The Light company Houston Lighting & Power P.O. Box 1700 Houston, Texas 77001 (713) 228-9211

October 12, 1985
ST-HL-AE-1409
File No.: G9.17

Mr. George W. Knighton, Chief
Licensing Branch No. 3
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, DC 20555

South Texas Project
Units 1 and 2
Docket Nos. STN 50-498, STN 50-499
Responses to DSER/FSAR Items
For Chapters 7 and 12 on ALARA and TMI Item II.F.1

Dear Mr. Knighton:

The attachments enclosed provide STP's response to Draft Safety Evaluation Report (DSER) or Final Safety Analysis Report (FSAR) items.

The item numbers listed below correspond to those assigned on STP's internal list of items for completion which includes open and confirmatory DSER items, STP FSAR open items and open NRC questions. This list was given to your Mr. N. Prasad Kadambi on October 8, 1985 by our Mr. M. E. Powell.

The attachments include mark-ups of FSAR pages which will be incorporated in a future FSAR amendment unless otherwise noted below.

The items which are attached to this letter are:

Attachment	Item No.*	Subject
1	N/A	Answer to question from reviewer about the in-Containment fuel racks
2	D 12.1-2, D 12.3-2	ALARA - lighting fixtures; Height that walls are painted
3	D 12.4-1, F 12.3-1	Exposure Estimates
4	D 12.3-1	Location for temporary shielding
5	F 12.3-21	Building ventilation
6	F 7.0-34, 35 & 37	TMI Item II.F.1 (delete reference to "Later")

* Legend
D - DSER Open Item
F - FSAR Open Item
C - DSER Confirmatory Item
Q - FSAR Question Response Item

B001 1/1

L1/DSER/b

8510180122 851012
PDR ADOCK 05000498
E PDR

Houston Lighting & Power Company

If you should have any questions concerning this matter, please contact Mr. Powell at (713) 993-1328.

Very truly yours,

M. R. Wisenburg
Manager, Nuclear Licensing

MEP/bl

Attachments: See above

L1/DSER/b

cc:

Hugh L. Thompson, Jr., Director
Division of Licensing
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Robert D. Martin
Regional Administrator, Region IV
Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 1000
Arlington, TX 76011

N. Prasad Kadambi, Project Manager
U.S. Nuclear Regulatory Commission
7920 Norfolk Avenue
Bethesda, MD 20814

Claude E. Johnson
Senior Resident Inspector/STP
c/o U.S. Nuclear Regulatory
Commission
P.O. Box 910
Bay City, TX 77414

M.D. Schwarz, Jr., Esquire
Baker & Botts
One Shell Plaza
Houston, TX 77002

J.R. Newman, Esquire
Newman & Holtzinger, P.C.
1615 L Street, N.W.
Washington, DC 20036

Director, Office of Inspection
and Enforcement
U.S. Nuclear Regulatory Commission
Washington, DC 20555

E.R. Brooks/R.L. Range
Central Power & Light Company
P.O. Box 2121
Corpus Christi, TX 78403

H.L. Peterson/G. Pokorny
City of Austin
P.O. Box 1088
Austin, TX 78767

J.B. Poston/A. vonRosenberg
City Public Service Board
P.O. Box 1771
San Antonio, TX 78296

Brian E. Berwick, Esquire
Assistant Attorney General for
the State of Texas
P.O. Box 12548, Capitol Station
Austin, TX 78711

Lanny A. Sinkin
3022 Porter Street, N.W. #304
Washington, DC 20008

Oreste R. Pirfo, Esquire
Hearing Attorney
Office of the Executive Legal Director
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Charles Bechhoefer, Esquire
Chairman, Atomic Safety &
Licensing Board
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dr. James C. Lamb, III
313 Woodhaven Road
Chapel Hill, NC 27514

Judge Frederick J. Shon
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Mr. Ray Goldstein, Esquire
1001 Vaughn Building
807 Brazos
Austin, TX 78701

Citizens for Equitable Utilities, Inc.
c/o Ms. Peggy Buchorn
Route 1, Box 1684
Brazoria, TX 77422

Docketing & Service Section
Office of the Secretary
U.S. Nuclear Regulatory Commission
Washington, DC 20555
(3 Copies)

Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
1717 H Street
Washington, DC 20555

Revised 9/25/85

Attachment 1

9.1 FUEL STORAGE AND HANDLING

Facilities for the receipt and storage of new fuel and the storage and transfer of spent fuel are housed in the Fuel-Handling Building (FHB). A separate and independent FHB is provided for each unit of the STP. Each FHB is designed as a controlled-leakage seismic Category I structure. The design of the FHB Heating, Ventilating and Air Conditioning System is discussed in Section 9.4.2. The structural design considerations are described in Section 3.8.4.

9.1.1 New Fuel Storage

9.1.1.1 Design Bases. The new fuel storage pit is a reinforced concrete pit and an integral part of each seismic Category I FHB. This pit provides temporary dry storage for approximately one-third core (66 fuel assemblies) of new fuel. The fuel is stored in racks (Figure 9.1.1-1) composed of individual vertical cells fastened together to form three 2 x 11 modules which may be bolted to anchors in the floor and walls of the new fuel storage pit. The new fuel racks are classified as seismic Category I components, as defined by Regulatory Guide (RG) 1.29, and American Nuclear Society (ANS) Safety Class (SC) 3 (see Section 3.2).

The new fuel racks are designed with a center-to-center spacing of 21 in. This spacing provides a minimum of 12 in. between adjacent fuel assemblies. This separation is sufficient to maintain a subcritical array assuming optimum moderation. Space between storage positions is blocked to prevent insertion of fuel. All rack surfaces that come into contact with the fuel assemblies are made of annealed austenitic stainless steel, and the support structure is painted carbon steel.

The racks are designed to withstand normal operating loads, as well as to remain functional with the occurrence of a Safe Shutdown Earthquake (SSE). The new fuel racks are designed to withstand a maximum uplift force of 5,000 pounds and to meet the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III, Appendix XVII.

The new fuel storage pit access hatch is a three-section cover. This cover will minimize the introduction of dust and debris into the pit. The cover is designed to withstand the impact force of a new fuel assembly dropped from the maximum elevation allowed by the 2-ton hoist of the FHB overhead crane.

9.1.1.2 Facilities Description. The FHB abuts the south side of the ~~Reactor Containment Building~~ (RCB)s and is adjacent to the west side of the Mechanical-Electrical Auxiliaries Building (MEAB) of each unit. The locations of the two FHBs are shown in the station plot plan on Figure 1.2-3. For general arrangement of the new fuel storage facilities, refer to Figures 1.2-32 through 1.2-40.

New fuel assemblies are received in the receiving area of each FHB and temporarily stored in the shipping containers in the new fuel handling area. In the new fuel handling area, each new fuel assembly is removed from its shipping container and inspected visually to confirm the assembly has not been damaged during shipment. The new fuel assemblies are transported from the inspection area to the new fuel storage pit or to the new fuel elevator by the

In addition space is provided for the storage of fuel during refueling inside the ~~RS~~ Reactor Containment Building (RCB). See Section 9.1.2.1 for a description of the racks.

floor of the spent fuel ~~pit~~ pool. The spent fuel racks are classified as seismic Category I, as defined by RG 1.29, and as ANS SC 3.

Another 400 storage spaces are provided in four 7x7, four 6x7, and one 6x6 modules. These modules have 14-inch center-to-center spacings (Figure 9.1.2-1b) and are "free standing" in that they rest on vertical shear pins attached to adapter plates that are bolted to the spent fuel pit floor. 6
Spaces between storage cells are blocked to prevent improper insertion of fuel. Both the 16-in. and 14-in. spacings provide sufficient separation between fuel assemblies to maintain a subcritical array. All rack surfaces that come into contact with fuel assemblies are made of annealed austenitic stainless steel. These materials are resistant to corrosion during normal and emergency water quality conditions. 23

~~In the RCB space is~~ provided for 64 ~~spent~~ fuel assemblies in four 4x4 modules having 16-inch center-to-center spacing (Figure 9.1.2-1(a)).

The racks are designed to withstand normal operating loads as well as to remain functional with the occurrence of an SSE. The racks are designed with adequate energy absorption capabilities to withstand the impact of a dropped spent fuel assembly from the maximum lift height of the spent fuel pit bridge hoist. The racks are designed to withstand a maximum uplift force equal to the uplift force of the bridge hoist. The racks also meet the requirements of ASME Code, Section III, Appendix XVII.

Shielding for the spent fuel pool is adequate to protect plant personnel from exposure to radiation in excess of published guideline values as stated in Section 12.1. A depth of at least 10 ft of water over the top of the spent fuel assemblies will limit direct radiation to 2.5 mR/hr (surface dose rate).

The FHB Ventilation Exhaust System is designed to limit the offsite dose in the event of a significant release of radioactivity from the fuel, as discussed in Sections 12.3.3, 15.7.4 and 9.4.2.

The FHB is designed to prevent missiles from contacting the fuel. A more detailed discussion on missile protection is given in Section 3.5.

9.1.2.2 <u>Facilities Description</u>. The FHB abuts the south side of the RCB and is adjacent to the west side of the MEAB of each unit. The locations of the two FHBs are shown in the station plot plan on Figure 1.2-3. For general arrangement of the spent fuel storage facilities, refer to Figures 1.2-32 through 1.2-40.

In addition, space is provided for storage of fuel during refueling inside the Reactor Containment Building ...

The spent fuel storage facilities are designed for the underwater storage of spent fuel assemblies and control rods after their removal from the reactor
vessel. The spent fuel is transferred to the FHB and handled and stored in 6
the spent fuel pool underwater. The fuel is stored to permit some decay, then 7
transferred offsite. For a detailed discussion of spent fuel handling, see 6
Section 9.1.4.

The spent fuel pool is located in the northwest quadrant of each FHB. The floor of the pool is at El. 21 ft 11 in., with normal water level at El. 66 ft
6 in. The top of a fuel assembly in a storage rack is El. 39 ft. The storage 6
of 724 spent fuel assemblies is accommodated in the pool. The fuel assemblies 7
are loaded into the spent fuel racks through the top and are stored vertically. 6

ATTACHMENT 1
ST-HL-AE-1409
PAGE 4 OF 4

neutron flux (low setting) reactor trip. Protection against uncontrolled boron dilution is described in Section 15.4.6.

The following significant points are assured by the refueling procedure:

1. The refueling water and the reactor coolant contain approximately 2,500 ppm boron. This concentration is sufficient to keep the core approximately 5 percent $\Delta k/k$ subcritical during the refueling operations with all control rods removed and the core refueled to provide sufficient excess reactivity for operation to the next refueling outage. 40 40

2. The water level in the refueling cavity is high enough to keep the radiation levels within acceptable limits when the fuel assemblies are being removed from the core.

The refueling operation is divided into four major phases: (1) preparation, (2) reactor disassembly, (3) fuel handling, and (4) reactor assembly. A general description of a typical refueling operation through the four phases is given below:

This description applies to rapid (unrodded) refueling which will normally be used during a refueling shutdown to maximize plant availability. The description also points out the different steps that would be included in a non-rapid (rodded) refueling operation, which is typically used for extended shutdowns involving non-routine maintenance. 40

9.1.4.2.2.1 Phase I - Preparation - The reactor is shut down (rods in) and then simultaneously borated and cooled down to refueling shutdown condition. Following initiation of normal purge and a radiation survey of the Containment Building refueling operations may proceed. The fuel transfer equipment and refueling machines are checked for proper operation. Each new fuel assembly is brought from dry storage in the FHB as described in Section 9.1.4.2.1. After transfer through the fuel transfer tube, the FTS fuel assembly container is pivoted to the vertical position by the in-containment upender. The refueling machine transfers the new fuel into the in-Containment storage racks (shown in Figure 9.1.2.1.a). Refer to Figures 1.2-14 through 1.2-18 for general arrangement of the in-containment fuel storage area. 40

When the Reactor Coolant System (RCS) has been cooled to 150°F, RCS draining is started. The RCC assemblies (control rods) are withdrawn to their full-out position, and each control rod's holdout device is activated to ensure that the rod is held in its withdrawn position inside its upper internals guide tube and reactor head pressure housing. As soon as RCS draining has lowered reactor coolant level to the reactor vessel nozzle centerline, degassing operations are performed. 40

For non-rapid refueling, the preparation for refueling is similar except that the control rods are not withdrawn from the core and the RCS temperature is reduced to 140°F before draining.

9.1.4.2.2.2 Phase II - Reactor Disassembly - The seismic tie rods attached to the missile shield are disconnected and stored. The insulation is removed from the vessel head flange area, and the Roto-Lok studs are detensioned and removed from the vessel flange. A stud hole plug is installed in each hole after the stud is removed to prevent entry of water. In addition all flux mapping detectors and thimbles are retracted through the bottom of 40

Attachment 2

ATTACHMENT 2
ST-HL-AE-1409
PAGE 2 OF 3

to hold the sections in place. This arrangement provides for rapid removal of the insulation for inservice inspection and a reduction in radiation exposure. 4 Q331.9

37

10. Transfer of expended liquid filter elements from the filter area to the Solid Waste Processing System, (SWPS) as described in Section 11.4, results in a reduction of radiation exposure during handling of these contaminated filters. 37

11. Permanent shielding is provided, so that workers may stay behind walls or in areas of lower radiation level when not actively involved in work in radiation areas. For some jobs, temporary shielding, such as lead blankets draped over a pipe on either side of a valve or concrete blocks stacked around a piece of equipment, is used. Temporary shielding will be used if the total exposure, which includes the exposure received during installation and removal of shielding, is substantially reduced. 37

Insert "A"

12.1.3.2 <u>ALARA Considerations for Steam-Generator Repairs</u>. Applicable techniques in Section 12.5.3.2 are normally used when inspecting and plugging SG tubes. SG repairs will be made using state of the art methods, thus reducing time spent by personnel inside the channel heads. 37

12.1.3.3 <u>Specific ALARA Considerations for Reactor Head Removal and Installation</u>: Applicable techniques in Section 12.5.3.2 normally are used when removing and installing the reactor head and performing the other associated activities necessary for refueling. The rapid refueling system significantly reduces personnel exposure. This system allows the rapid disengagement of the reactor studs, which in the past has caused considerable exposure to personnel, to be done on a remote basis. Also, with this system it is not necessary to disconnect the electrical cables to the reactor head, which saves time and exposure. 4

A permanent welded seal ring, rather than a bolted one, is used to seal the cavity prior to flooding, thus reducing both direct exposure and airborne activity during installation. Because of the above considerations, time spent in the refueling cavity has become minimal. When jobs of long duration are required, however, communications are provided up to the refueling floor, thus reducing lost time and exposure to personnel. When jobs of long duration are required in areas with very high radiation levels, temporary shielding is provided for personnel protection. 37

12.1.3.4 <u>Specific ALARA Considerations for Inservice Inspections (ISIs)</u>. Applicable techniques in Section 12.5.3.2 are normally used when performing ISIs. Where feasible, remote testing devices are used. An example of this is the remote eddy current probe positioner and probe indexer used to examine the SG tubes. Another remotely operated device is used to perform the ISI of the reactor vessel and nozzle welds. 37

12.1.3.5 <u>Specific ALARA Considerations for Other Exposure-Related Jobs.</u> Other operations such as refueling, radwaste handling, spent fuel handling, loading and shipping, routine maintenance, sampling, and calibration are discussed in Section 12.5.3. Various combinations of the techniques in Section 12.5.3.2, as applicable to the particular job, are used on these jobs. One of the techniques expected to reduce exposures on some high exposure jobs is the 4 37

Insert "A"

12. To facilitate decontamination, the walls of rooms which have the potential to become ~~radioactive~~ contaminated with radioactivity ~~contamination~~ are painted. This decontaminable surface is applied to the walls to a minimum of [redacted] 40 inches above the floor surface.

13. Lighting fixtures used in high radiation areas are provided with extended life incandescent lamps to reduce maintenance time required for relamping. Where possible each room or area has at least two fixtures with switches installed locally in areas to minimize exposure.

Attachment 3

ATTACHMENT 3
ST-HL-AE-1409
PAGE 2 OF 6

2. Exposure times are based on estimates of the average times required to perform the designated tasks. 39

3. Radiation protection personnel exposures are based on the ~~1976 occupational exposure reports (Ref. 12.4-1) submitted to the NRC by PWR power plants. Health physics personnel average doses varied from 2.35 rem/yr to 3.86 rem/yr. The mean dose for all plants surveyed was 2.66 rem/yr. It is assumed that the dose received by STP radiation protection personnel during 2,000 hours of plant patrol is 2.50 rem.~~ 39 12

Insert "B"

4. It is assumed that techniques for inservice inspections described in WCAP-8872, "Design, Inspection, Operation and Maintenance Aspects of the W NSSS to Maintain Occupational Exposures as Low as Reasonably Achievable" (Ref. 12.4-2), are used. Steam generator inspection is assumed to be made using the Westinghouse remote positioner fixture. 39

5. Airborne dose estimates are reported for thyroid and total-body exposures. These are based on the plant airborne activity concentrations listed in Table 12.2.2.-2. 39

6. Maintenance operations are assumed to be divided between mechanical, I&C, and electrical personnel as follows: mechanical - 40 percent, I&C - 40 percent, electrical - 20 percent. Components are isolated by an Auxiliary Operator, and it is assumed that his assistance is required for 1.0 man-hour per task.

7. All personnel located in unrestricted (Zone 1) areas are assumed to be subject to approximately the same average dose rate of 0.04 mrem/hr. 39

8. The following data are used for determining plant personnel assignments. The operating shift crew of STP Unit 1 and Unit 2 is described in Section 13.1.2.3. 39

- Plant Manager (1) 39

  The Plant Manager is normally at work 8 hours per day, 5 days per week, 50 weeks per year. He spends 95 percent of his time in the plant office and control room area and 5 percent of his time in the plant area.

- Plant Superintendent (1) 39

  The Plant Superintendent is normally at work 8 hours per day, 5 days per week, 50 weeks per year. It is assumed that he spends 95 percent of his time in the plant office and control room area and 5 percent of his time in the plant area.

- Reactor Operations Superintendent (1) 39

  The Reactor Operations Superintendent is normally at work 8 hours per day, 5 days per week, 50 weeks per year. It is assumed that he spends approximately 75 percent of his time in the plant office and control room area and 25 percent of his time in the plant.

INSERT "B"

and Table 12.4-4.

... information provided below in item 8. The resulting doses can be found in Table 12.4-4.

ATTACHMENT 3
ST-HL-AE-1409
PAGE 4 OF 6

REFERENCES

Section 12.4:

12.4-1 1976 Occupational Exposure Reports, supplied by Barbara Brooks/NRC/OMIPC. Deleted

12.4-2 "Design, Inspection, Operation and Maintenance Aspects of the W NSSS To Maintain Occupational Exposures As Low As Reasonably Achievable," WCAP-8872.

49

ATTACHMENT 3
ST-HL-AE-1409
PAGE 5 OF 6

12.3.4.2 Airborne Radioactivity Monitoring. The monitors provided in 44
the RMS for measuring plant airborne radioactivity are identical in design and
installation to the process and effluent airborne monitors described in
Section 11.5.2.3. Each is an off-line type monitor which pulls an air sample 44
from the HVAC System duckwork. Alarm, display, and recording capabilities, as
described in Section 11.5, are provided for the airborne radioactivity 44
monitors.

The airborne radioactivity monitors provide a continuous surveillance of the 44
airborne radioactivity level within selected plant areas to warn if airborne
activity exceeds 10 times the MPC-hr (maximum permissible concentration-hour),
as set forth in 10CFR20, Appendix B, Table I, for iodine and particulate
radiation. Selection of areas to be monitored is based on the potential
airborne radioactivity sources in the areas and the frequency with which the 44
area is occupied.

Three mobile airborne radioactivity monitors are provided. The mobile 44
monitors each have particulate, iodine, and gaseous detector channels, as
described in Section 11.5.2.3.1, and can be used for measuring the airborne 44
radioactivity level in any specific area. Local alarm, display, and recording
capabilities of a monitor, as described in Section 11.5, are provided. 44

The range of the airborne radioactivity monitors was chosen to achieve the 44
low-level radiation detection required for monitoring MPC concentrations.
Design parameters for the airborne radiation monitors are shown in Table 44
12.3.4-2.

The calibration and testing of the fixed and mobile airborne monitors are
identical to those of the process and effluent monitors, as described in
Section 11.5.2.1.5. 44

12.3.4.3 In-Plant Sampling.

1. Criteria

The criteria for the selection of air sampling and monitoring instrumentation used to determine the concentrations of airborne radioactivity in plant areas and effluents are given in Section 12.5.2.2(5).

2. Methods

Airborne radioactivity concentrations in areas accessible to personnel are
determined by use of both fixed and mobile airborne radiation monitors and 39
high-volume "grab"-type air samplers.

The fixed monitors sample the exhaust air in the HVAC ducts from selected cubicles. These monitors are described in Section 12.3.4.2 and the locations are shown on Figures 9.4.3-2&4.

The mobile monitors are used to collect representative samples of airborne 39
radioactive concentrations. In addition, the mobile monitors are used to
determine the exact source of airborne radioactivity when the fixed monitors
have reached an alarm condition, as follows.

5. Detectors are located to provide easy access so that minimal maintenance equipment is required and to provide an uncluttered area near the detector to allow for field alignment and calibration.

12.3.4.1.2 <u>Design Criteria</u>: The basic design criteria of the area radiation monitors are identical to those of the RMS, as discussed in Section 11.5, and are supplemented by the following:

1. Each has two local audible and visible alarms which are initiated on a "high" or "alert" alarm at the detection unit or at an area where the local visual alarm can be seen before entry into the detection area. 44

2. A monitor's detector's can be separated from the electronics. 39

3. There are two RCB monitors which have post-accident monitoring capability for a Condition IV accident, as discussed in Chapter 15. These (RCB High Range Monitors) are considered safety-related and meet the requirements of RG 1.97 and NUREG-0737. They are physically located as shown on Figure 12.3.1-6 44

4. The range of the monitors is selected such that the maximum zone dose rate and any anticipated operational occurrences which would increase the dose rate can be detected.

5. A portable area radiation monitor is provided which can be used to monitor the area radiation level in locations not normally monitored or in maintenance areas. 39

6. RG 1.97 radiation monitoring requirements are addressed in Section 7.5 and Appendix 7B. 44

7. A monitor has a minimum accumulative accuracy of ± 25 percent of the actual intensity value. 44

8. The design of the fuel pool racks precludes criticality under all postulated normal and accident conditions. Therefore, criticality monitors, as stated in 10CFR 70.24 and Regulatory Guide 8.12, are not needed.

12.3.4.1.3 <u>Equipment Description</u>: The area monitors are an integral part of the RMS, as discussed in Section 11.5. The area monitors have the identical alarm, display, and recording capabilities, locally and in the main control room, as described for the process and effluent monitors in Section 11.5. 39

Table 12.3.4-1 lists the location, range and alarm setpoints of the area monitors. The monitor locations are also shown on Figures 12.3.1-1 thru 16.

12.3.4.1.4 <u>Monitor, Calibration and Testing</u>: The Area Radiation 39
Monitoring System is calibrated using one or more reference standards certi- 3
fied by NBS or using standards from suppliers participating in measurement 39
assurance activities with NBS. The calibration is verified for at least three 3
points over the entire range of the monitor.

A channel calibration that includes a channel functional test is performed at least once every eighteen months or during the refueling outage if the detector is not readily accessible. In the event that a calibration is questionable, the channel can be isolated and a more thorough calibration performed. 39

Attachment 4

7. Piping containing radioactive material is appropriately routed to minimize exposure to plant personnel. This involves:

   a. Routing of piping containing radioactive material through radioactive pipe chases or behind shield walls (cubicles) suitably designed to ensure that the design dose rates in radiation shielding zones 39
   are not exceeded. These zones are described in Table 12.3.2-1.

   b. Wherever possible, piping which normally contains radioactive material is separated from piping containing nonradioactive material. This ensures that exposures to personnel performing inspection or maintenance on nonradioactive systems are ALARA.

   c. Pipelines are sloped, wherever possible, so that the slope will assist in removing crud deposits from the line prior to maintenance operations. Pipes that handle spent demineralizer resin have a 39
   minimum 5 diameter bend radii.

   d. To prevent crud traps, thermal expansion loops are horizontal or 39
   vertical upwards rather than vertical downwards and eccentric reducers are used on piping lines.

   e. The letdown line is routed within the Reactor Containment Building (RCB) secondary shield and provides sufficient delay time for decay of the reactor coolant N-16 activity so that N-16 is not a factor in shielding outside the RCB.

8. The principal shielding material is concrete with a dry density of 136 lb/ft^3. Where necessary to save space, steel is used to supplement the concrete shielding. 39

9. Procedures for concrete shield wall construction present implementation methods that correspond to the position of RG 1.69, which generally endorses the requirements of ANSI-N101.6-1972. 39

10. Consideration has been given to provision for shielding of major sources 39
of radiation during inservice inspection to allow access and minimize radiation exposure of personnel (Section 6.6).

11. Shielding discontinuities caused by shield plugs, concrete hatch covers, and shield doors to high-radiation areas are provided with offsets to reduce radiation streaming.

12. Control of access through posting of radiation sources and areas and use of physical barriers is applied in accordance with guidelines contained in NUREG-0452; Section 6.12. See Table 12.3.2-1. 39

In addition to the above criteria, to ensure that occupational exposures are ALARA according to RG 8.8 criteria, plant radioactive systems and shielding designs are constantly reviewed, updated, and modified as necessary during 39
plant design and construction. Personnel responsible for the various shielding design phases are the shielding engineers and supervisors.

13. As described in Section 12.1.3.1 temporary shielding may be used in order to reduce the doses during activities in radiation areas. Consideration has been made, as part of the ALARA program, for allowing room where possible for the placement and use of temporary shielding.

Attachment 5

12.3-34

TABLE 12.3.4-2

BUILDING VENTILATION MONITORS

Monitor	Service	Sample Location	Detector Number	Detector Type	Analysis Performed	Range (μCi/cc)	MDC (1) (μCi/cc)	Controlling Isotope	Alert Alarm (μCi/cc)	High Alarm (μCi/cc)
RT-8014	Mechanical Auxiliary Building (MAB) Ventilation	MAB El. 10' See Fig. 9.4.3-2	RE-8014A Particulate (P)	(2)	Gross Beta	(4)	(7)	Cs-137	8.5×10^{-10}	1.7×10^{-9}
			RE-8014B Iodine (I)	(3)	Gross Gamma	(5)	(8)	I-131	7.5×10^{-10}	1.5×10^{-9}
			RE-8014C Nobel Gas (NG)	(2)	Gross Beta	(6)	(9)	Kr-85	~~(later)~~ 4.2×10^{-7}	~~(later)~~ 8.4×10^{-7}
RT-8015	MAB Ventilation	MAB El. 10' See Fig. 9.4.3-2	RE-8015A (P)	(2)	Gross Beta	(4)	(7)	Cs-137	1.8×10^{-10}	3.6×10^{-10}
			RE-8015B (I)	(3)	Gross Gamma	(5)	(8)	I-131	1.6×10^{-10}	3.2×10^{-10}
			RE-8015C (NG)	(2)	Gross Beta	(6)	(9)	Kr-85	~~(later)~~ 4.2×10^{-7}	~~(later)~~ 8.4×10^{-7}
RT-8016	MAB Ventilation	MAB El. 10' See Fig. 9.4.3-2	RE-8016A (P)	(2)	Gross Beta	(4)	(7)	Cs-137	4.1×10^{-10}	8.3×10^{-10}
			RE-8016B (I)	(3)	Gross Gamma	(5)	(8)	I-131	3.7×10^{-10}	7.4×10^{-10}
			RE-8016C (NG)	(2)	Gross Beta	(6)	(9)	Kr-85	~~(later)~~ 4.2×10^{-7}	~~(later)~~ 8.4×10^{-7}
RT-8017	MAB Ventilation	MAB El. 60' See Fig. 9.4.3-6	RE-8017A (P)	(2)	Gross Beta	(4)	(7)	Cs-137	1.5×10^{-10}	2.9×10^{-10}
			RE-8017B (I)	(3)	Gross Gamma	(5)	(8)	I-131	1.3×10^{-10}	2.6×10^{-10}
			RE-8017C (NG)	(2)	Gross Beta	(6)	(9)	Kr-85	~~(later)~~ 4.2×10^{-7}	~~(later)~~ 8.4×10^{-7}
RT-8018	MAB Ventilation	MAB El. 41' See Fig. 9.4.3-6	RE-8018A (P)	(2)	Gross Beta	(4)	(7)	Cs-137	3.1×10^{-10}	6.2×10^{-10}
			RE-8018B (I)	(3)	Gross Gamma	(5)	(8)	I-131	2.8×10^{-10}	5.6×10^{-10}
			RE-8018C (NG)	(2)	Gross Beta	(6)	(9)	Kr-85	~~(later)~~ 4.2×10^{-7}	~~(later)~~ 8.4×10^{-7}

Notes:

1. Minimum Detectable Concentration
2. Beta Scintillation Detector
3. Gamma Scintillation Detector
4. 3.3×10^{-11} to 3.3×10^{-6} μCi/cc
5. 7.7×10^{-12} to 3.8×10^{-6} μCi/cc
6. 2.8×10^{-7} to 2.8×10^{-1} μCi/cc
7. 1.7×10^{-11} μCi/cc
8. $4.\!\!?\times10^{-11}$ μCi/cc 4
9. 2.1×10^{-7} μCi/cc

~~*Based on Xe-133~~

10. 3.3×10^{-11} μCi/cc

Amendment 44

STP FSAR 44

ATTACHMENT 5
ST.HL.AE. 1409
PAGE 2 OF 3

TABLE 12.3.4-2 (Continued)

BUILDING VENTILATION MONITORS

Monitor	Service	Sample Location	Detector Number	Detector Type	Analysis Performed	Range (μCi/cc)	MDC (1) (μCi/cc)	Controlling Isotope	Alert Alarm (μCi/cc)	High Alarm (μCi/cc)
RT-8029	MAB Ventilation	MAB El. 41'	RE-8029A (P)	(2)	Gross Beta	(4)	(7)	Cs-137	1.6×10^{-10}	3.3×10^{-10}
		See Fig.	RE-8029B (I)	(3)	Gross Gamma	(5)	(8)	I-131	1.5×10^{-10}	3.0×10^{-10}
		9.4.3-6	RE-8029C (NG)	(2)	Gross Beta	(6)	(9)	Kr-85	~~(later)~~ a 4.2×10^{-7}	~~(later)~~ a 8.4×10^{-7}
RT-8030	MAB Ventilation	MAB El. 41'	RE-8030A (P)	(2)	Gross Beta	(4)	(7)	Cs-137	1.7×10^{-10}	3.5×10^{-10}
		See Fig.	RE-8030B (I)	(3)	Gross Gamma	(5)	(8)	I-131	1.5×10^{-10}	3.1×10^{-10}
		9.4.3-6	RE-8030C (NG)	(2)	Gross Beta	(6)	(9)	Kr-85	~~(later)~~ a 4.2×10^{-7}	~~(later)~~ a 8.4×10^{-7}
RT-8020	EOF Ventilation	EOF Building	RE-8020 (I)	(3)	Gross Gamma	(5)	~~(8)~~ (10)	I-131	~~(later)~~ a 10^{-9}	~~(later)~~ a 10^{-8}

Notes:

1. Minimum Detectable Concentration
2. Beta Scintillation Detector
3. Gamma Scintillation Detector
4. 3.3×10^{-11} to 3.3×10^{-6} μCi/cc
5. 7.7×10^{-12} to 3.8×10^{-6} μCi/cc
6. 2.8×10^{-7} to 2.8×10^{-1} Ci/cc
7. 1.7×10^{-11} μCi/cc
8. 4 $4.\cancel{8}\times10^{-11}$ μCi/cc
9. 2.1×10^{-7} μCi/cc

~~*Based on Xe-133~~ a

10. 3.3×10^{-11} μCi/cc

44

Attachment 6

ATTACHMENT 6
ST-HL-AE-1409
PAGE 2 OF 3

II.F.1 (Continued)

STP Response

Implementation of the NUREG-0737, ITEM II.F.1, instrumentation was integrated with the activities of NUREG-0737, Supplement 1, specifically the Control Room Design Review and the implementation of Regulatory Guide (RG) 1.97 as described in Sections S.5 and S.6 of this appendix respectively. A human factors analysis was performed during the Control Room Design Review.

Appendix 7B, Table 7B.10-1 identifies the variables which satisfy the II.F.1 requirements. Instrumentation adequacy and qualifications are addressed in the analysis presented in Appendix 7B. Table 7.5-1 provides further information as to instrument ranges, qualifications, and display methodology.

Instrumentation calibration requirements will be identified in the Technical Specifications. A calibration program will be in place as identified in Section 13.5.

Instrumentation provided by STP to respond to each attachment of NUREG-0737, Item II.F.1 is further discussed below.

(1) Noble Gas Monitor 40

The STP design includes two wide range noble gas monitors, one for the unit vent and one for the condenser vacuum pumps which detect and measure concentrations of noble gas fission products in plant gaseous effluents during and following an accident. Three detectors with overlapping ranges provide a monitoring range from normal to 10^5 μCi/cc for each monitor.

An adjacent-to-line monitor is provided for each main steamline to monitor the concentration in steam that is released to the environment via the steam generator (SG) safety valves or the SG power-operated relief valves (PORVs).

The range of each monitor is identified in Table 7.5.1. The monitors are powered from either Class-1E emergency standby power or other reliable power sources.

The instrumentation is a part of the Radiation Monitoring System (RMS) as described in Section 11.5. ~~(later)~~

Procedures for use of the instrumentation in determining release rates will be provided as described in Section 13.5.

(2) Iodine/Particulate Sampling

Iodine and particulate isokinetic sampling capability, with onsite analysis, of the plant gaseous effluents is continuously provided, both during and following an accident.

The sampling station for the unit vent is located on the 60' elevation of the auxiliary building. The sampling station for the condenser vacuum pumps is located in the turbine building. The stations are accessible post-accident.

II.F.1 (Continued)

The plant effluent sampling system and analysis capability are further discussed in Section 11.5. ~~(later).~~ 40

(3) Containment High Range Radiation Monitor

Redundant Class-1E, monitors are provided in the Containment Building, 180° apart on the operating deck (elevation 68 ~~ft~~ foot). The range of the monitors is 1R/hr to 10^8 R/hr gamma. 44

(4) Containment Pressure

40

Redundant Class-1E containment pressure and extended range containment pressure monitoring channels provide continuous monitoring and recording of containment pressure. These monitors cover a range of -5 to 180 psig; accuracy of these monitors is approximately ± 3 percent. 49

(5) Containment Water Level

The STP design includes redundant, Class-1E, wide range level monitors. These monitors are located on the containment floor at elevation -11'3" as shown in Figure 7A.II.F.1-1. The wide range monitoring channels provide indication ranging from the containment floor to an elevation corresponding to a water volume of 609,000 gallons. 40 In addition, Class-1E narrow range monitors are provided in the normal and secondary sumps. The narrow range monitoring channels provide indication from the bottom to the top of the normal and secondary sumps.

These level monitors detect the presence of water at discrete predetermined levels. The accuracy of detection at each point is approximately $\pm 1/4$ inch.

The wide range monitors position the detection points more closely at the bottom than at the top. In addition, the detection points of the three monitors are chosen to provide overlap. Above the floor at El. -11'3", for the first foot, detection points are one inch apart (i.e., four points per monitor, 3 inches apart). For the next two feet, detection points are three inches apart. For the next 3 1/2 feet, detection points are six inches apart.

49

The narrow range monitors provided in the normal sump (bottom at El. -17'3") and the secondary sump (bottom at El. -12'3") use a detection point spacing of six inches. The normal sump monitor provides level detection between El. -17'0" and -11'6"; the secondary sump monitor provides level detection at El. -11'9" and -11'3".

These monitoring channels provide continuous monitoring and recording of the containment water level for use in diagnosis of a Loss-of-Coolant Accident.