NORTHEAST UTILITIES

THE CONNECTICUT LIGHT AND POWER COMPANY
WESTERN MASSACHUSETTS ELECTRIC COMPANY
HOLYOKE WATER POWER COMPANY
NORTHEAST UTILITIES SERVICE COMPANY
NORTHEAST NUCLEAR ENERGY COMPANY

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HARTFORD, CONNECTICUT 06141-0270
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March 23, 1984

Docket No. 50-423

B11093

Director of Nuclear Reactor Regulation
Mr. B. J. Youngblood, Chief
Licensing Branch No. 1
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Reference: (1) B. J. Youngblood to W. G. Counsil, Millstone Nuclear Power Station, Unit No. 3 Draft SER.

Dear Mr. Youngblood:

Millstone Nuclear Power Station, Unit No. 3
NRC Radiological Assessment Branch (RAB) Review Meeting

A meeting was held between the NRC-RAB, Northeast Nuclear Energy Company (NNECO), and Stone & Webster at the Millstone site on March 1, 1984 to discuss 9 RAB Draft SER open items contained in Reference (1). Additionally, the NRC reviewers identified concerns regarding 2 previously submitted responses to RAB questions. During the meeting each of the 11 items was discussed. A status of each question was noted as defined by one of the following three categories:

Closed - No further NNECO input or action is needed to resolve the NRC concern.

Confirmatory - NNECO must provide the requested information on the Millstone 3 docket, either by a letter or FSAR amendment.

Open - No resolution possible at this time, NNECO to address.

Attachment I provides the status of items discussed in the meeting. It was agreed that NNECO will transmit a letter to the NRC providing a written response to each item discussed by March 23, 1984. NNECO also agreed to provide all additional information as committed to in confirmatory items as the information becomes available.

Attachment II contains responses to the items discussed in the meeting and formalizes the responses given orally at the meeting. A summary of discussion is also provided for each item. The responses contained herein are being provided as they will appear in FSAR Amendment 8 which is scheduled for submittal in early May 1984.

B001
1/40

If you have any concerns related to the information contained herein or any questions related to our responses, please contact our Licensing representative directly.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY, ET AL
By Northeast Nuclear Energy Company, Their Agent

W G Counsil

W. G. Counsil
Senior Vice President

STATE OF CONNECTICUT)
) ss. Berlin
COUNTY OF HARTFORD)

Then personally appeared before me W. G. Counsil, who being duly sworn, did state that he is Senior Vice President of Northeast Nuclear Energy Company, Applicant herein, that he is authorized to execute and file the foregoing information in the name and on behalf of the Applicants herein and that the statements contained in said information are true and correct to the best of his knowledge and belief.

Lorraine J. D'Amico

Notary Public

My Commission Expires March 31, 1988

Open Items

Radiological Assessment Branch (RAB)

RAB-1 Drawings for High Range Containment Area Monitors (Draft SER Section 12.3.4.1)

Provide drawings which show the locations of area radiation monitors (ARMS).

Response (3/84)

Refer to revised FSAR Figure 3.8-59.

Summary of Discussion (3/84)

Response is acceptable.

Status (3/84)

Closed.

Attachment I

Status of Millstone 3 Draft SER Open Items
and NRC-RAB Questions Discussed at the Meeting With the NRC-RAB March 1, 1984

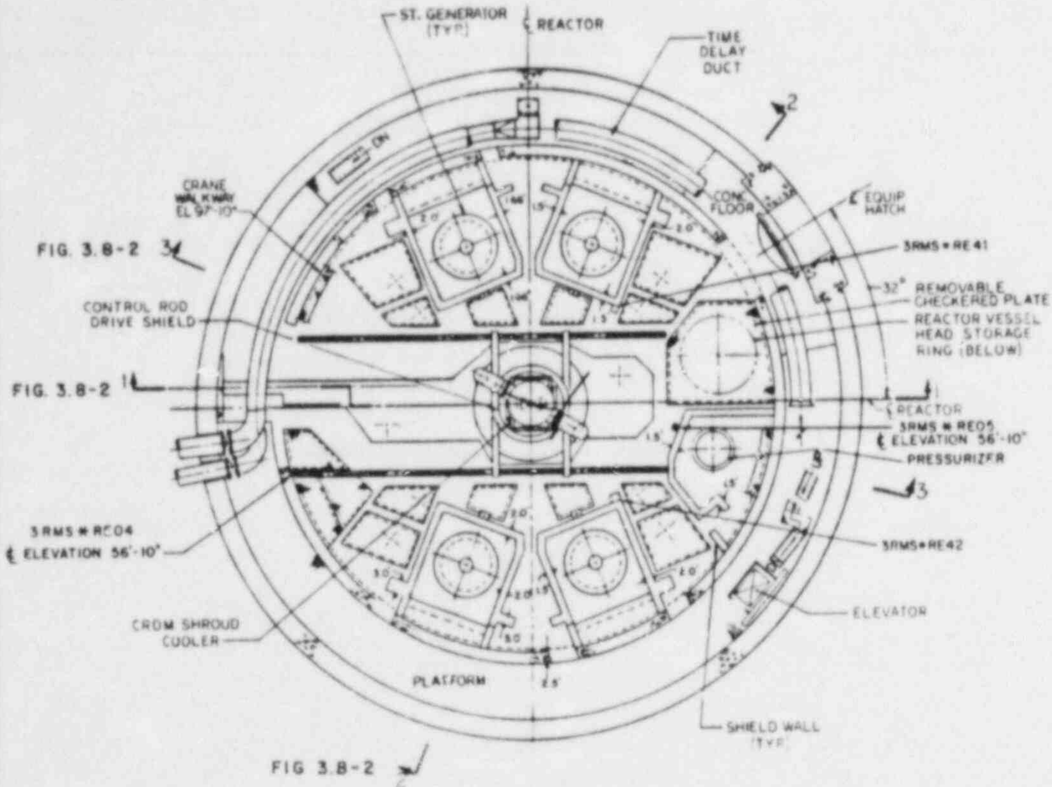
Open Item/Question No.	Status
RAB-1	Closed
RAB-2 (471.13)	Open
RAB-3 (471.16)	Closed
RAB-4	Closed
RAB-5 (471.11)	Open
RAB-6 (471.12, 471.22)	Closed
RAB-7 (471.7)	Closed
RAB-8 (471.23)	Open
RAB-9 (471.30)	Closed
471.1	Confirmatory
471.6	Confirmatory

Summary - Closed - 6
Confirmatory - 2
Open - 3

Attachment II

Responses to NRC-RAB DSER Open Items

and Questions Discussed at March 1, 1984 Meeting



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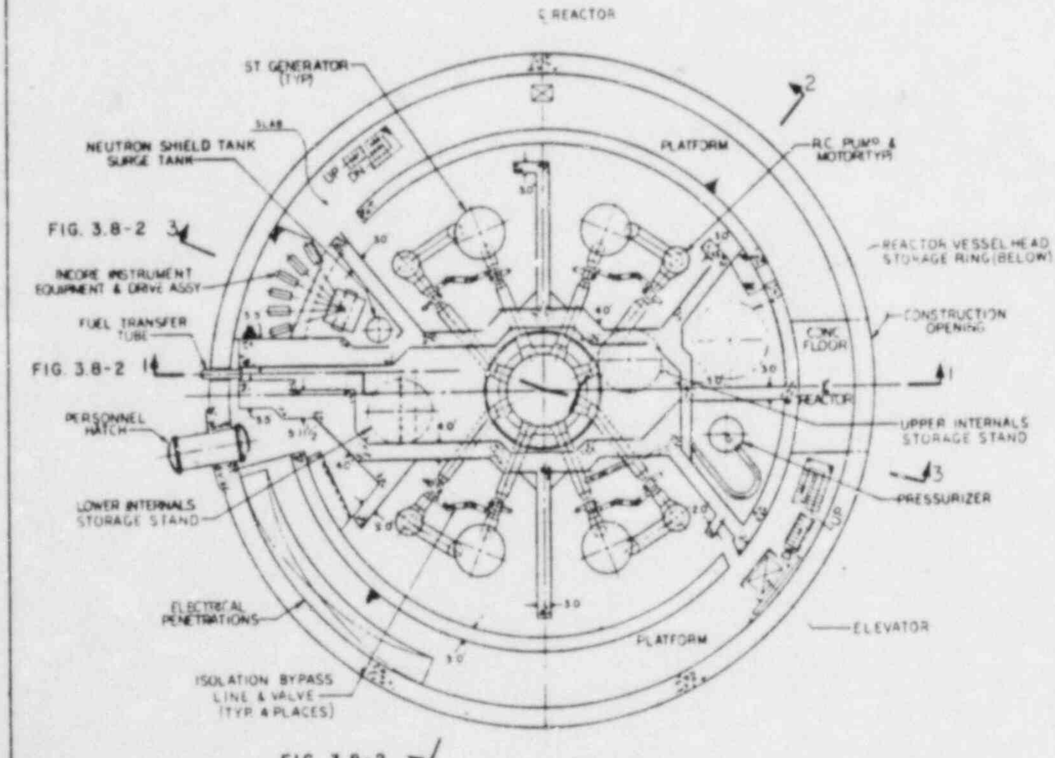

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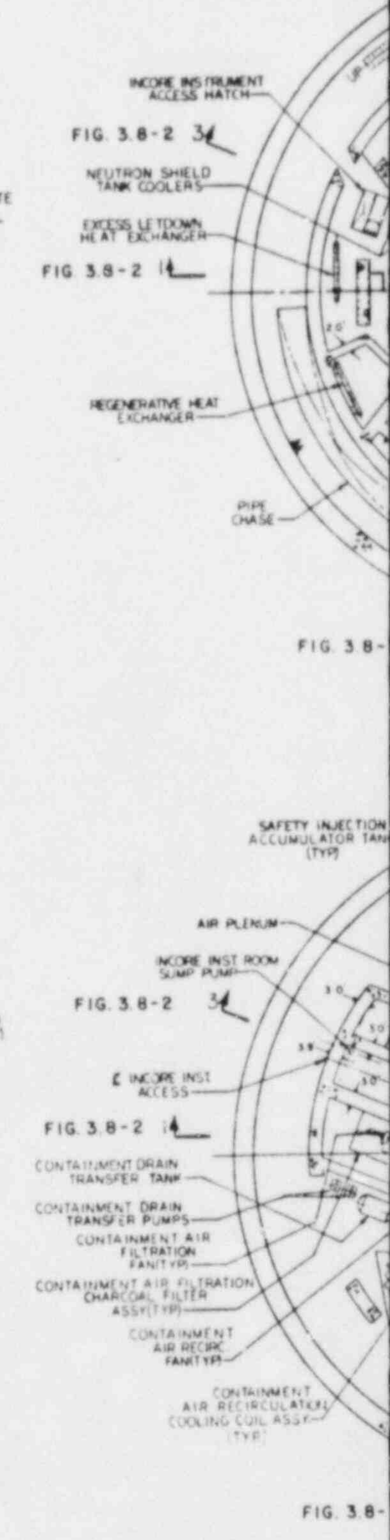

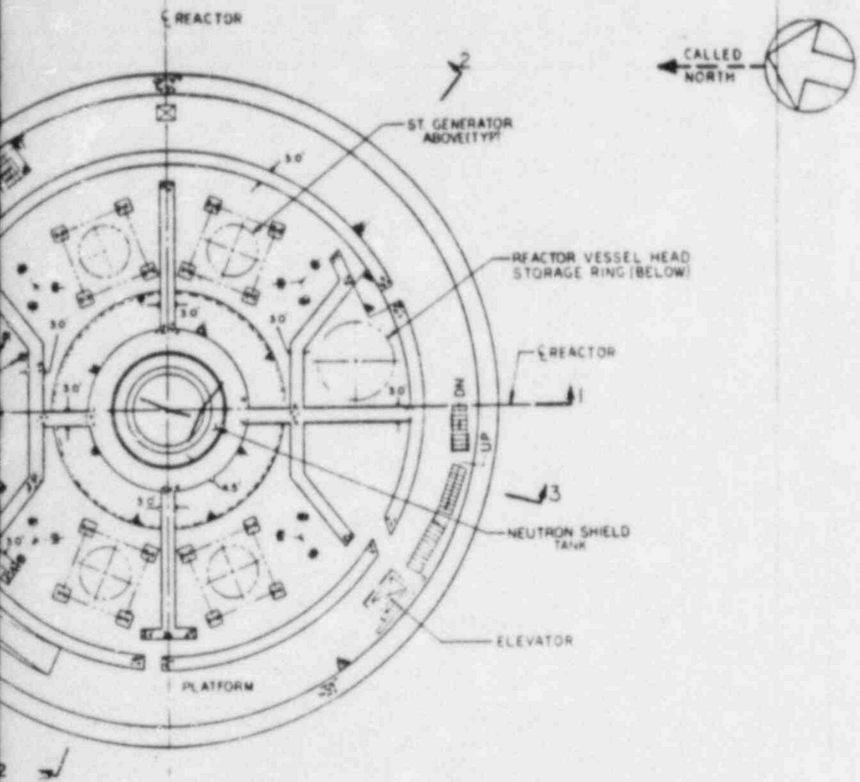

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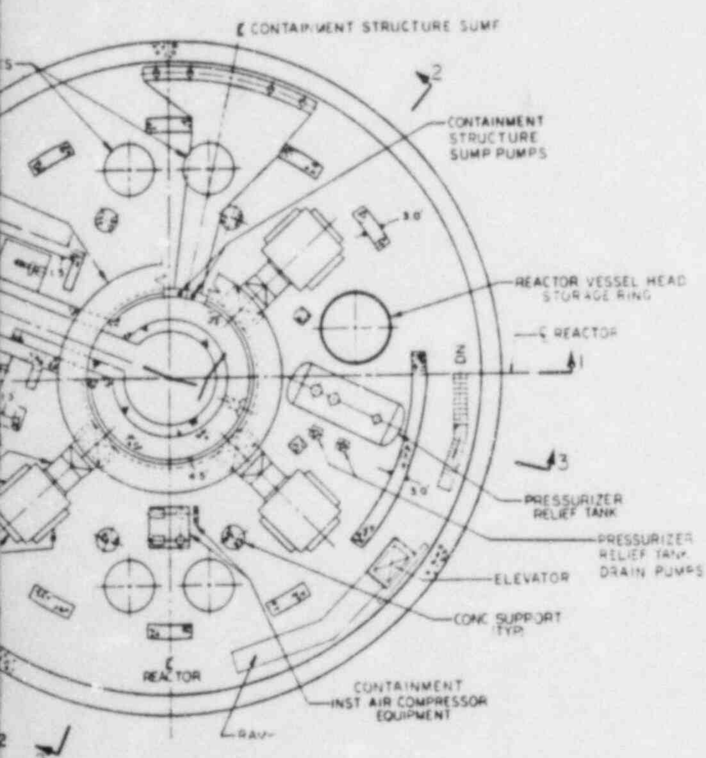


PLAN EL. (-) 24'-6"

TI APERTURE CARD

Also Available On Aperture Card

FIGURE 3.8-59
CONTAINMENT STRUCTURE,
PLAN VIEWS
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

Open Items

Radiological Assessment Branch (RAB)

RAB-2 Analysis for Vital System Operation and TSC Occupancy (12.3.2)

Provide information concerning access routes, doses and shielding requirements for the TSC.

Summary of Discussion (3/84)

The design of the Millstone TSC has not been finalized. A response will be provided at a later date.

Status (3/84)

Open.

Open Items

Radiological Assessment Branch (RAB)

RAB-3 Spent Fuel Transfer Tube (Draft SER Section 12.3.2)

Provide a description of the Spent Fuel Transfer Tube.

Response (3/84)

Refer to response to question 471.16.

Summary of Discussion (3/84)

Response is acceptable.

Status (3/84)

Closed.

Open Items

Radiological Assessment Branch (RAB)

RAB-4 Personnel Exposure Experience (Draft SER Section 12.4)

Expand FSAR Table 12.4-2 to include man-rem exposure data for the years 1981, 1982 and 1983.

Response (3/84)

Refer to revised FSAR Section 12.4.2 and Revised FSAR Table 12.4-2.

Summary of Discussion (3/84)

Response is acceptable.

Status (3/84)

Closed.

12.4 DOSE ASSESSMENT

12.4.1 Estimated Personnel Occupancy in Plant Areas and Resultant Design Dose Rate Criteria

Estimates of allowable personnel occupancy times in all plant areas of Millstone 3 have been made for both full power operation and shutdown/refueling conditions. These estimates are reflected on Figures 12.3-1 through 12.3-4 and 12.3-6 through 12.3-9 which are based upon the 5 Rem per year criterion given in 10CFR20, and the maximum dose rate corresponding to each radiation zone defined in Table 12.4-1.

12.4.2 Estimated Annual Man-Rem Doses

The occupational radiation doses and number of personnel required at a nuclear power station is primarily a function of the amount of routine and special maintenance performed in a given year. According to Stone & Webster Engineering Corporation (1975), data available from operating stations indicate that the average per-worker exposure is approximately one Rem per year for all occupational categories and that 75 percent of this value is related to maintenance, repair, and refueling activities, with the remaining 25 percent related to normal operational exposures.

Millstone 2 personnel exposure experience for the years 1979, 1980, 1981, and 1982 is presented in Table 12.4-2 (NUSCo. 1980). This table presents man-Rem exposures by plant system, regardless of how these exposures were obtained (e.g., during normal operations, maintenance, repair, or refueling activities) and by whom (e.g., by plant operations personnel, plant maintenance personnel, contractor/vendor personnel, etc). However, assuming that Millstone 2 exposures are consistent with those from other operating stations with regard to the percent split between exposures related to maintenance, repair, and refueling activities versus exposures related to normal operations activities, it is reasonable to assume that approximately 10 to 25 percent of the man-Rem exposures tabulated in Table 12.4-2 are normal operations-related. This assumption may be supported by summing the exposure data for the radiation work permit (RWP) category and non-system exposure data and assigning it to the normal operations exposure category. By doing so, the percentage of exposure from normal operations for the years 1979, 1980, 1981, and 1982 are 31, 21, 22, and 15 percent, respectively. These values are in agreement with the data available from operating stations described earlier. Note that the 1982 value of 15 percent is due to extensive steam generator maintenance performed during that year. Given that both Millstone 2 and 3 are pressurized water reactors of roughly similar design and power rating, operated by the same utility in accordance with the same basic operating, maintenance, repair, and refueling procedures, it is reasonable to view the data in Table 12.4-2 as a valid basis for predicting potential personnel exposures at Millstone 3.

8-17

12.4.3 Annual Dose at the Site Boundary and to Construction Workers Due to Radiation from Onsite Sources 1.55

The Millstone 3 radiation shield analysis is based upon conservative 1.58
operating parameters, which include a nuclide inventory associated 2.1
with 1 percent fuel defects. The shield walls are designed to meet 2.2
the Zone I criterion of 0.25 mRem per hour in the yard areas. The 2.3
most significant structures which contribute to the yard dose rate
are the containment, fuel building, waste disposal building, 2.4
auxiliary building, refueling water storage tank, and the boron
recovery tanks.

The calculated dose rate levels in the unrestricted areas are based 2.5
upon full power normal plant operations assuming fuel defects 2.6
producing expected quantities and concentrations of radionuclides
consistent with NUREG 0017. At the site boundary, the calculated 2.7
dose rate is approximately 0.43 mRem per year.

Radiation exposure to construction workers during construction of 2.8
Millstone 3 due to operation of Millstone 1 and 2 is maintained ALARA 2.9
through the use of monitoring, administrative procedures, and
physical barriers (e.g. fences, locked gates, and locked buildings). 2.10
The size of the construction work force is given in Table 12.4-3. 2.11
Also presented in Table 12.4-3 are the projected total man-Rem 2.12
exposures to construction workers for the years 1981 through 1986, 2.13
assuming:

1. That 100 percent of the work force is exposed 2.15

2. That this work force is on the job 40 hours per week and 2.16
50 weeks per year

3. A plume dose rate at ground level outside the Millstone 3 2.17
containment from Millstone 1 and 2 of 0.2 mRem per year 2.18
based upon 8,760 hours per year of occupancy and a plant
capacity factor of 80 percent

4. A skyshine dose rate at ground level outside the Millstone 3 2.19
containment from Millstone 1 of 11.3 mRem per year, based 2.20
upon 8,760 hours per year of occupancy and a plant capacity
factor of 80 percent (NUSCo. 1981)

The total annual dose rate to Millstone 3 construction workers is 2.22
below the maximum permissible level in unrestricted areas as given in 2.23
10CFR20.105.

12.4.4 Measures Taken to Reduce Estimated Man-Rem Doses 2.25

The primary purpose of a radiation protection program is to protect 2.26
personnel against exposure to radiation and radioactive materials and 2.28
to promote the ALARA objectives stated in Regulatory Guide 8.8. This 2.29
is best accomplished through administrative exposure control
procedures, adequate work planning to minimize time spent in 2.30
radiation areas, and safe practices in all activities related to
plant operation and maintenance. The Applicant's radiation 2.31

TABLE 12.4-2

MAN-REM EXPOSURES AT MILLSTONE UNIT 2
BY PLANT SYSTEM

Plant System	1979 (1) Rem	1979 (1) Hours	1980 (1) Rem	1980 (1) Hours	1981 (2) Rem	1981 (2) Hours	1982 (2) Rem	1982 (2) Hours
Chemical and Volume Control	14.4	1,900	50.0	6,820	11.5	1,244	51.4	5,216
Emergency Core Cooling	12.5	1,490	18.4	2,020	39.6	4,823	37.5	2,417
Reactor Containment	16.3	3,480	29.1	10,850	8.3	1,300	10.9	922
Fuel Handling	3.9	732	9.5	680	3.3	182	4.6	229
Safety-Related Display Instrumentation	14.2	1,280	20.8	773	4.0	368	25.1	1,319
Fire Protection Systems	1.1	356	12.8	2,690	0.3	41	1.3	61
Solid Radioactive Waste Management	4.4	543	11.5	9,000	6.1	871	3.9	162
Reactivity Control	4.1	453	50.7	7,930	1.5	248	5.6	569
Reactor Vessel and Appurtenances	28.1	1,910	65.7	4,360	14.7	796	39.2	2,248
Coolant Recirculation	40.1	2,990	72.9	4,146	9.3	351	42.9	1,423
Steam Generato	65.2	3,120	64.0	4,100	187.2	1,163	762.6	27,603
Other Coolant Subsystems and Controls	0.2	20.8	17.4	2,940	0.4	40	0.2	26
Blanket RWP	61.4	15,000	70.6	12,600	62.1	8,120	106.5	24,372
System Code (not applicable)	48.3	8,200	55.8	9,700	46.6	8,163	113	9,217
Total	314.2		549.2		394.9		1,204.7	
Percent of Yearly Total	90.7		89.6		79.3		84.6	

Source:

1. Millstone Unit 2 Personnel Exposure Panel (1979-1980)

2. NUSCo Health Physics Occupational Radiation Exposure Program

Open Items

Radiological Assessment Branch (RAB)

RAB-5 Borated Silicon Shields (Draft SER Section 12.3.2)

Provide a description of the borated silicon shields used in the annular region around the reactor pressure vessel for neutron shielding.

Response (3/84)

Refer to response to question 471.11.

Summary of Discussion (3/84)

NRC to review.

Status (3/1/84

Open.

NRC Letter: May 3, 1983

Question No. Q471.11

In accordance with the acceptance criteria of Section 12.2 of the SRP, NUREG-0800, your FSAR indicates that borated silicon shields are employed in the annular region for neutron streaming. Provide a description of this shield that includes shield thickness, boron loading, and the source strength that your design is based on.

Response:

Refer to FSAR Section 12.3.1 for the response to this question.

3. Optimize the combination of primary and secondary shielding 1.9
by reducing the radiation level from the reactor so that it 1.10
is commensurate with radiation levels from other sources

The primary shield consists of a water filled neutron shield tank and 1.12
a 4.5-foot thick reinforced concrete shield wall. The neutron shield 1.14
tank has an annular thickness of 3 feet and is located between the
reactor vessel and the concrete shield wall. To maintain the 1.16
integrity of the primary shield, a streaming shield fabricated from
borated silicon elastomer (Dow Corning Cylgard 170 or equivalent) is 1.17
installed in the upper annular gap between the vessel flange and the
neuⁿron shield tank and around the nozzles. (Refer to 1.19 neutron *
Figure 12.3-12.)

This shield is designed to minimize the leakage of neutrons to the 1.20
annular region and streaming to the upper levels of the containment, 1.21
thus reducing the neutron dose rate on the operating floor, during
normal operations, to acceptable levels. 1.22

It is estimated that the neutron dose rate in the annulus area 1.28
between the containment wall and the crane wall at the operating 1.29
floor level would not exceed 5 mRem hour with the shield place.

Material composition of the shield is given in Table 12.3.5. 1.30

The total neutron leakage source strength used as the basis for the 1.31
shield design is 6.3×10^{14} neutrons per second. The axial and 1.33
energy distribution of the leakage current is given in Tables 12.3-6
and 12.3-7, respectively.

12.3.1.2 Secondary Shielding 1.36

Secondary shielding consists of reactor coolant loop shielding, the 1.37
crane wall, containment structure shielding, fuel handling shielding, 1.39
auxiliary equipment shielding, waste storage shielding, control room
shielding, and yard shielding.

Secondary shielding thicknesses within the containment structure are 1.40
based on nitrogen-16 being the major source of radioactivity in the 1.41
reactor coolant during normal operation. This source establishes a 1.42
required shielding thickness of the reactor coolant loop shielding,
crane wall, and containment structure wall. The shutdown radiation 1.44
levels in the reactor coolant loop cubicles are established by the
activities of the activated corrosion and fission products in the 1.45
reactor coolant system.

The crane wall provides shielding for limited access to the annulus 1.46
between the crane wall and the containment structure wall and 1.47
provides additional exterior shielding during power operation.

The containment structure shielding consists of a steel-lined 1.48
reinforced concrete cylinder and hemispherical dome. This shielding, 1.49
together with the crane wall, attenuates radiation during full power
operation and during an accident. This shielding keeps radiation 1.50

levels below the design levels at the outside surface of the
containment structure and at the exclusion area boundary (EAB). 1.51

The fuel handling shielding, including both water and concrete, 1.52
attenuates radiation from spent fuel assemblies, control rods, and 1.53
reactor vessel internals to design levels and permits the removal and
transfer of spent fuel and control rods to the fuel pool in the fuel 1.54
building.

The refueling cavity above the reactor is formed by a stainless 1.55
steel-lined, reinforced concrete structure. This refueling cavity 1.56
becomes a pool when filled with borated water to provide shielding
during th refueling operation. 1.57

The depth of the shielding water in the cavity is such that the 1.58
radiation dose rate at the surface of the water from a spent fuel 1.59
assembly does not exceed 2.5 mRem per hour during the short time
intervals when the fuel handling operation brings the spent fuel 2.1
assembly to its closest approach to the pool surface.

The cavity is large enough to provide storage space for the upper and 2.2
lower internals and miscellaneous refueling tools. 2.3

The fuel pool in the fuel building is filled with water to provide at 2.4
least the minimum depth of water shielding specified in Section 9.1.2 2.5
rwn 2.6

TABLE 12.3-5

MATERIAL COMPOSITION OF NEUTRON STREAMING SHIELD

Element	Density (g/cc)
Hydrogen	0.06
Carbon	0.24
Silicon	0.37
Oxygen	0.70
Boron	0.02
Minimum Specific Gravity	1.39

TABLE 12.3-6

AXIAL DISTRIBUTION OF REACTOR VESSEL LEAKAGE CURRENT

Height from Core Midplane (cm)	Relative Leakage Current
0	1.0
11.76	1.0
50.52	1.0
82.02	1.0
112.02	8.89E-01[1]
139.52	6.89E-01
158.52	4.98E-01
172.02	3.57E-01
181.20	2.68E-01
191.02	1.85E-01
206.02	1.05E-01
220.24	5.85E-04
228.52	3.96E-04
235.02	2.82E-04
262.0	1.0E-04
293.0	3.8E-04
338.0	1.2E-04
366.0	6.0E-04
402.0	3.4E-04
445.0	2.4E-04

NOTE:

1. 8.89E-01 = 8.89×10^{-1}

TABLE 12.3-7

ENERGY DISTRIBUTION OF REACTOR VESSEL LEAKAGE CURRENT

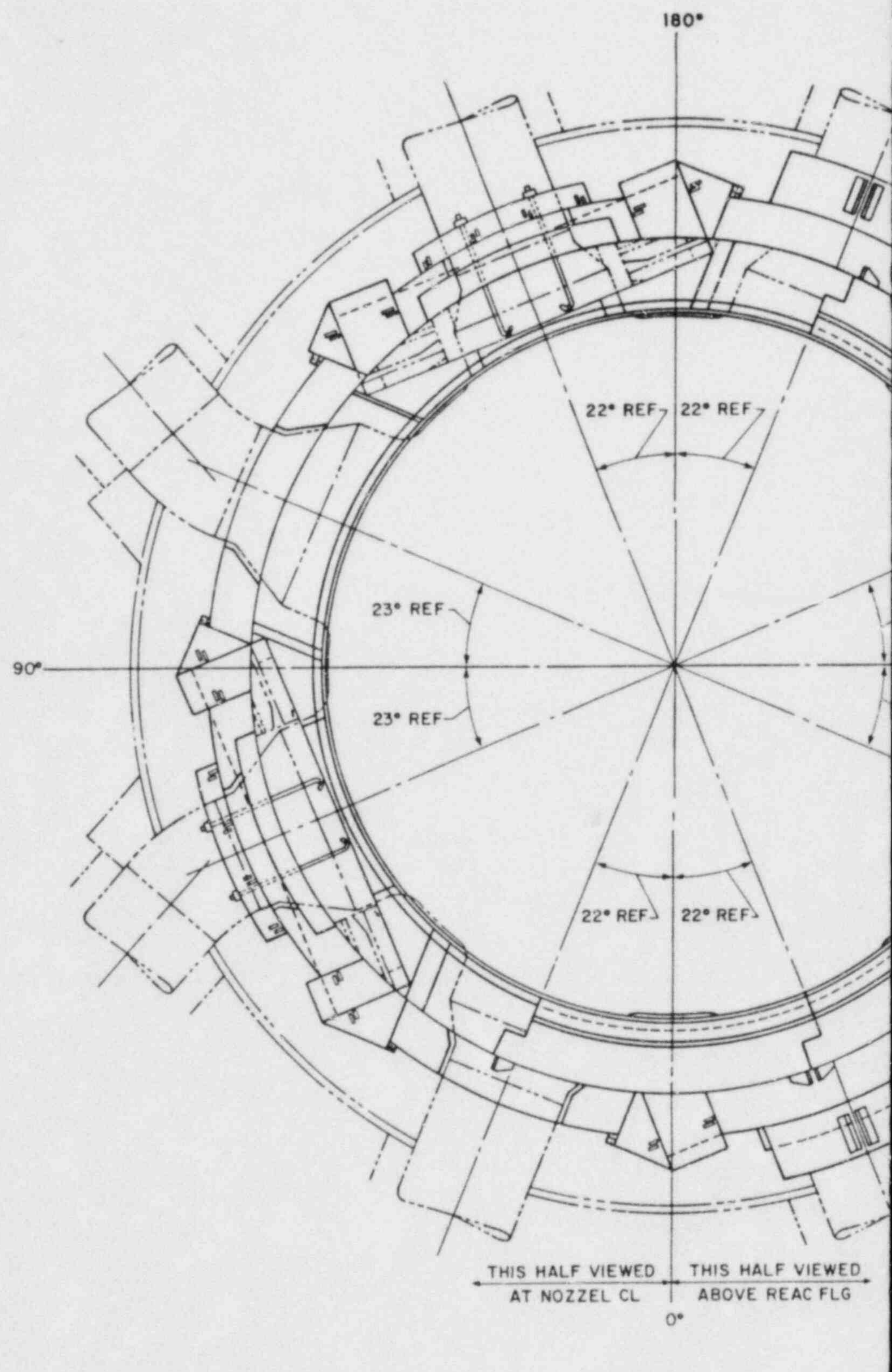
Lower Energy Limit (Mev)	Fraction
7.79	8.99E-04
6.07	2.75E-03
4.72	3.69E-03
3.68	2.47E-03
2.87	6.19E-03
2.23	1.26E-02
1.74	2.23E-02
1.35	3.15E-02
1.05	6.09E-02
8.21E-01[1]	9.61E-02
3.88E-01	2.44E-01
1.11E-01	3.54E-01
4.09E-02	9.28E-02
1.5E-02	4.81E-02
5.53E-03	2.18E-02

471.11

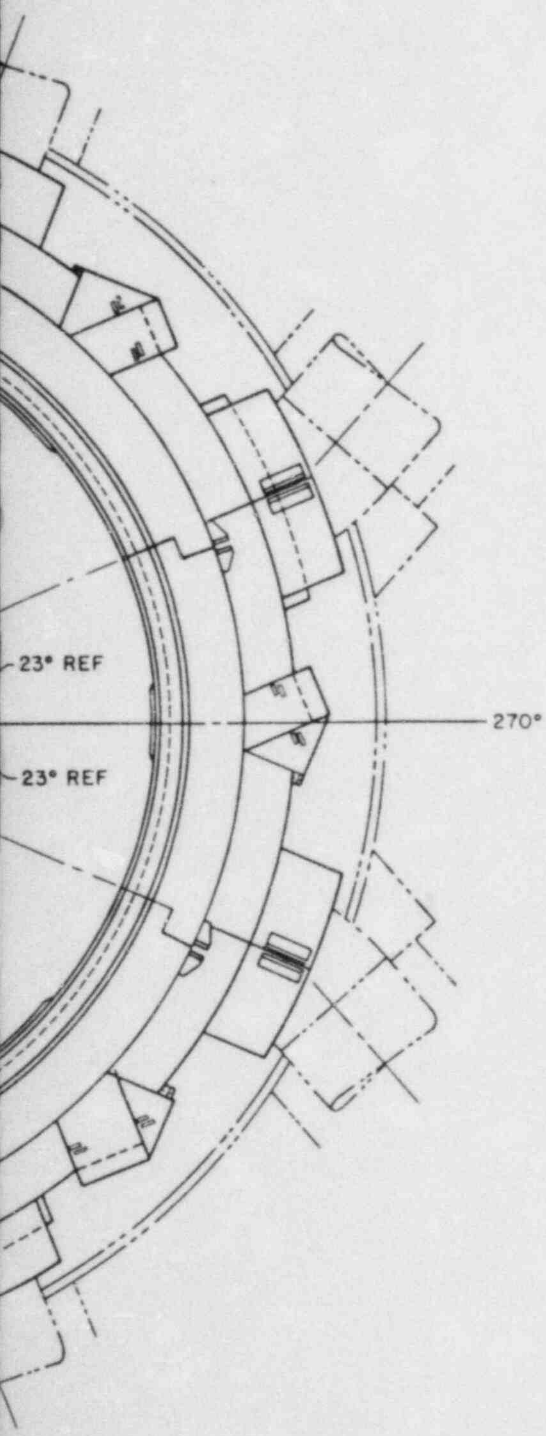
NOTES:

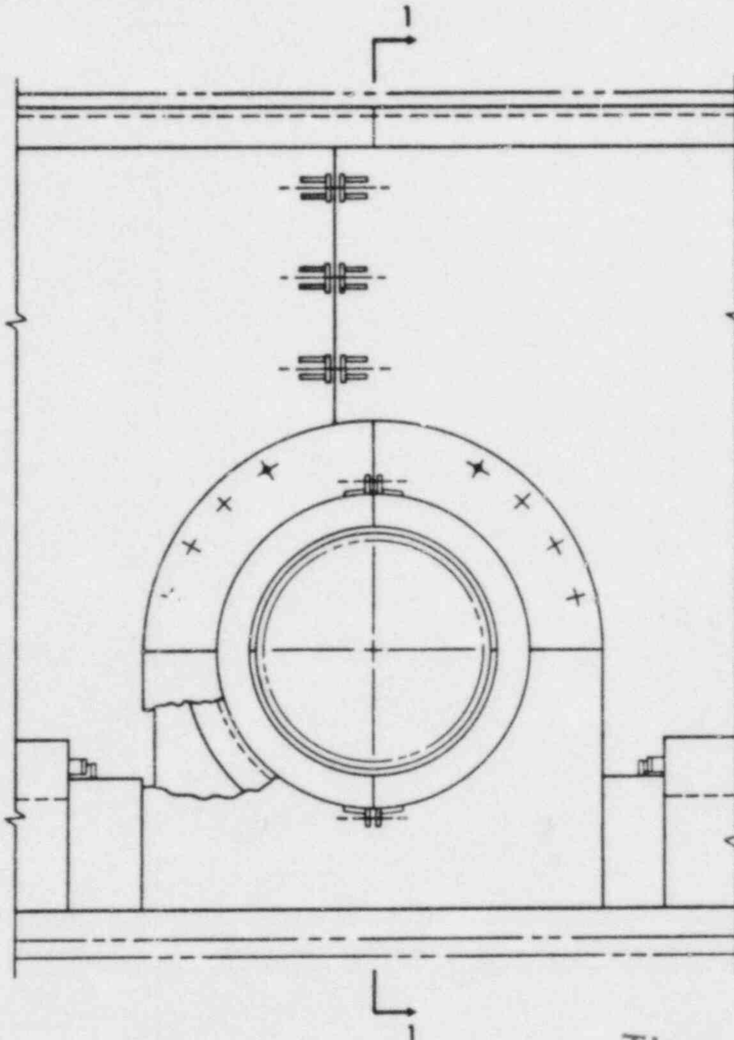
1. 8.21E-01 = 8.21×10^{-1}

The total leakage source strength is 6.3×10^{14} neutrons/sec.

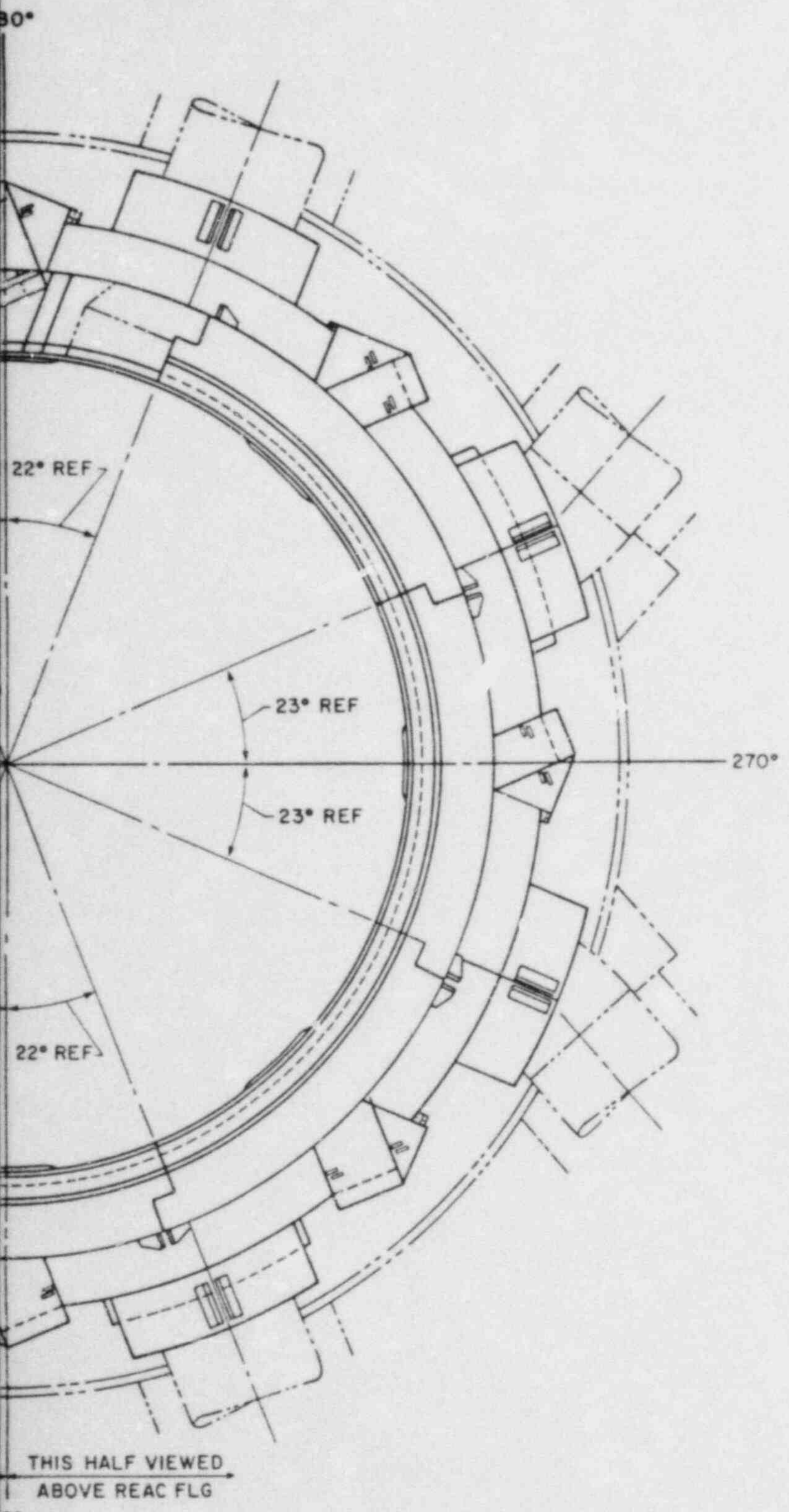


PLAN
REACTOR VESSEL NEUTRON SHIELD

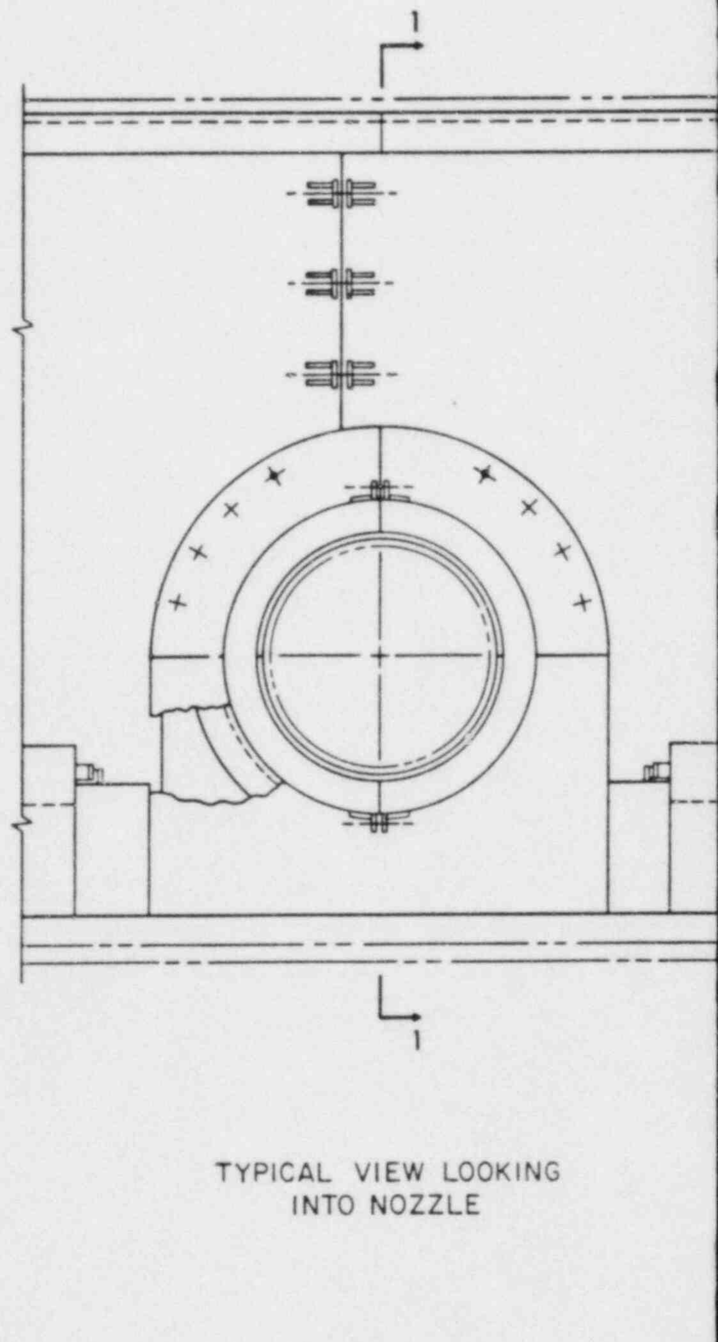




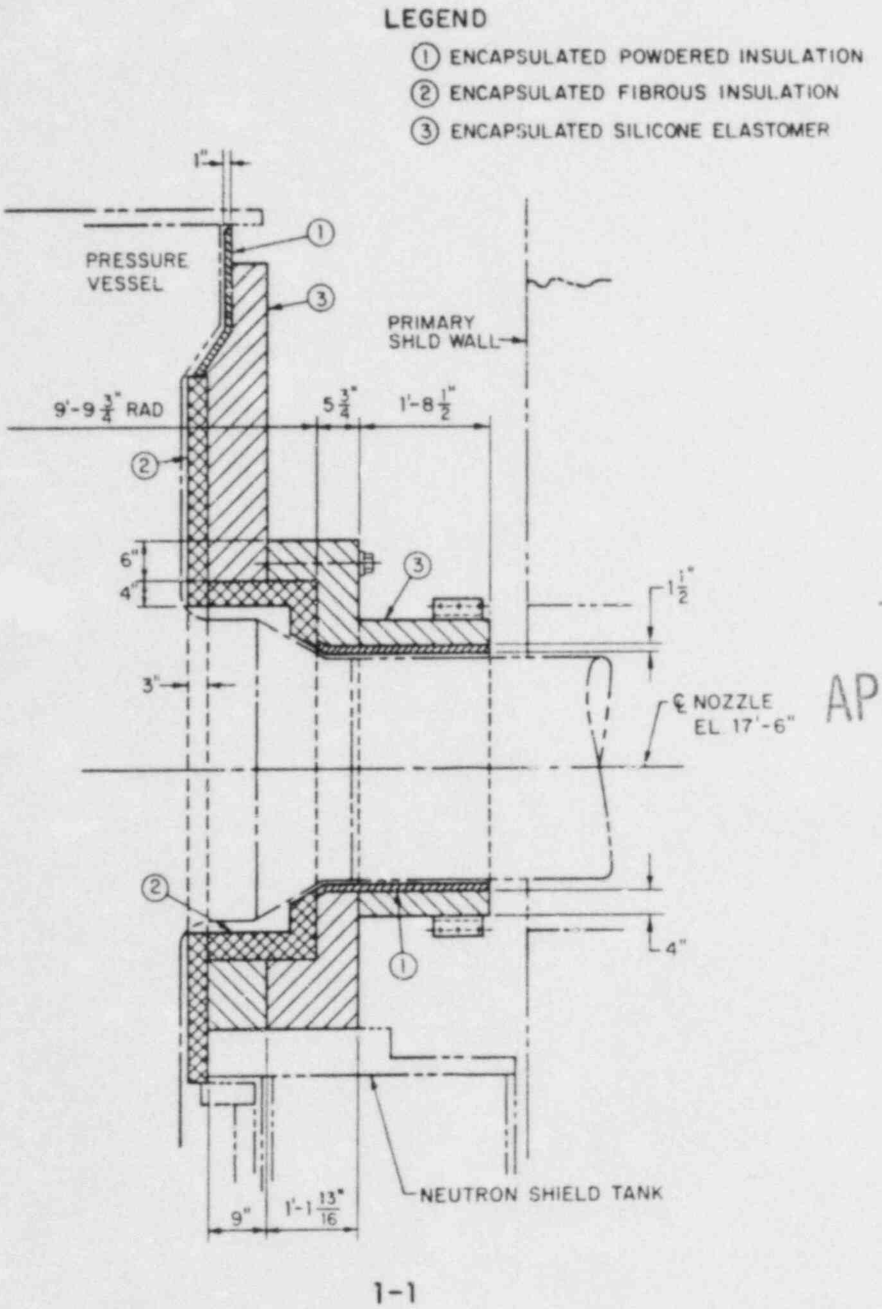
TYPICAL VIEW LOOKING INTO NOZZLE



LAN
NEUTRON SHIELD



TYPICAL VIEW LOOKING
INTO NOZZLE



TI APERTURE CARD

Also Available On Aperture Card

FIGURE 12.3-12
UPPER REACTOR CAVITY
NEUTRON SHIELD
MILLSTONE NUCLEAR POWER PLANT
UNIT 3
FINAL SAFETY ANALYSIS REPORT

Amendment 8 May 1984

8404030411-03

Open Items

Radiological Assessment Branch (RAB)

RAB-6 Review Against R. G. 8.8 Criteria (Draft SER Section 12.1.2)

Provide a design review against R. G. 8.8 criteria.

Response (3/84)

The Millstone Unit 3 design review against Regulatory Guide 8.8 criteria is approximately 75% complete. The present schedule for completion of the review is July, 1984.

No commitment was made to supply the results of the design review in the responses to Question No.'s Q471.12 and Q471.22, dated May 3, 1983. Hence no additional information is intended to be submitted. Details of the design review are on file with the NUSCO Radiological Assessment Branch for review.

Summary of Discussion (3/84)

Documentation of the design review including resolution of non-compliances will exist on file in the NUSCO Radiological Assessment Branch and can be audited by the NRC. The reviewers found this to be acceptable.

Status (3/84)

Closed.

Open Items

Radiological Assessment Branch (RAB)

RAB-7 Dose Assessment for Inhalation and Submersion (Draft SER Section 12.4)

Provide a tabulation of the maximum expected doses to personnel caused by airborne radioactivity from inhalation and submersion.

Response (3/84)

Refer to FSAR Section 12.4.5 and Table 12.4-4. The numbers given are maximum exposures based on Millstone Unit 2 exposure histories. Credit for respirator protection factors has been taken. Note that the submersion exposures are included and the values represent non-outage conditions, (during outages the values are substantially lower and the probability of exceeding MPC limits is much less.)

Summary of Discussion (3/84)

Response is acceptable.

Status (3/84)

Closed.

protection program at Millstone 1 and 2 is currently a functioning reality. This program will also be implemented at Millstone 3.

All new employees will receive a radiation protection orientation prior to their work assignment in the control area. This orientation will cover all pertinent radiation protection practices and procedures to a degree sufficient to allow an employee to perform his assignment without incurring unnecessary radiation exposure.

During extended periods of maintenance, department heads will be furnished a daily report of exposure of the personnel serving under them. This system will permit a more equal distribution of exposure to all personnel.

The procedure for obtaining a Radiation Work Permit (RWP) will be followed to provide radiation safety and awareness in areas where a potentially significant exposure of personnel might occur. All personnel entering a high radiation area, an airborne radioactivity area, or a contaminated area must be listed on a properly executed RWP. Work approval will not begin until the RWP has been signed by the immediate supervisor, health physics, and the shift supervisor consistent with the site radiation protection procedures.

Frequent surveys will be taken in all RWP areas, and appropriate action will be taken to reduce radiation levels as low as reasonably achievable by removal of unnecessary sources of radiation from access ways and operating stations, and by shielding of radiation with portable or temporary sheilding. Good housekeeping practices and decontamination will also aid in reducing radiation levels.

12.4.5 Maximum Expected Inhalation and Submersion Doses to Personnel Due to Airborne Radioactivity

Table 12.4-4 provides a tabulation of the maximum expected inhalation and submersion exposures to personnel due to airborne radioactivity. The numbers given are maximum exposures based on Millstone Unit 2 exposure histories. Credit for respiratory protection has been taken. Note that the submersion exposures are included and the values represent non-outage conditions. (During outages, the values are substantially lower and the probability of exceeding MPC limits is much less.) 8-18

12.4.6 References for Section 12.4

Northeast Utilities Service Company (NUSCo.) 1980. Northeast Utilities Corporate Management Program for Maintaining Occupational Exposures As Low As Reasonably Achievable. Annual Report, 1980.

NUSCO. (1981). Letter NEC-4459 dated September 4, 1981, Subject: Millstone Unit 3 - Dose Rates to Millstone 3 Construction Workers Due to Normal Operation of Units 1 and 2.

Stone & Webster Engineering Corporation (1975). Stone & Webster Topical Report RP-8A, Radiation Shielding Design and Analysis Approach for Light Water Reactor Power Plants.

TABLE 12.4-4

MAXIMUM EXPECTED INHALATION AND SUBMERSION MPC-HOURS[1]

Inhalation and Submersion (excluding Tritium)

Approximate Number of Individuals per Quarter	Exposure Ranges (MPC-hrs)
80	0-1.99
100	2-9.99
20	10-15

Inhalation and Submersion (Tritium)

Approximate Number of Individuals per Quarter	Exposure Ranges (MPC-hrs)
8	0-1.99
1	2-9.99
1	10-15

NOTES:

1. Based on Millstone Unit 2 exposure history.

   Credit for respiratory factors have been taken. Above values represent non-outage conditions.

Open Items

Radiological Assessment Branch (RAB)

RAB-8 Radiation Protection Manager (Draft SER Section 12.5.1)

The Millstone Unit 3 plant organization shows the Radiation Protection Manager reporting to the Radiological Service Supervisor. It is the staff's position that, in matters relating to radiological health and safety, the individual responsible for the activities of the radiation protection support group has a direct responsibility to both employees and management. This responsibility can best be fulfilled if the Radiation Protection Manager not only has access to the Station Superintendent, but is independent of plant operations management, whose prime responsibility is continuity or improvement of station operability. This apparent lack of independence is an open item.

Response (3/84)

Refer to response to Question 471.23.

Summary of Discussion (3/84)

The NRC-RAB reviewers identified two concerns. They were:

a. Provide an updated resume for the Millstone Radiation Protection Manager (John P. Kangley) which shows compliance with Regulatory Guide 1.8.

b. It is the NRC-RAB's position that the RPM should be a regular member of the PORC to ensure that matters of radiological safety are adequately addressed by the PORC. Appoint the Millstone RPM to the PORC or provide justification for not having the RPM as a PORC member.

The NNECO responses are as follows:

a. An updated resume for John P. Kangley is attached.

b. Appointing the RPM as a member of the PORC would dilute his effectiveness as a manager and weaken the administration of the radiation protection program due to the numerous number of PORC meetings and procedure reviews required of PORC members.

   It is understood by the PORC membership that radiological safety is not the responsibility of just one individual but that it is the responsibility of each and every one of them in their review process. When matters of radiological safety are to be reviewed or are identified in the review process the RPM and/or his designee is required to participate in the meeting. Radiological safety is strictly administered and controlled by the Unit/Station radiation protection procedures. These procedures are the responsibility of the RPM and/or his designee. Any changes or revisions to these procedures require approval and will be presented to the PORC by one of these individuals. Additionally, the entire radiation protection staff has the authority to stop work when in their opinion radiological safety is being compromised.

RAB-8 Cont.

Both the Radiation Protection Manager and PORC are part of the present station organization whose duties and responsibilities are defined in our administrative control procedures. Both the organization and the use of administrative control procedures have been approved by the NRC and have been found by them to adequately address matters of radiological safety. Appointing the Radiation Protection Manager as a member of the Millstone Unit No. 3 PORC would be inconsistent with our present organization and would create a separate organization for Unit No. 3. This would have the effect of weakening the radiation protection program for the whole station.

Status (3/84)

Open.

TABLE 13.1-4 (Cont.)

John P. Kangley - Radiological Services Supervisor

B. S. in Chemistry - St. Francis Xavier University, 1965.

1965 - 1969

Math and Science Teacher, Mt. St. John School.

1969 - 1976

Connecticut Yankee Atomic Power Plant, Haddam Neck Plant - Chemist.

Alternate to the Chemistry/Health Physics Supervisor. (Department Head) 1970 to 1975. Responsible for overall supervision of the department during his absence. Chemistry and Health Physics was a combined department until January 1976.

Alternate to the Health Physicist' 1970 to 1975. Responsible for the radiation protection program and supervision of the Chem/H.P. technicians during his absence. Assigned special projects/duties within the radiation protection group such as supervision of spent fuel shipments; disposal of high-level spent resin liners and special decontamination projects.

Was part of the department review group which evaluated and made recommendations for the improvement of the radiation protection program and it's procedures.

Supervised all Chemistry and radiation protection activities during the evening shift for the major shutdowns/refueling outages (1970, 1971, 1972, 1973, 1975 and 1976). Assisted the Health Physics Supervisor during the day shift refueling outages of 1977 and 1979.

Chemistry Supervisor. (Department Head) 1976 to 3-1980 - Responsible for overall supervision of the Chemistry Department. The Chemistry/Health Physics Department was split into two individual departments in January 1976.

License at Connecticut Yankee. I did not pass the N.R.C. walk through exam and was unable to retake the exam since I had been promoted to Chemistry Supervisor at the Millstone Station and was no longer eligible to hold a S.R.O. License at Connecticut Yankee.

Millstone Nuclear Power Station
Chemistry Supervisor - March 1980 to September 1982. Responsible for overall supervision of the Chemistry Department for all three units.

Radiological Services Supervisor: October 1982 to present
Plans, schedules, coordinates and provides overall supervision for Radiological Services consisting of radiation protection, chemistry, medical facility, radioactive material handling, building services and emergency planning in compliance with applicable federal, state and corporate rules and regulations.

Open Items

Radiological Assessment Branch (RAB)

RAB-9 Description of Activated Corrosion Product Control Features (Draft SER Section 12.3.1)

Describe the features that you have incorporated into your design to maintain occupational radiation exposure ALARA by minimizing and controlling the buildup, transport and deposition of activated corrosion products in reactor coolant and auxiliary systems. Include information on the following steps taken to minimize CO-58 and CO-60, including:

(1) The use of reduced nickel content in systems in contact with reactor coolant.

(2) The low cobalt impurity specification in systems in contact with reactor coolant.

(3) The minimization of high cobalt, hard facing wear materials in the systems in contact with reactor coolant.

(4) The use of high flow rate/high temperature filtrations for systems in contact with reactor coolant.

(5) The selection of valves and packings materials to minimize crud buildup and maintenance.

(6) Provisions for decontamination of components and systems contaminated with activated corrosion products.

(7) The types of cleanup systems for removal of crud from primary coolant during operation.

Response (3/84)

Refer to response to Question 471.30.

Summary of Discussion (3/84)

Response is acceptable.

Status (3/84)

Closed.

NRC Letter: February 17, 1984

Question No. Q471.30

Describe the features that you have incorporated into your design to maintain occupational radiation exposure ALARA by minimizing and controlling the buildup, transport, and deposition of activated corrosion products in reactor coolant and auxiliary systems. Include information on the following steps taken to minimize Co-58 and Co-60, including:

1. The use of reduced nickel content in systems in contact with reactor coolant.

2. The low cobalt impurity specification in systems in contact with reactor coolant.

3. The minimization of high cobalt, hard facing wear materials in the systems in contact with reactor coolant.

4. The use of high flow rate/high temperature filtrations for systems in contact with reactor coolant.

5. The selection of valves and packing materials to minimize crud buildup and maintenance.

6. Provisions for decontamination of components and systems contaminated with activated corrosion products.

7. The types of cleanup systems for removal of crud from primary coolant during operation.

Response:

1. Cobalt-59 appears as a residual impurity and Cobalt-58 is formed by the nickel used as one of the components in the manufacture of steam generator tubing material, Inconel 600. Any reduction in the nickel content must consider the cost, availability of materials, and the effectiveness in the reduction of radiation levels. FSAR Table 12.3-5 shows the maximum weight percent of residual cobalt allowed in metals in contact with primary coolant. The table shows that a maximum of 0.1 weight percent cobalt is the limit for Inconel. A reduction in this level would be cost prohibitive.

2. With respect to reduction of the source exposure, Table 12.3-5 lists the NSSS vendor's limits on the cobalt content of incore materials and the materials wetted by the reactor coolant.

3. Materials choice is based upon consideration of superior wear characteristics, reduced maintenance requirements, and contribution of radiation source. Experience shows that

alternate materials do not have the required wear characteristics
of the Haynes Alloy and Stellite. 1.42

4. High flow rate/high temperature filtration systems, (graphite 1.43
beds or electromagnetic filters), are not installed at 1.44
Millstone 3. 1.45

5. High pressure and temperature graphite packing is used in the 1.46
primary system valves. Primary packing, a lantern ring, and 1.47
secondary packing are used in the stuffing box. The use of a 1.48
lantern ring allows a leak off line and drain system to be
incorporated into the valve. The drain system is a closed system 1.49
and prevents the leakage of contaminated coolant within the
containment building. Additionally, Westinghouse uses the bolted 1.50
body-to-bonnet forgings concept on their manufactured valves.
This design facilitates disassembly/assembly, and eliminates the 1.51
need of gaskets as required by the pressure seal valve design. 1.52
It also permits the use of ultrasonic testing in place of 1.53
radiography during ISI. Valves in the primary coolant system and 1.54
charging system are installed in the "stem-up" position and are
provided leak-off lines and have back seat capabilities, either 1.55
remotely or manually. (Refer to FSAR Section 12.3.2.4.) 1.56

6. Means for flushing and draining of potentially contaminated 1.57
systems are incorporated in the fluid system design of
Millstone 3. (Refer to FSAR Sections 12.3.2.7, 12.3.2.8, and 1.59
12.3.2.9.)

7. The chemical and volume control system at Millstone 3 contains 2.1
two mixed bed demineralizers which are designed to reduce the
concentration of fission and corrosion products in the 2.2
purification stream. The reactor coolant filter collects resin 2.3
fines and particulates from the letdown stream. A letdown filter 2.4
is placed in the letdown line to prevent particulates from
collecting in the mixed bed demineralizers. (Refer to FSAR 2.5
Sections 9.3.4.2.2.4, 9.3.4.3.2, and Table 9.3-5).

12.3.2.15 Maximum Expected (Technical Specification Limit) 1.9
Failures of Fuel Element Cladding and Steam Generator 1.10

Design features such as shielding and radiation zones accommodate 1.13
1 percent fuel defects and primary to secondary steam generator tube 1.14
leaks of 1,370 lb per day.

12.3.2.16 Sampling Stations 1.16

Sample points are provided with sample sinks and ventilation hoods, 1.17
splash screens, and valves located outside each splash screen. 1.20
Samples are provided with recirculation paths behind shield walls at 1.21
sample sinks, with reach rods for operators. 1.22

12.3.2.17 Cobalt Impurity Specifications 1.24

Table 12.3-8 provides a list of specifications for residual cobalt 1.25
weight percentages for materials in contact with reactor coolant. 1.26 471.30
These values represent the maximum limits allowed for component 1.28
manufacture and were chosen with regard to source reduction, cost, 1.29
and materials availability.

12.3.3 Ventilation 1.31

12.3.3.1 Design Objectives 1.32

The function and design bases of the ventilation systems are given in 1.33
Section 9.4. Consistent with these, the following specific 1.35
objectives pertain to radiation protection and the commitment that
occupational radiation exposures will be ALARA, in accordance with 1.36
Regulatory Guide 8.8. 1.37

1. The airborne radioactivity concentrations from radioactive 1.39
sources released into the fuel building and turbine
building, as shown in Table 12.2-11, are small fractions of 1.40
values given in Column 1, Table 1 of 10CFR20, Appendix B. 1.41
Radwaste piping system and process components in the 1.42 471.18
auxiliary building and the waste disposal building are
separated from normally accessed areas by walls, and are 1.43
provided with ventilation systems which supply air from
clean, occupied areas and exhaust from duct openings located 1.44
within the process system cubicles. Ultimately, routine 1.45
plant surveys by plant radiation protection personnel shall
provide appropriate controls and protective measures 1.46
described in Section 12.5.3 when access is needed to areas
which are not normally occupied.

2. Concentrations in areas accessible to administrative 1.47 471.18 ~~41.18~~
personnel are less than 25 percent of the concentrations
given in Column 1, Table II of Appendix B to 10CFR20. 1.50

3. The airborne concentrations in all plant areas are ALARA. 1.55

4. The containment atmosphere filtration system, with only one 1.56
of its two 12,000 cfm fan units in operation, is capable of 1.57
reducing the airborne iodine concentration in the
containment atmosphere to below 1 MPC of I-131 in less than 1.58
16 hours of filter operation under the conditions of 1.59
expected reactor coolant radioactivity concentration and 1.60
leakage described in NUREG-0017.

TABLE 12.3-8

COBALT IMPURITY CONTENT IN MATERIALS IN CONTACT WITH REACTOR COOLANT

	Weight Percent Cobalt (Maximum)
Reactor Internals (SS)	0.12
Reactor Vessel Clad (SS)	0.20
RCS Piping (SS)	0.20
Reactor Internals Bolting Materials (SS)	0.25
RCS Pumps (SS)	0.20
Auxiliary Heat Exchanger Surfaces Exposed to RCS (SS)	0.20
Steam Generators (Inconel)	0.1
Fuel (SS)[1]	0.12
Fuel (SS)[2]	0.08
Fuel (Inconel)	0.10
Fuel (Zircaloy)	0.002

471.30

NOTES:

SS = Stainless Steel

1. Refers to stainless steel outside active region on zircaloy clad fuel (e.g., top and bottom nozzles).

2. Refers to stainless steel inside active region.

Question No. Q471.1 (Section 12.1.2)

Discuss the provisions of your radiation protection plan and how they are consistent with the provisions of NUREG-0761.

Response:

NUREG-0761 was issued in March 1981 in draft form for comment. Numerous comments were provided to the NRC by the industry. Until such comments are incorporated and the NUREG is issued in final form, the Applicant does not consider it appropriate to commit to the requirements of NUREG-0761. However, the Applicant does believe that current Millstone radiation protection procedures and practices comply with the intent of all of the draft NUREG-0761 recommendations. The NRC's special Health Physics Appraisal Program follows very closely the scope of NUREG-0761 in performance of their appraisals. Such an appraisal was performed in August 1980 and the Millstone Health Physics Program was found to be adequate for operation. Weaknesses identified during the appraisal have been subsequently corrected. A continuation of this same program, with appropriate updates and improvements will be implemented at Millstone Unit 3. The basic detail of this program are discussed in Section 12.5 of the FSAR.

Summary of Discussion (3/84)

The NRC-RAB reviewers indicated that the present response is unacceptable and that a comparison of the Millstone 3 radiation protection plan against NUREG-0761 should be performed. Show where the Millstone 3 radiation protection plan differs from the guidance of NUREG-0761.

A comparison is currently being performed. The response to Question 471.1 will be revised and submitted at a later date.

Status (3/84)

Confirmatory.

Question No. Q471.6 (Section 12.3.4)

Indicate whether, and if so, how, the guidance of Regulatory Guide 8.2, 8.8, and 8.12 and ANSI N13.1-1969 has been followed; if not followed, describe the specific alternative methods used.

Response:

Regulatory Guide 8.2 - Guide for Administrative Practices in Radiation Monitoring

The Applicant complies fully with this Regulatory Guide and the referenced ANSI standard N13.2-1969 (same title). These guides are fairly limited in scope and are intended to provide some very general guidelines to companies just developing a radiation monitoring program. The radiation monitoring program at Millstone has been in effect for over 10 years and as indicated in response to 471.1 has been found acceptable when compared against guidelines that are much more detailed in scope than Regulatory Guide 8.2.

Regulatory Guide 8.8 - Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be As Low As is Reasonably Achievable (ALARA) - Revision 3 - 1978.

The Applicant is in full compliance with the intent of each of the guidelines of Regulatory Guide 8.8. Sections 12.1 and 12.5 of the FSAR provide a description of the Applicants ALARA program, radiation protection program and radiation protection facilities and hence describe the basic compliance with Regulatory Positions C.1, C.3 and C.4 of the Regulatory Guide. Because of a favorable evaluation of our ALARA program, Northeast Utilities was one of three utilities invited by INPO to help prepare INPO ALARA Good Practices. Hence, the ALARA procedures to be used at Millstone 3 form the basis for many of the INPO Good Practices for an ALARA program.

In regard to Regulatory Position C.2, "Facility and Equipment Design Features", it should be noted that a significant percent of the station design was completed prior to the issuance of Regulatory Guide 8.8. Regardless, because of the significant experience of Westinghouse, Stone & Webster and Northeast Utilities in the design and operation of nuclear power plants and the growing concern in the 1970's for reducing occupational exposure, ALARA design principles were an important consideration in the design of Millstone Unit 3. To verify this, the applicant is currently performing a detailed review of the Millstone Unit 3 design against each of the individual ALARA design recommendations of Section C.2 of Regulatory Guide 8.8. If any potential ALARA concerns are identified, a cost-benefit evaluation is performed to determine if a plant design change is warranted. To date, the review is approximately 30% complete. Thus far, the review has confirmed that ALARA principles are inherent in the design of Unit 3.

Regulatory Guide 8.12 - Criticality Accident Alarm Systems

Implementation of the guidelines contained in Regulatory Guide 8.12 has been described in the response to Question 471.5.

ANSI N13.1-1969 - Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities.

The design basis of sampling airborne radioactive material implements the considerations described in ANSI N13.1-1969 regarding isokinetic probes, location of sample taps, and the concerns of deposition due to elbows and line losses.

The ANSI N13.1 recommendation of locating sampling points a minimum of five times the major dimension (diameter) downstream of a duct bend has been followed where possible. Where this is not possible, due to duct configuration, at least three diameters plus three feet has been provided downstream of a bend.

Summary of Discussion (3/84)

The NRC-RAB reviewers indicated that the present response, specifically the portion addressing Regulatory Guide 8.8, is unacceptable. The reviewers took exception to the words "Complies with the intent of the guidelines of Regulatory Guide 8.8." They stated that the response should be revised to more clearly indicate where the guidance of Regulatory Guide 8.8 has been followed. In cases where the guidance of Regulatory Guide 8.8 has not been followed, specific alternatives should be described.

The response to Question 471.6 will be revised to address these concerns and submitted at a later date.

Status (3/84)

Confirmatory.