluated with regard to "what is reasonably achievable" by all affected management. If necessary, final decisions are rendered by the Station Manager.

Note: Section 12.1 will be changed to reflect the current position titles of station personnel.

Question 471.2
(FSAR Section 12.1)

This section identifies specific responsibilities that may be accorded to the Station Superintendent. As specified in Regulatory Guide 1.70, Section 12.1.1, you should describe the applicable responsibilities and the related activities to be conducted by the management individuals having responsibility for radiation protection and the policy of maintaining occupational radiation exposures ALARA. Identify the individual(s) who will be delegated responsibility for such activities.

Response to 471.2

Refer to FSAR Section 12.5.1 for a description of the Health Physics Department organization and responsibilities with regard to radiation protection.

Refer to Question 470.8 response, "Compliance with Regulatory Guide 8.8" for delineation of authority and responsibilities regarding ALARA activities.

Question 471.3
(FSAR Section 12.1.1.2)

This section identifies ALARA responsibilities for members of the Seabrook Station management organization, which can be accomplished through careful consideration of certain guidelines. As specified in Regulatory Guide 1.70, Section 12.1.1, you should describe the applicable responsibilities and the related activities to be conducted by the management individuals having responsibility for radiation protection and the policy of maintaining occupational radiation exposures ALARA. Identify the individual(s) responsible for deciding which, if any, provisions will not be carried out; provide examples of guidelines which may not be followed and criteria for deciding not to follow them.

Response to 471.3

Refer to Question 471.1 and 471.2 responses.

Question 471.4
FSAR Sections
12.1.2 & 12.3.1.3(a)

This section states that design reviews are performed; section 12.3.1.3(a) states that periodic ALARA reviews of plant design and equipment arrangement are performed. As specified in Regulatory Guide 1.70, Section 12.1.2, you should describe any mechanisms that provide for design review by a competent professional in radiation protection.

a. Describe the nature of these reviews as they relate to design changes, construction, and field run piping.

b. Identify by title the individual(s) responsible for the radiation protection design review, and describe how they relate to the individual responsible for the overall design.

c. Provide a breakdown by title of radiation protection personnel who have been or will be participating in these reviews, tabulating the health physics education and experience required of each.

d. Describe formal arrangements and procedures for assuring that adequate, independent radiation protection reviews are performed throughout the design and construction processes, and that adequate records are kept to document the completion of each such review.

Response to 471.4

The following discussion responds to this question and will be added to FSAR Section 12.1.2 in a future amendment:

12.1.2.5 Management of Radiation Protection Design Review

The Seabrook Station ALARA program for construction, design changes and reviewing field run piping is the joint responsibility of Westinghouse Electric Corporation (Westinghouse), United Engineers and Constructors Inc. (UE&C), Yankee Atomic Electric Company - Nuclear Services Division (YAEC), and Public Service Company of New Hampshire (PSNH).

Westinghouse is responsible for the design, fabrication and delivery of the nuclear steam supply system, related auxiliary systems and the nuclear fuel. Technical direction for the installation of this equipment and technical assistance throughout the pre-operational testing, initial core loading and testing programs are further responsibilities of Westinghouse.

United Engineers and Constructors (UE&C) is responsible for the engineering, design and construction management of the station. Included in their scope is the supply and installation of the balance of plant systems, components, and structures such that a complete and integrated facility is assured.

The radiation design review performed by UE&C is the responsibility of the Chief Power Engineer. The Chief Power Engineer ensures that an overall design program is implemented to help maintain occupational radiation exposures As Low As Reasonably Achievable (ALARA) during operation of the facility. He provides appropriate guidance to Chief Discipline Engineers regarding ALARA design implementation and verifies implementation.

The Chief Discipline Engineers (electrical, mechanical, instrumentation, etc.) provide for incorporation of ALARA considerations. This is accomplished by providing guidance to engineers responsible for the design of Seabrook Station. The Chief Discipline Engineers or designees review the various systems to ensure provided guidance is used.

In the case of Seabrook Station, the Yankee Nuclear Services Division (YNSD) functions as the Engineering Department for PSNH. In this position, YNSD carries the responsibility and authority required to act on all engineering related matters. This is the same position YNSD occupied on the Yankee Nuclear Power Station project, the Connecticut Yankee Haddam Neck project, the Vermont Yankee project and the Maine Yankee project. Although the vehicle for ownership of Seabrook Station is not identical to the predecessor Yankee projects, the role of YAEC in the project is unchanged from the one it had on those earlier projects.

The overall responsibility for coordinating YNSD activities between PSNH and YNSD is assigned to the YAEC Office of the Vice President. The responsibilities for directing the YNSD engineering and construction activities on Seabrook Station are assigned to the YNSD Project Manager and Construction Manager, respectively.

The PSNH Seabrook Station Quality Assurance Manual states that the Vice President - Operations of YAEC is responsible for reviews of selected plant system specifications and drawings for plant operability and maintainability. The Quality Assurance Manual also states contractor responsibilities are as follows:

> "Each contractor shall maintain design control measures as required by ANSI N45.2.11. These design measures shall be applied to areas such as the following: ...accessibility for inservice inspection, maintenance and repair...".

The YNSD Project Office establishes appropriate reviewers as determined by the Quality Assurance Manual and Subsection 17.1 of the Seabrook Station FSAR. Project policies indicate primary and secondary reviewers of UE&C specifications; Westinghouse specifications; UE&C system descriptions; Westinghouse system descriptions; FSAR chapters sections, and subsections, engineering changes and general arrangement drawings of the containment,

fuel storage and primary auxiliary building. Project policies also indicate the type of documentation required for reviews. The documentation may be in the form of an Engineering Review Report, a memorandum or other report.

12.1.2.5.1 Design Reviewers

YNSD Radiation Protection personnel do not always possess the necessary expertise to perform complete ALARA reviews. The YNSD Radiation Protection Group relies on engineers with the expertise to determine equipment compatability, accessibility (ladders, platforms, laydown space), operability and maintenance history (low-maintenance). Individuals performing reviews are usually Engineer grade or higher (B.S. degree or equivalent and 3 years professional experience). Individuals within departments who perform the reviews are choosen based on their general knowledge of the system and equipment. The departments within YNSD who perform reviews are listed below with some of their responsibilities:

a. Plant Engineering Department (Instrumentation and Control Group, Electrical Engineering Group, Mechanical Engineering Group and Systems Engineering Group)

- Support the Project Office in general and detailed technical review and guidance for plant concept, design construction and licensing in the field of Fluid Systems, Instrumentation and Control, Electrical Engineering, Mechanical Engineering, Systems Engineering, Materials Engineering and Structural Engineering.

- Coordinate electrical and control design between the architect-engineer and nuclear steam system supplier.

- Review conceptual design and detailed engineering of all assigned primary and secondary fluid systems including types of components selected, modes of operation and physical arrangements.

- Review all electrical and control equipment specifications, logic and wiring diagrams. These reviews include transformers, motors and switchgear, plant control devices, nuclear instrumentation and reactor control and protection system equipment.

b. Nuclear Engineering Department (PWR Transient Analysis Group, Reactor Physics Group and LOCA Analysis Group)

- Review the reactor physics design work performed by the reactor supplier to assure adherence to the design criteria and the use of adequate methods and assumptions.

- Review and/or participate in the design of Instrumentation for core monitoring.

Question 471.4 (continued)

- Verify that operational requirements are given adequate consideration and are appropriately factored into the design.

- Assist in the development of Technical Specifications and station operating procedures for accident conditions.

- Review the various station anticipated transients and accidents in order to ensure conformance with all applicable criteria.

- Review and analyze data from the station to verify the reactor physics design.

c. Fuel Management Department (Nuclear Materials Group, Economic Analysis Group and Core Components Group)

- Review and approve mechanical designs and specifications for nuclear fuel assemblies and components.

- Review specifications, procedures, purchase orders and drawings for proper definition of quality assurance requirements for nuclear fuel assemblies.

d. Environmental Engineering Department (Radiological Engineering Group, Radiation Protection Group, Environmental Sciences Group and Environmental Laboratory)

- Establish functional requirements of engineered safeguard systems and evaluation of their performance.

- Participate in the design and review of solid, liquid and gaseous radioactive waste treatment systems.

- Participate in the establishment of system design requirements for plant process and area radiation monitoring systems.

- Participate in the establishment of Station Radiological Equipment and Facilities.

12.1.2.6 Independent Reviews

The ongoing interactions between PSNH, YAEC, Westinghouse and UE&C engineers and radiation protection personnel results in continuous cross-checking or "independent" reviews of each organization's design, construction and operational activities. Proposals to modify or establish designs receive appropriate levels of review by these diverse organizations.

Additionally, these organizations recognize that professional contractor organizations are available for use, as necessary, to provide assistance in specialized areas of radiation protection.

Question 471.4 (continued)

Contractor organizations have already been used to provide specialized evaluations in such cases as the analysis of the proposed removeable shielding for the reactor vessel annulus and the assessment of the radiological impact of the spent resin transfer system design. Such special evaluations significantly contribute to design review efforts directed toward ensuring that occupational exposures are maintained ALARA.

12.1.2.7 Field Reviews

The latter example, cited in section 12.1.2.6, is an instance of the on-site ALARA reviews being conducted during the construction phase under the auspices of the Seabrook Station Health Physics Department. These evaluations are performed on systems, components and areas in accordance with station ALARA goals.

These evaluations are used to identify potential beneficial modifications as well as to provide background information for use during operations. Expertise and assistance is obtained, as necessary, from other applicable station departments.

Coordination of major actions and the final decisions are the responsibility of appropriate station and corporate management. YNSD and, when necessary, UE&C are party to these station activities.

Day-to-day aspects of this ALARA effort are conducted by station Health Physics Technicians who are supervised and coordinated by a Health Physics Working Foreman under the cognizance of a Health Physicist. Minimum experience and educational qualifications for Health Physics Department personnel are described in FSAR, Section 12.5.1.

The on-site ALARA field review program, in concert with ALARA design considerations addressed by Westinghouse, UE&C, YNSD and selected professional contractors, as discussed in previous sections, is being utilized by Seabrook Management to ensure that the construction and operation of Seabrook Station will result in occupational radiation exposures that are as low as reasonably achievable.

Question 471.5
(FSAR Section 12.3)

As specified in Regulatory Guide 8.8, you should attain the objectives in Section C to provide reasonable assurance that exposures of station personnel to radiation will be ALARA, throughout the plant, from design planning and design through decommissioning. Describe the features that have been incorporated into the Seabrook design to maintain doses ALARA during eventual decommissioning.

Response to 471.5

Seabrook Station will rely on preparations in several areas to ensure that occupational radiation exposures are maintained as low as reasonably achievable (ALARA) during plant operations and decommissioning.

An important aspect of the decommissioning procedure will be the use of specific ALARA practices tailored to deal with the particular decommissioning method employed. Delineation of these specific ALARA practices (including engineering design modifications) will take place during decommissioning planning, after information concerning the specific decommissioning method becomes available. Consistent with the guidance provided by 10 CFR 20 and Regulatory Guide 8.8, Revision 3, the specific practices implemented will be based on "an assessment of the state of technology and economic considerations" prevalent at the time of decommissioning. The state of technology and the economics that will prevail 40 years in the future are unknown factors and, therefore, performance of a cost benefit analysis is precluded at this time.

Many basic ALARA design features incorporated into Seabrook Station for operations, maintenance and refueling will enhance exposure reduction during those phases as well as at decommissioning, regardless of the specific decommissioning procedure. In effect, this is a generic, ALARA approach to operations and decommissioning.

Specific design features that will be used to maintain occupational radiation exposures ALARA include the following:

- A separate building exists for waste processing and disposal. This will ensure availability of waste processing facilities while other systems and components are being maintained or dismantled.

- A full capacity polar crane exists. It is capable of removing the reactor vessel, steam generators and pressurizer with minimal displacement of permanent concrete shielding to afford its maximum effectiveness.

- Seabrook containments are equipped with 27 foot diameter equipment hatches that will facilitate removal of large equipment intact.

Generic design features that will be used to maintain occupational radiation exposures ALARA include the following:

Question 471.5 (continued)

- Location of Equipment: As stated in FSAR Section 12.1.2.1, paragraph (4), "Equipment location is used, where practical, to eliminate exposure by placing equipment in non-radiation control areas." This philosophy is embodied in the segregation of areas with radioactive systems and components. The design philosophy "to minimize the extent of areas housing radioactive equipment and piping through efficient arrangement of equipment and systems" is expressed in FSAR Section 12.3.1 (b).

- Equipment Accessibility and Removability: FSAR Section 12.1.2.1 indicates that equipment is designed and located to maximize accessibility so as to facilitate rapid, efficient work. Equipment is also designed and placed so as to enhance removal operations and thus, minimize exposure time.

- Plant Layout: The FSAR Section 12.3.1.3 (a) explains that "plant layout includes optimal location of radioactive components.

- Flush/Drain Connections: The provision of flush and drain connections on many systems and components will enable extensive chemical decontamination prior to operating phase maintenance and later, decommissioning.
- Corrosion Control: Internal accumulation of radioactive material is limited through effective corrosion control methods. Careful selection of plant materials and an aggressive chemistry control program greatly reduces source terms that must be dealt with during operating phase maintenance and decommissioning.

Seabrook Station has indicated, in response to RAI 470.8, that Regulatory Guides 8.8, Revision 3 and 8.10, Revision 1, will be used as guidelines in the formulation and implementation of the station-wide operating policies as well as the Station Radiation Projection Program (FSAR Section 12.1.1). This commitment includes an acknowledgement and understanding that the process of preparing for eventual decommissioning with occupational exposure as low as reasonably achievable is ongoing in nature.

As indicated above, Seabrook possesses design features that will offer significant exposure reduction during decommissioning. These design features are merely the basis for the performance of an ALARA-oriented decommissioning. An effective, ALARA-oriented radiation protection program during plant operations as well as effective ALARA preparations during decommissioning planning are all important contributing factors.

Question 471.6
FSAR Section (12.3.2.1)

As specified in Regulatory Guide 1.70, Section 12.3.1.2, you should describe any special protective features that use shielding, geometric arrangement, or remote handling to assure that ORE are ALARA. Figure 1.2-3 depicts personnel access to space immediately adjacent to the spent fuel transfer tube, just inside containment at radius 10. Describe precautions taken to prevent inadvertent personnel access to all unshielded potentially very high radiation areas in the vicinity of the spent fuel transfer tube. It is our position that all accessible portions of the spent fuel transfer tube and or canal must be shielded during fuel transfer. Use of removable shielding for this purpose is acceptable. This shielding shall be such that the resultant contact radiation levels shall be no greater than 100 rads per hour. All accessible portions of the spent fuel transfer tube shall be clearly marked with a sign stating that potentially lethal radiation fields are possible during fuel transfer. If removable shielding is used for the fuel transfer tubes, it must also be explicitly marked as above. If other than permanent shielding is used, local audible and visible alarming radiation monitors must be installed to alert personnel if temporary fuel transfer tube shielding is removed during fuel transfer operations.

Response to 471.6

NRC Bulletin 78-08, "Radiation Levels from Fuel Element Transfer Tubes" described an incident in which individuals were exposed to the radiation from a bare fuel transfer tube. At that time action was taken to ensure that shielding around the tube and canal at the Seabrook Station was sufficient to protect personnel.

All accessible areas around the tube and canal are shielded. All shields were designed for a contact radiation dose rate of less than 100 mr/hr. Four inches of lead plate were added between the liner and concrete at the bottom of the canal. In the enclosure building a shield box was designed around the tube. This box consists of approximately 300 bricks weighing 50 pounds each. These bricks will be explicitly marked with a sign stating that potentially lethal radiation fields are possible if the bricks are removed during fuel transfer. The access point noted in Figure 1.2-3 is an inspection hatch (manway). This hatch is shielded with a three-foot concrete plug. This plug shall also be marked as noted above.

Question 471.7
FSAR Section (12.3.4.2(a))

As specified in NUREG-0800, Section 12.3.II.4.b.1, the monitoring system should be capable of detecting ten MPC-hours of particulate and iodine radioactivity from any compartment which has a possibility of containing airborne radioactivity and which normally may be occupied by personnel, taking into account dilution in the ventilation system. This section states that certain monitored points are within the ventilation system, and in ventilation exhaust ducts. Specify the number and locations of monitors, shielding dilution factors from various compartments, personnel occupancy times, background radiation levels and detector and collector efficiency which will indicate 10MPC-hours (particulate iodine, or gas) in any compartment (or area) for which the monitor is applicable. Discuss the criteria for the circumstances in which such monitoring will be provided.

Response to 471.7

The fuel storage building ventilation exhaust is monitored by a GM detector in the exhaust duct. The PAB miscellaneous ventilation is also monitored by a GM detector in the exhaust duct.

In addition, the waste processing building and the PAB exhaust air are monitored by skid mounted particulate, iodine and gaseous monitors. The particulate filter will have a collection efficiency of at least 99% for particulates 0.3 microns or larger in diameter. The charcoal cartridge will have a collection efficiency of at least 95% for iodine.

The PAB and waste processing buildings contain numerous cubicles. These buildings along with the fuel storage building are not normally occupied. When work is performed in these areas portable continuous air monitors (CAM's) will be used, as appropriate, to monitor for particulates and noble gases. In addition, appropriate grab samples (particulate, noble gas, iodine) will be taken to ensure proper measures are taken to protect personnel.

Station health physics procedures will specify under what circumstances air sampling will be performed. Such circumstances may include opening contaminated components and grinding or welding on such components.

Question 471.8
FSAR Section (12.5.I)

This section states that one of the health physicists may temporarily assume the responsibilities of the Health Physics Department Supervisor (RPM), for an extended period of time. Section II.A.2 of NUREG-0731 states that there should be in-depth experience at the Radiation Protection Manager level. Section 4.4.4 of the December 1980 draft ANSI 3.1, which will be adopted by Regulatory Guide 1.8 specifies that an individual who temporarily replaces the RPM should, as a minimum, have a B.S. degree in science or engineering and two years experience, six months of which should be onsite. You should specify the qualification for the individual who will temporarily replace the Health Physics Department Supervisor.

Response to 471.8

When selecting an individual to temporarily replace the Health Physics Department Supervisor, Seabrook Station will assign an individual who will have as a minimum, a B.S. degree or equivalent, in a related science on engineering discipline and two years experience in health physics, six months of which will be onsite.

Question 471.9

Provide commitments to conform to the provisions of the following Regulatory Guides, or describe alternative measures to be taken to provide a comparable degree of worker protection:

8.6	8.13	8.26	8.29
8.7	8.20	8.27	1.97

Response to 471.9

In regards to Regulatory Guide 8.6 "Standard Test Procedure for Geiger-Muller Counters":

When purchasing Geiger-Muller counters, Seabrook Station will require that Geiger-Muller tube manufacturers comply with the provisions of Regulatory Guide 8.6. If a manufacturer does not comply with a specific provision of the Regulatory Guide, that manufacturer will be required to describe the comparable alternative measures taken.

All tests that are performed on site to ensure the response of Geiger-Muller counters will be in accordance with the intent of Regulatory Guide 8.6, with the exceptions that gamma sources (other than Co-60) may be used and source geometries other than "unshielded and uncollimated" may be used.

In regards to Regulatory Guide 8.7, "Occupational Radiation Exposure Records Systems":

Regulatory Guide 8.7, "Occupational Radiation Exposure Records Systems" endorses the recommendations of the American National Standards Institute document N13.6 - 1966 (R1972), "American National Standard Practice for Occupational Radiation Exposure Records Systems" as generally acceptable and indicates that the recommendations provide an adequate basis for complying with the pertinent record-keeping requirements of 10 CFR Part 20.

Accordingly, Seabrook Station intends to use the endorsed standard as general guidance during the establishment of its occupational radiation exposure records system. Specifically, the system will contain the following:

1. Radiation exposure records will be maintained in accordance with 10 CFR 20.401.

2. Comprehensive records will be generated and retained including:

    a. Radiation exposure records that are related to an individual such as:

        . bioassay data

Question 471.9 (continued)

- dosimetry discrepancy/investigation records
- results from individually worn TLD's

b. Records related to the radiological status of work areas such as:

- air sampling results
- radiation survey results
- contamination survey results

c. Records describing the technical and administrative bases for the radiation protection program such as:

- standards
- policies
- procedures and evaluation methods

3. Each individual (radiation worker) will have records established and maintained for:

a. external radiation exposure

b. internal bioassay data, where bioassay is defined as the determination of the kind, quantity or concentration and location of radioactive material in the human body by direct (in vivo) measurements or by analysis of materials excreted or removed from the body.

c. supplementary information such as investigation reports.

4. Each individual (radiation worker) will be identified according to a unique identification system using a combination of name, social security number, badge number, birthdate and sex.

5. Each individual (radiation worker), prior to authorization to receive radiation exposure in excess of 1.25 rem/quarter will have all previous occupational radiation exposure summarized and documented.

Each individual (radiation worker), prior to authorization to receive radiation exposure in excess of 0.312 rem/quarter but less than 1.25 rem/quarter will have all current quarter occupational radiation exposure summarized and documented or will have a signed estimate thereof.

6. A visitor log sheet will be maintained for individuals who enter the Radiation Control Area (RCA) and have not been authorized to receive radiation exposure in excess of 0.312 rem/quarter as measured by a self-reading pocket dosimeter.

Question 471.9 (continued)

7. Records will be maintained that indicate that the response of the dosimeters (TLD) used to measure personnel radiation exposure provide an accurate estimate of the dose received. Successful completion of the soon to be implemented National Voluntary Laboratory Accreditation Program (NVLAP) will ensure the accuracy of TLD exposure readings.

8. The following information will be available from an individual's and TLD data records:

   a. identification of the TLD wearer;

   b. period of exposure;

   c. control and calibration factors; and

   d. notation of operational abnormalities.

9. Results of the TLD badges (or other official dose determination methods) will be added to the individual's current exposure to facilitate comparison with Seabrook Station radiation exposure administrative guides and regulatory limits.

10. The Seabrook Station Radiation Protection Program will specify what types and when TLD dosimeters shall be worn.

11. If a TLD badge yields an invalid result, an estimate of the exposure will be recorded. The record of the investigation will include (when appropriate) the following:

   a. identification of the individual;

   b. dates involved;

   c. nature of abnormality;

   d. location and tasks to which individual was assigned;

   e. readings of other dosimeters worn by individual;

   f. dose received by others in work party;

   g. conclusion as to magnitude and type of occupational exposure;

   h. signature of individual and/or supervisor; and

   i. signature of investigator.

12. Records of internal exposure will include one of the following:

   a. Whole Body Count records;

      . identification of individual and date

Question 471.9 (continued)

- identity and location of radionuclide and magnitude of body burden (organ burden)
- quantitative output counting data

b. In vitro analysis;

- identification of individual and date
- date of suspected intake
- collection period
- sample information (type, size)
- identity of radionuclide
- laboratory performing analysis

c. If interpretation was made from the above results, records will also include;

- bioassay results
- assumptions and calculations used in the analysis
- assigned body burden and/or dose

d. If internal exposure is determined by air sample(s) and exposure time(s), records will include;

- identification of exposure by individual
- period of exposure (from the Radiation Work Permit (RWP))
- cross reference of the RWP and radiation survey (when appropriate)
- estimate of internal exposure (i.e., intake)

13. Records of unusual events involving potential or actual exposure will be recorded such as:

a. exposure in excess of regulatory limits;

b. suspected or actual exposures (external or internal); and

c. use of special or emergency exposure limits.

The information, if applicable to the event, that will be documented is as follows:

Question 471.9 (continued)

- individual(s) involved
- time, date and location of event
- description of the event
- results of the event
- probable cause of the event
- action taken at the time of event
- reference to action to prevent or avoid recurrence of event
- identification of investigators

14. Records of individual training received will be maintained.

15. Records of medical services provided following injuries will be maintained at the medical facility where service was obtained.

16. Records of the program and/or procedures to maintain occupational radiation exposure to individuals "As Low As Reasonably Achievable" will be maintained.

17. Radiation Work Permits (RWP's) will be used to monitor and control work activities in the Radiation Control Area (RCA) as necessary. Records or information associated with the RWP's that will be retained include:

    a. effective dates

    b. identity of personnel authorized to perform the work

    c. location of the work

    d. description of the work

    e. procedures, instructions and/or precautions to be observed

    f. protective equipment/apparel required

    g. dosimetry required

    h. signatures of individuals authorizing RWP

    i. radiological conditions of the work area

18. Radiation and contamination survey records will be maintained and include the following information:

Question 471.9 (continued)

a. date and time of survey;

b. survey location;

c. specific location or object where measurements or sampling performed;

19. Area monitoring instrumentation and continuous air monitors will meet the intent of ANSI 13.6-1966 (R1972) in terms of record maintenance. Air sampling records will include the following:

a. date and time of sample

b. location of sample

c. type of sample collected

d. instrument identification

e. collection efficiency (by procedure)

f. flow rate, duration and total air volume

g. counting (analysis) data

h. calculated airborne concentration

i. identification of individual performing survey and/or analysis

20. Radiation protection policies are identified in various documents and will be maintained in the Final Safety Analysis Report, administrative and operational procedures or Station directives.

21. Health Physics administrative and operational procedures will be maintained as controlled documents.

22. Capabilities of TLDs will be assured through the accreditation and dosimetry program. Results of dosimeter calibrations will be maintained and performance assured according to criteria established in health physics operational procedures. A calibration schedule will be adhered to according to procedure.

23. Radioactive sources used for calibration will be traceable to the National Bureau of Standards (NBS), or to an approved laboratory accredited or recognized by the NBS.

24. Results and records of health physics quality surveillance activities will be maintained.

25. Retention periods for records will be, as a minimum, established in accordance with applicable Federal regulations.

26. Storage of individual exposure data records will rely on a combination of computer storage, microform, and fire-proofed cabinets.

In regards to Regulatory Guide 8.13, "Instruction Concerning Pre-Natal Radiation Exposure":

Seabrook Station will comply with the provisions of Regulatory Guide 8.13, Revision 1, by presenting instruction concerning pre-natal radiation exposure to personnel during Radiation Worker Training.

In regards to Regulatory Guide 8.20, Revision 1, "Applications of Bioassay for I-125 and I-131":

Refer to the response to Question 471.9 regarding Regulatory Guide 8.26, "Applications of Bioassay for Fission and Activation Products" that follows.

In regards to Regulatory Guide 8.26, 1980, "Applications of Bioassay for Fission and Activation Products"

A bioassay program for fission and activation products will be maintained at Seabrook Station as discussed below. The program is based on Regulatory Guide 8.26 "Applications of Bioassay for Fission and Activation Products", recommendations of the document ANSI N343-1978 "American National Standard for Internal Dosimetry for Mixed Fission and Activation Products" and alternative measures that are considered to provide comparable protection.

A summary of the bioassay program is as follows:

1. Seabrook Station will have an internal dosimetry program which will provide measurements for estimating the quantity of fission and activation products deposited in body organs in order to establish a basis for judgements that significant depositions have or have not occurred.

2. The Seabrook Station health physics department staff will be responsible for;

   a. designating the areas in the facility in which routine air sampling or air monitoring is necessary for purposes of internal exposure control;

   b. establishing and maintaining in designated areas a routine air sampling or air monitoring program; and

   c. establishing and maintaining an internal dose assessment program which includes appropriate bioassay procedures.

3. Regarding participation in the program, all facility and contract personnel who routinely enter airborne radioactivity areas for operations or maintenance work will be scheduled for a baseline, final and yearly (if applicable) in vivo measurement. For non-routine entries, a health physicist or designee will determine the need for measurement on a case-by-case basis.

Question 471.9 (continued)

4. An excreta bioassay will be performed in any special situation where this bioassay is considered necessary by a health physicist for reasons of compliance and/or backup information and documentation. The need for excreta bioassay may be stimulated by any of or a combination of the following:

   a positive invivo measurement

   b. elevated air sample results

   c. chemical/physical form of the nuclide

   d. metabolic data and recommendations of current, accepted standards (ICRP, NCRP, ANSI, MIRD).

5. Additional bioassays will be performed based upon evaluations of the following:

   a. radiological conditions during a work activity;

   b. nasal swab results;

   c. internal exposure in excess of 40 MPC-hours in any seven consecutive days;

   d. any accidental internal exposure whether real or suspected;

   e. an investigation level of 10% MPBB (or MPOB) is reached where the MPBB is defined per ICRP Publication #2;

   f. need to verify effectiveness of internal exposure control methods.

6. Results of bioassay data will be used to determine the following:

   a. work restrictions;

   b. requirement of therapeutic treatment;

   c. need to improve/upgrade respiratory protection or air sampling/monitoring procedures/programs;

   d. need for additional training; and

   e. need to evaluate other members of a work party or work location.

7. All methods of internal dose assessments will be based on currently accepted models and will be clearly referenced and recorded.

8. The collection and handling of excreta samples will be performed in accordance with ANSI N343-1978.

9. Excreta samples will be evaluated by a licensed analytical laboratory.

Question 471.9 (continued)

10. The in vivo system detector(s) will be sufficiently shielded and located to allow measurements of 5% MPBB for at least 95% of the in vivo measurements performed.

11. The precision of the in vivo system will be empirically determined for selected phantom-activity combinations by generating populations of replicate measurements and by calculating the standard of these distributions.

12. Each in vivo estimate will be reported as a radioactivity value with an indication of the confidence in that value.

13. The influence of external contamination on personnel during in vivo counting is recognized and will be controlled as necessary. Contamination "free" clothing will be worn during the measurement and objects containing radioactive materials such as radium dial watches will be removed prior to counting.

14. Health physics quality surveillance activities to verify operation and results of the in vivo measurement equipment will be established and include the following:

    a. written calibration procedures;

    b. phantom measurements;

    c. performance checks; and

    d. periodic review of in vivo counting data by a health physicist or designee.

15. Records will be maintained according to the response to Question 471.9 regarding Regulatory Guide 8.7.

16. The bioassay program will be updated in a timely fashion when new methodolgies are endorsed by ANSI and the NRC. This description of the bioassay program does not preclude Seabrook Station from using more up-to-date models where they result in more realistic and accurate assessment of internal exposures.

In regards to Regulatory Guide 8.27, "Radiation Protection Training and Personnel at Light-Water-Cooled Nuclear Power Plants":

Seabrook Station will comply with the intent of Regulatory Guide 8.27.

Extent of worker training varies, depending on previous experience, extent of worker supervision and nature of work. Personnel having essentially unlimited access to all plant areas receive up to 40 hours of radiation protection training (e.g., operators, health physics technicians, chemistry technicians).

Question 471.9 (continued)

Generic "practical factors" are included in training classes for personnel assigned to work inside Radiation Control Areas. Seabrook Training Department procedures define specific practical factors used in this training.

On-the-job instruction and performance testing will be provided on a case-by-case basis. Decisions for use of on-the-job training and performance testing are based, primarily, on ALARA considerations.

Implementation of special training evolutions and special training devices (such as mock-ups) is also dictated by ALARA considerations. Health Physics Department ALARA procedures provide guidance for the evaluation of the cost effectiveness (i.e., what is reasonably achievable?) of additional training evolutions and equipment.

In regards to Regulatory Guide 8.29, "Instruction Concerning Risks from Occupational Radiation Exposure":

Seabrook Station will comply with the provisions of Regulatory Guide 8.29. The contents of Reg. Guide 8.29 will be orally presented to all occupational radiation workers during Radiation Worker Training. Copies of Reg. Guide 8.29 will be made available for inspection at this time.

In regards to Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Access Plant and Environs Conditions During and Following an Accident":

Table 2, section E of Regulatory Guide 1.97 requires high range area radiation monitors inside buildings or areas where access is required to service equipment important to safety. This requirement is addressed in the response to Question 471.13. No other requirements regarding "measures to be taken to provide a comparable degree of worker protection" were noted in Reg. Guide 1.97.

Question 471.10
FSAR Section (12.5.2.5)

As specified in Regulatory Guide 1.70, Section 12.5.II.B.5, you should provide specified personnel protection equipment. Describe the kinds and quantities of respiratory protection equipment and supplies to be provided.

Response to 471.10

Seabrook Station will maintain full-face respirators, self-contained breathing apparatus (SCBA), air-fed hoods, airlines, filter cartridges and all necessary repair parts and support accessories for use as required in the radiological respiratory protection program. Initially, Seabrook will purchase and maintain five hundred (500) full-face respirators, fifteen (15) SCBA's, twenty four (24) spare SCBA bottles and two hundred (200) air-fed hoods.

Question 471.11

Provide the information requested in II.B.2, II.F.1(3) and III.D.3.3 of NUREG-0737, "Clarification of TMI Action Plan Requirements".

Response to 471.11

Refer to letter dated February 12, 1982, from PSNH to USNRC, Attn: Frank Miraglia, a copy of which is attached.

ATTACHMENT TO RAI 471.11

PSNH PUBLIC SERVICE
Company of New Hampshire

SEABROOK STATION
Engineering Office:
1671 Worcester Road
Framingham, Massachusetts 01701
(617) - 872 - 8100

February 12, 1982

SBN-212
T.F. B 7.1.7

United States Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. Frank Miraglia, Chief
Licensing Branch #3
Division of Licensing

References:
(a) Construction Permits CPPR-135 and CPPR-136, Docket Nos. 50-443 and 50-444
(b) USNRC Letter, dated September 30, 1981, "Acceptance Review for Operating Licenses for Seabrook Station, Units 1 and 2," D. G. Eisenhut to W. C. Tallman
(c) PSNH Letter, dated November 27, 1981, "Response to Acceptance Review Requests for Additional Information," J. DeVincentis to D. G. Eisenhut

Subject: Implementation of TMI Action Plan Requirements of NUREG-0737

Dear Sir:

In Reference (b), it was stated that, "...the Seabrook FSAR addresses the requirements contained in NUREG-0737." In addition, RAI 100.2 requested that PSNH, "...identify the FSAR section where details of each applicable TMI Action Item (NUREG-0737) can be found."

Reference (c) indicated that, "Amendment 44 will include a new FSAR Section 1.9 which will provide a statement of our compliance to each applicable item of NUREG-0737," and, "...will also provide a reference to additional locations in the FSAR (if any) where the item is addressed in greater detail."

Based on conversations with Mr. Louis Wheeler (Project Manager), it was mutually agreed that FSAR Section 1.9 would function as a bare reference section only. This letter serves to provide the initial discussion of our compliance with each applicable item of NUREG-0737 (attached). Amendment 45 will include the bare reference section (1.9) and incorporate additional information into the FSAR where appropriate.

Very truly yours,

John DeVincentis

J. DeVincentis
Project Manager

Attachment

these vents form a part of the reactor coolant pressure boundary, the design of the vents shall conform to the requirements of Appendix A to 10 CFR, Part 50, "General Design Criteria." The vent system shall be designed with sufficient redundancy that assures a low probability of inadvertent or irreversible actuation.

Each licensee shall provide the following information concerning the design and operation of the high point vent system.

(1) Submit a description of the design, location size, and power supply for the vent system along with the results of analyses for loss-of-coolant accidents initiated by a break in the vent pipe. The results of the analyses should demonstrate compliance with the acceptance criteria of 10 CFR 50.46.

(2) Submit procedures and supporting analysis for operator use of the vents that also include the information available to the operator for initiating or terminating vent usage.

Response

(1) Refer to FSAR Section 5.2.6, Reactor Coolant Vent System.

(2) Procedures for the use of the Reactor Coolant Vent System will be developed three months prior to fuel load.

Task II.B.2 Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems Which May Be Used in Post-Accident Operations (NUREG-0737)

Position

With the assumption of a post-accident release of the radioactivity equivalent to that described in Regulatory Guides 1.3 and 1.4 (i.e., the equivalent of 50% of the core radio-iodine, 100% of the core noble gas inventory, and 1% of the core solids are contained in the primary coolant), each licensee shall perform a radiation and shielding design review of the spaces around systems that may, as a result of an accident, contain highly radioactive materials. The design review should identify the location of vital areas and equipment, such as the control room, radwaste control stations, emergency power supplies, motor control centers, and instrument areas in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by the radiation fields during post-accident operations of these systems.

Each licensee shall provide for adequate access to vital areas and protection of safety equipment by design changes, increased permanent or temporary shielding, or post-accident procedural controls. The design review shall

determine which types of corrective actions are needed for vital areas throughout the facility.

Response

A design review of plant shielding and qualification of equipment is in progress. The impact of the above releases of radioisotopes is being assessed.

Time-integrated radiation doses from contained post-accident sources have been established for all areas outside the containment containing safety-related equipment. The effect of a 50% cesium release is under investigation.

Maximum dose rates have been calculated for most areas outside containment including all vital areas requiring occupancy in the critical period immediately following an accident. The acceptance criterion for the dose received in locations requiring continuous occupancy is 15 mrem/hr. averaged over the first 30 days. For locations requiring infrequent access the maximum acceptable dose will be 5 rem per task.

The task of establishing post accident radiation levels in accordance with NUREG-0737 will be completed by May 1, 1982.

Task II.B.3 Post-Accident Sampling Capability (NUREG-0737)

Position

A design and operational review of the reactor coolant and containment atmosphere sampling line systems shall be performed to determine the capability of personnel to promptly obtain (less than 1 hour) a sample under accident conditions without incurring a radiation exposure to any individual in excess of 3 and 18-3/4 rem to the whole body or extremities, respectively. Accident conditions should assume a Regulatory Guide 1.3 or 1.4 release of fission products. If the review indicates that personnel could not promptly and safety obtain the samples, additional design features or shielding should be provided, to meet the criteria.

A design and operational review of the radiological spectrum analysis facilities shall be performed, to determine the capability to promptly quantify (in less than 2 hours) certain radio-nuclides that are indicators of the degree of core damage. Such radio-nuclides are noble gases (which indicate cladding failure), iodines and cesiums (which indicate high fuel temperatures), and non-volatile isotopes (which indicate fuel melting). The initial reactor coolant spectrum should correspond to a Regulatory Guide 1.3 or 1.4 release. The review should also consider the effects of direct radiation from piping and components in the auxiliary building and possible contamination and direct radiation from airborne effluents. If the review indicates that the analyses required cannot be performed in a prompt manner with existing equipment,

then design modifications or equipment procurement shall be undertaken to meet the criteria.

In addition to the radiological analyses, certain chemical analyses are necessary for monitoring reactor conditions. Procedures shall be provided to perform boron and chloride chemical analyses, assuming a highly radioactive initial sample (Regulatory Guide 1.3 or 1.4 source term). Both analyses shall be capable of being completed promptly (i.e., the boron sample analysis within an hour and the chloride sample analysis within a shift).

Response

The shielding and operation of the reactor coolant and containment atmosphere sampling systems has been designed to provide the capability of personnel to promptly obtain (less than 1 hour) a sample under accident conditions without incurring a radiation exposure in excess of the limits delineated for this requirement. A post-accident sampling panel has been designed to NUREG-0737. However, additional requirements presented in Regulatory Guide 1.97 are presently being reviewed. Resolution of these additional requirements will be completed by July 1, 1982.

Procedures to obtain post-accident samples and the radiological and chemical analyses will be developed three months prior to fuel load.

Task II.B.4 Training for Mitigating Core Damage (NUREG-0737)

Position

Licensees are required to develop a training program to teach the use of installed equipment and systems to control or mitigate accidents in which the core is severly damaged. They must then implement the training program.

Response

A training program to teach the use of equipment and systems to mitigate accidents involving core damage will be developed prior to fuel load and be completed prior to full-power operations. Operating personnel from the Station Manager through the operations chain to the licensed operators will receive training equivalent to that identified in Enclosure 3 to H. R. Denton's March 28, 1980 letter. Portions of the training will also be administered to supervisors and technicians in the Instrumentation and Control, Health Physics, and Chemistry departments commensurate with their responsibilities.

the isolation signal is reset and manual action is taken to reopen the valve".

(5) Phase A containment isolation ("T" signal) isolates all non-essential process lines on receipt of a safety injection signal. This isolation signal is assumed to be generated when the containment pressure reaches a maximum of 7.4 psig, which includes a drift variation from the nominal value of 5.0 psig. The low set point value, 2.6 psig, which accounts for drift below nominal, is the minimum compatible with normal operating conditions, i.e., 0.5 psig normal to 1.5 psig maximum. See Section 6.2.1.

(6) The containment isolation purge supply air valves, COP-V1 and COP-V2, as well as the containment isolation purge exhaust air valves, COP-V3 and COP-V4, are ANSI Safety Class 2, Seismic Category I valves. They are redundant valves in series and are provided with ANSI Safety Class 2, seismic Category I, penetration piping between them. The valves are required to be shut immediately following a containment ventilation isolation or containment high radiation signal. Since the valves may be open during normal plant operation, start-up, and hot standby, these valves will be periodically tested to insure valve and valve actuator performance. The applicable General Design Criteria, valve position, closure time, etc., are given in Table 6.2-83. A full description of containment isolation valves is given in Section 6.2.4.

(7) Per Table 6.2-83, the containment isolation purge supply air valves, COP-V1 and COP-V2, as well as the containment isolation purge exhaust air valves, COP-V3 and COP-V4, close on a high radiation signal as well as on a containment ventilation isolation signal (CVIS).

Task II.F.1 Additional Accident-Monitoring Instrumentation

Task II.F.1, Attachment 1 Noble Gas Effluent Monitor (NUREG-0737)

Position

Noble gas effluent monitors shall be installed with an extended range designed to function during accident conditions as well as during normal operating conditions. Multiple monitors are considered necessary to cover the range of interest.

(1) Noble gas effluent monitors with an upper range capability of 10^5 Ci/cc (Xe-133) are considered to be practical and should be installed in all operating plants.

(2) Noble gas effluent monitoring shall be provided for the total range of concentration extending from normal condition (as low as reasonably achievable - ALARA) concentrations to a maximum of 10^5 Ci/cc (X-133). Multiple monitors are considered to be necessary to cover the range of interest. The range capacity of individual monitors should overlap by a factor of ten.

Response

Refer to FSAR Section 12.3.4.2.b.2.(e) and FSAR Table 12.3-14.

Task II.F.1, Attachment 2 Sampling & Analysis of Plant Effluents (NUREG-0737)

Position

Because iodine gaseous effluent monitors for the accident condition are not considered to be practical at this time, capability for effluent monitoring of radio-iodines for the accident condition shall be provided with sampling conducted by adsorption on charcoal or other media, followed by on-site laboratory analysis.

Response

Seabrook Station will have equipment to collect and analyze representative samples of radioactive iodines and particulates in station gaseous effluents during and following an accident.

The NRC will be kept appraised of our progress in selecting equipment.

Task II.F.1, Attachment 3 Containment High-Range Radiation Monitor (NUREG-0737)

Position

In containment radiation-level monitors with a maximum range of 10^8 rad/hr shall be installed. A minimum of two (2) such monitors that are physically separated shall be provided. Monitors shall be developed and qualified to function in an accident environment.

Response

Refer to FSAR Table 7.5-1, item 16. (Range: 10^0 to 10^7 R/hr, gamma only)

(1) Immediate leak reduction -

(a) Implement all practical leak reduction measures for all systems that could carry radioactive fluid outside of containment.

(b) Measure actual leakage rates with system in operation and report them to the NRC.

(2) Continuing Leak Reduction - Establish and implement a program of preventive maintenance to reduce leakage to as-low-as-practical levels. This program shall include periodic integrated leak tests at intervals not to exceed each refueling cycle.

Response

(1) (a) A leak reduction program, identifying all systems that could carry radioactive fluid outside of containment, will be prepared four months prior to fuel load. During pre-operational and Hot Functional testing, these systems will be visually inspected and all practical leak reduction measures will be implemented.

(b) Actual leakage rates with the systems in operation will be provided as a part of the initial Startup Test Report.

(2) An ongoing leak reduction program, including preventative maintenance to reduce leakage to as-low-as-practical levels, and periodic integrated leak tests, at intervals not to exceed each refueling, shall be prepared four months prior to fuel load.

Task III.D.3.3 Improved Inplant Iodine Instrumentation Under Accident Conditions (NUREG-0737)

Position

Each licensee shall provide equipment and associated training and procedures for accurately determining the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident.

Response

Seabrook Station will provide equipment and associated procedures and training for accurately determining the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident. The NRC will be kept appraised of our progress in selecting equipment.

Question 471.12
FSAR Section (12.5)

Provide additional information on how your exposure tracking and exposure reduction program includes the elements of Regulatory Guide 1.70, Section 12.1.3 and 12.5.3, and Regulatory Guide 8.8, Section C.3.9(8)(j), C.3.8(2), and C.3.c(2)(5), including rem-tracking, self-reading pocket dosimeter use, post task, actual exposure evaluation, and how these results are used to make changes in future work. Verify that annual exposure reviews are performed by plant management and that these are used to identify groups with the highest exposure in order to assure that doses are ALARA.

Response to 471.12

The exposure reduction program is essentially the application of ALARA philosophies delineated in FSAR Section 12.1. Individual dose-tracking is accomplished on a daily basis through the use of self-reading pocket dosimeters (SRPD's). Specific guidance is provided in health physics procedures. Dose tracking with respect to specific jobs, work groups and components is possible through correlation of data obtained through the RWP system and exposure records. Dose tracking in these areas is applied as time, personnel and radiological conditions warrant or health physics supervision deems necessary.

The general operational concept of ALARA evaluation, analysis and review is presented in FSAR Section 12.1.3.1. Specific ALARA review levels and guidelines are contained in health physics department procedures. Elements of Regulatory Guide 8.8, Revision 3 are exercised through the use of preemptive analysis and planning, ongoing observations and audits and historical analysis and review.

Question 471.13
FSAR Section (12.3.4)

Your listing of area radiation monitors in Section 12.3.4 shows only one monitor with a range of 10^4 R/hr or above. Your listing of portable radiation survey instruments in Table 12.5.3 shows no instruments with a range of 10^4 R/hr. Regulatory Guide 1.97 (Rev. 2) specifies that portable survey meters and the area radiation monitors in areas requiring access after an accident should have a range up to 10^4 R/hr. You should provide a commitment in your FSAR to have such portable instruments and should specify locations of area radiation monitors in areas requiring access after an accident.

Response to 471.13

Area monitors with a range up to 10^4 R/hr will be installed as follows:

Location	No. of Detectors Per Unit
Lower Level PAB	2
PAB Entrance	2
Fuel Storage Bldg.	1
RHR Pump Vaults	2

In addition, the station health physics staff is currently evaluating vendors who offer portable survey meters with ranges of at least 10^4 R/hr. These instruments shall be purchased. Further information will be available on site for the NRC review after the purchase is made.

FSAR Table 12.5-1 will be revised in a future FSAR amendment as per the attached copy to list the high range portable instruments.

TABLE 12.5-1

PORTABLE HEALTH PHYSICS INSTRUMENTATION

Type of Instrument	Quantity Minimum	Sensitivity	Range	Calibration Method	Calibration Frequency
Ion Chamber (Low Range)	16	Beta, Gamma	0 to 1R/hr.	Source	Quarterly
Ion Chamber (Mid Range)	8	Gamma	0 to 1,000R/hr.	Source	Quarterly
Ion Chamber (High Range)	4	Gamma	Up to 10,000 R/hr.	As recommended by Manufacturer	
Geiger Mueller Detector	6	Beta, Gamma	0 to 50,000 CPM	Source and Pulse Generator	Quarterly
	10	Beta, Gamma	0 to 200mR/hr.	Source	
Alpha Scintillation or Proportional Detector	4	Alpha	0 to 500,000 CPM	Source	Quarterly
Teledetector	4	Gamma	0 to 1,000R/hr.	Source	Quarterly
Neutron Dose Rate Detector	3	Neutron	0.001 to 10R/hr.	Source and Pulse Generator	Quarterly
Air Sampler (Low Volume)	10	Particulate and Iodine	--	Flow Rate	Semi-Annual
Air Sampler (High Volume)	6	Particulate and Iodine	--	Flow Rate	Semi-Annual
Air Sampler (Personnel)	10	Particulate	--	Flow Rate	Semi-Annual
Portable Air Sampler (C.A.M.) (Part of RMS)	4	Particulate and Iodine	--	Flow Rate	Semi-Annual

Attachment to Question 471.13

Question 471.14

FSAR Section (13.1.1.1)

As specified in Regulatory Guide 1.70, Section 13.1.1.3 and NUREG-0737, you should provide an outline of the qualifications of the individual designated as your Radiation Protection Manager (RPM), and the individual designated to act as replacement RPM. It is our position that the RPM should meet or exceed the qualifications specified in Regulatory Guide 1.8 for Radiation Protection Manager. Identify and provide an outline of the qualifications of the individual who will act as RPM.

Response to 471.14

The Seabrook Station "Health Physics Department Supervisor" is the individual designated to be the "Radiation Protection Manager" as that position is described in ANS 3.1-1978 and Regulatory Guide 1.8, Revision 1-R.

His duties and responsibilities are outlined in FSAR Section 13.1.2.2.c.2. His qualifications, as stated in FSAR Section 13.1.3.1 and as shown by his resume in FSAR Chapter 13, Appendix 13D, equal or exceed those required in the above noted references.

In the case of an extended absence of the Health Physics Department Supervisor (RPM) the individual designated to act as the replacement Health Physics Department Supervisor (RPM) will meet the minimum qualification requirements for this individual as discussed in the response to Question 471.8.

Question 471.15

FSAR Section (13.1)

Figure 13.1-3 shows that only 12 health physics technicians are planned, for both units. Actual experience at 2-unit operating stations has indicated that 12 technicians would not be sufficient to accommodate the duties associated with operation of the station during normal operation, anticipated operational occurrences and outages. You should evaluate the manpower requirements for HP technicians and revise the proposed number of technicians accordingly.

Response to 471.15

The twelve health physics technicians shown in Figure 13.1-3 is the PSNH commitment for the minimum number of regular technicians to be on staff to meet the site manning requirement of section 6.2.2.c of the Technical Specifications.

This minimum basic staff will be augmented as necessary to meet the requirements of normal station operations, anticipated operational events, and outage workloads. The additional personnel will be Health Physics Technicians on staff, other specially trained personnel on the Station Staff and, when required, could include appropriately qualified contractor personnel.