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Nuclear Energy

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The Northeast Utilities System

JAN 2 5 2001

Docket No. 50-423
B18312

RE: 10 CFR 50.90

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Millstone Nuclear Power Station, Unit No. 3
Response to a Request for Additional Information
Technical Specifications Change Request 3-6-00
Fuel Handling Accidents and Ventilation Systems

In a letter dated June 29, 2000,[1] Northeast Nuclear Energy Company (NNECO) requested a change to the Millstone Unit No. 3 Technical Specifications. Many of the proposed Technical Specification changes were associated with revised fuel handling accident analyses. During conference calls conducted on January 10, 17, and 22 of 2001, NNECO addressed questions contained in a facsimile from the Nuclear Regulatory Commission dated December 12, 2000.[2] The purpose of this letter is to transmit the requested written responses, which are contained in Attachment 1.

There are no regulatory commitments contained within this letter.

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(1) R. P. Necci letter to U.S. Nuclear Regulatory Commission, "Millstone Nuclear Power Station, Unit No. 3, Technical Specifications Change Request 3-6-00, Fuel Handling Accidents and Ventilation Systems," dated June 29, 2000.

(2) V. Nerses facsimile to Northeast Nuclear Energy Company, "Millstone Nuclear Power Station, Unit No. 3, Facsimile Transmission, Draft Request for Additional Information (RAI) to be Discussed in an Upcoming Conference Call (TAC No. MA9364)," dated December 12, 2000.

A001

If you should have any questions on the above, please contact Mr. Ravi Joshi at (860) 440-2080.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

Raymond P. Necci
Vice President - Nuclear Technical Services

Sworn to and subscribed before me

this 25th day of January, 2001

Notary Public

My Commission expires SANDRA L ANTON NOTARY PUBLIC COMMISSION EXPIRES MAY 31, 2005

Attachments (2)

cc: H. J. Miller, Region I Administrator
V. Nerses, NRC Senior Project Manager, Millstone Unit No. 3
A. C. Cerne, Senior Resident Inspector, Millstone Unit No. 3

Director
Bureau of Air Management
Monitoring and Radiation Division
Department of Environmental Protection
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Hartford, CT 06106-5127

Attachment 1

Millstone Nuclear Power Station, Unit No. 3

Response to a Request for Additional Information
Technical Specifications Change Request 3-6-00
Fuel Handling Accidents and Ventilation Systems
Questions and Responses

**Response to a Request for Additional Information**
**Technical Specifications Change Request 3-6-00**
**Fuel Handling Accidents and Ventilation Systems**
**Questions and Responses**

In a letter dated June 29, 2000,[1] Northeast Nuclear Energy Company (NNECO) requested a change to the Millstone Unit No. 3 Technical Specifications. Many of the proposed Technical Specification changes are associated with revised fuel handling accident analyses. During conference calls conducted on January 10, 17, and 22 of 2001, NNECO addressed questions contained in a facsimile from the Nuclear Regulatory Commission (NRC) dated December 12, 2000.[2] The questions and associated responses are presented below.

Item 1 - TS 3/4.7.7 and TS 3/4.7.8

1. The licensee's submittal indicates the use of a "dedicated individual." Clarify the term, "dedicated individual."

   Response:

   Two terms, designated and dedicated, were used with respect to the use of administrative controls associated with the proposed Technical Specification changes. A "designated individual" will be used when the person assigned the task may have additional duties provided those additional duties do not prevent completion of the assigned action within the specified time interval. A "dedicated individual" will be used when the sole responsibility of the assigned individual is performance of the required action. The dedicated individual will not have any additional duties.

Item 2 - TS 3/4.7.9

1. How much time does the "designated individual at the control switch" have to "immediately" return the switch to the "auto" position without adversely affecting the availability of the system? Please provide specific information to describe the verification performed to demonstrate that operations personnel can reliably perform the action under simulated accident conditions.

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(1) R. P. Necci letter to U.S. Nuclear Regulatory Commission, "Millstone Nuclear Power Station, Unit No. 3, Technical Specifications Change Request 3-6-00, Fuel Handling Accidents and Ventilation Systems," dated June 29, 2000.

(2) V. Nerses facsimile to Northeast Nuclear Energy Company, "Millstone Nuclear Power Station, Unit No. 3, Facsimile Transmission, Draft Request for Additional Information (RAI) to be Discussed in an Upcoming Conference Call (TAC No. MA9364)," dated December 12, 2000.

Response:

Operation of the system addressed by the proposed change, Charging/Reactor Plant Component Cooling Water (CHS/RPCCW) Pump Area Ventilation System, was discussed. The actions to be performed by a designated individual do not represent any new approach to system operation. The actions to be performed by the designated individual reflect how the system is normally operated, and are covered by normal operating procedures. The manual actions are associated with normal shifting of operating equipment to equalize run times. No manual action is required in response to a design basis accident, except for the remote possibility of a design basis accident occurring when shifting equipment.

2. What is/are the consequences and risk associated with failing to complete the action in the time permitted?

Response:

The CHS/RPCCW Pump Area Ventilation System is a support system for the Supplementary Leak Collection and Release System (SLCRS). The SLCRS establishes a negative pressure in the Secondary Containment following a design basis accident. Failure of this system (e.g., alignment not restored within approximately 60 seconds after the occurrence of a design basis accident) will degrade the performance of the SLCRS such that the negative pressure required by Technical Specifications may not be established. However, the SLCRS will still be able to establish a negative pressure in the Secondary Containment. Therefore, the consequences of the designated individual failing to restore the CHS/RPCCW Pump Area Ventilation System is minimal. In addition, it is very unlikely system restoration will not be accomplished considering the ease of the task (normal system manipulations) and the administrative controls that will be used (procedural guidance and designated individual).

Item 3 - TS 3/4.9

1. Please explain what is meant by the statement, "...provided at least one personnel access door is under administrative control such that the door can be closed within 10 minutes." Specifically, what is meant by "administrative control?" Please provide specific information to describe the training and testing of personnel on the administrative control used to determine the adequacy of the control and whether the control can be reliably performed by those personnel responsible for it. Also, please explain how 10 minutes was determined to be sufficient time to close the door and, from what point in time does the 10 minutes begin.

Response:

Millstone Unit No. 3 Normal Operating Procedure OP 3260A, "Conduct of Outages," provides guidance to plan and manage outage shutdown risk. This procedure establishes the position of Containment Closure Coordinator to ensure that the ability to establish containment closure is maintained, as required, when the plant is shut down. The ability to establish containment closure prior to core boiling is a key aspect of the "Defense in Depth" approach to minimize shutdown risk. (The requirement to establish containment closure was established by Generic Letter 88-17, "Loss of Decay Heat Removal," dated October 17, 1988.) The same approach currently used to establish containment closure will be utilized to support the proposed Technical Specification change. Procedure changes will add the new closure requirement of 10 minutes for the containment personnel access hatch door during fuel handling activities inside containment. A designated individual will be assigned to establish containment closure within 10 minutes following a fuel handling accident inside containment. This individual will be required to stay in close proximity (general area) to the containment personnel access hatch to ensure availability for hatch closure, if required. Whether this designated individual can have other assigned duties will depend on what actions are necessary to close the access hatch door. For example, there may be hoses and cables going through the access hatch that must be removed before the door can be closed. If there are no hoses or cables through the containment personnel access hatch, the designated individual may be able to perform additional activities in the containment personnel access hatch area. If there are hoses or cables through the containment personnel access hatch, the designated individual will likely have no other duties.

If any lines are run through the containment personnel access hatch during fuel handling activities inside containment, they will be labeled with service and supply, and have quick disconnects so they can be isolated quickly and removed. This will ensure the lines can be rapidly removed to allow closure of the containment personnel access hatch.

Technical Specification 3.9.4 currently requires at least one personnel access hatch door to be closed during fuel handling activities inside containment. The proposed change will allow the personnel access hatch to remain open during fuel handling activities inside containment, provided at least one hatch door can be closed within 10 minutes following a fuel handling accident inside containment. The 10 minute closure time does not address evacuation of the personnel inside containment. Removing any obstructions and closing the hatch door within 10 minutes will restore plant configuration to the current requirement.

The 10 minute closure time is consistent with the radiological dose calculation (submitted with the Technical Specifications Change Request) performed to support the requested Technical Specification change to allow the containment personnel access hatch to remain open during fuel handling activities inside containment. The validity of the 10 minute closure time is based on a qualitative evaluation of the tasks involved with containment personnel access hatch door closure, including the door operating mechanisms. The containment personnel access hatch doors are hydraulically operated. To close a containment personnel access hatch door simply requires pushing the close pushbutton which starts a hydraulic pump and closes the door in less than one minute. If the hydraulic pump is not available, the door can be manually pumped closed. Since this is a manual evolution, closure times will vary, but it is reasonable to assume a manual closure time of less than three minutes.

It is extremely unlikely the assumed 10 minute closure time will ever be challenged. As previously stated, the designated individual will be required to remain in the vicinity of the door to be immediately available for door closure. In addition, the number of hoses and cables allowed to pass through the door will be administratively controlled, and provisions to rapidly remove the hoses and cables (e.g., quick disconnects) will be provided. Therefore, after considering the length of time to close the door and the administrative controls that will be in place, NNECO has concluded that a 10 minute containment personnel access hatch door closure time is appropriate.

2. What method was used to determine that the maximum number of cables and hoses expected to be used can be "rapidly removed to allow the door to be closed within the required time period?"

   Response:

   Refer to the response to Item 3, Number 1.

3. Please explain what is meant by, "... a designated individual must be continuously available for door closure." Please explain the difference between a "designated individual" and a "dedicated individual."

   Response:

   Refer to the response to Item 1, Number 1.

Item 4 - TS 3/4.9.12

1. Please provide a more detailed description of the dedicated individual at the opening [of the Fuel Building boundary] to include details such as, where the individual will be stationed; how communications with the control room will be established and maintained; what the method is for "rapidly" closing the opening for building isolation; how rapidly must the opening to be closed and, what basis there is for knowing that this closure can be accomplished within the required time, etc. Please provide specific information to describe the training and testing of personnel to determine whether the action can be reliably performed by those personnel responsible for it.

Response:

The proposed footnotes to Technical Specifications 3.7.7, 3.7.8, and 3.9.12 allow the respective boundaries to be opened intermittently under administrative control. As discussed in the associated Bases, this provision addresses normal entry and egress through the associated boundary doors, with the individual ensuring door closure after use. This provision may also be used for boundary openings if a dedicated individual in constant communication with the Control Room is assigned to ensure the opening is closed when required. At Millstone Station, a dedicated individual is used for very specific evolutions, under very controlled conditions. This will ensure the proper controls are in place to rapidly restore the boundary when using the footnote. In addition, the dedicated individual assigned to restore the Control Room boundary will be stationed within the Control Room boundary, and the dedicated individual assigned to restore the Fuel Building boundary will be stationed outside the Fuel Building boundary.

If the use of a dedicated individual is not appropriate to ensure boundary restoration (e.g., no procedural guidance or actions too complex), the Technical Specification Action Statements (including any proposed changes) will be utilized.

Item 5 - Clarification Issue

1. Attachment 2/Page 9, item #2. Please explain the statement, "the proposed changes do not introduce any new failure modes," the staff considers crediting personnel actions that were not previously credited in the licensee's accident analysis as a potential source for introducing new failure modes. In addition, although "unusual operator" actions might not be required as a result of the proposed changes, the staff believes that, for example, stationing a "dedicated individual at the opening..." (see Bases, 3/4.7.7, Control Room Ventilation System) constitutes an unusual manual action and, therefore, new (previously unanalyzed failure mode(s) may be introduced.

   Response:

   The personnel actions specified in the proposed changes do not represent any new or unusual actions for plant personnel. For example, the actions consist of closing a door, closing an access hatch, and opening a manual valve. These are routine tasks, well within the capability of the individuals assigned to perform the actions. Since these are routine actions, NNECO does not feel they could introduce a new failure. However, since these actions were not previously credited in the approved analyses, it is appropriate to request review and approval of the personnel actions by the NRC before use.

2. Attachment 2/Page 9, item #3. The accidents appear to have been analyzed for radiological consequences to control room operators and to the public (off-site release), with the conclusion that applicable 10 CFR dose limits are not exceeded. Please explain how the analyses have considered the radiological consequences to the "dedicated individual" who is responsible for "rapidly closing" the control room door and to the "designated individual" responsible for closing the personnel access hatch door. In addition, please explain how the radiological analyses and overall time estimates for manual actions have considered the potential effect of a fuel handling accident on personnel who are likely to be in containment during fuel handling operations (e.g, in the event of a fuel handling accident, can every one who is likely to be in the building be evacuated safely within the required time?).

   Response:

   The calculated whole body dose to the designated individual assigned to ensure closure of the containment personnel access hatch door is 2.2 Rem. A copy of Calculation M3FHA-01836R3, "MP3 FHAIC - Dose to Personnel Hatch Operator," is contained in Attachment 2.

   The radiological consequences to the dedicated individuals are addressed by where the individuals will be located, as discussed in the response to Item 4.

   The dose calculations that were performed are consistent with current industry regulations to ensure the health and safety of the public is maintained if a fuel handling accident were to occur. Dose calculations to the individuals inside containment have not been done. This is consistent with our current licensing basis. It is assumed the individuals will evacuate containment as soon as possible. Individuals will then be evaluated to determine the exposure received, and what actions are necessary to mitigate the effects.

3. Please provide a site map/drawing showing the location(s) of the containment personnel access hatch doors and their position relative to other building locations (e.g., does/do the access hatch door(s) open from containment directly to the outside environment or into another/other building(s)?).

   Response:

   The outer containment personnel access hatch door opens directly into the Auxiliary Building. This containment opening does not provide a direct access path to the outside environment. Refer to drawing 12179-EM-6B-13, Machine Location Auxiliary Building Plan EL 24'-6", which is contained in Attachment 2.

Attachment 2

Millstone Nuclear Power Station, Unit No. 3

Response to a Request for Additional Information
Technical Specifications Change Request 3-6-00
Fuel Handling Accidents and Ventilation Systems
Supplemental Information

**Response to a Request for Additional Information**
**Technical Specifications Change Request 3-6-00**
**Fuel Handling Accidents and Ventilation Systems**
**Supplemental Information**

The following additional items are included to support the responses provided in Attachment 1.

1. Calculation M3FHA-01836R3 — MP3 FHAIC - Dose to Personnel Hatch Operator

2. Drawing 12179-EM-6B-13 — Machine Location Auxiliary Building Plan EL 24'-6"

Approved 9-28-99 Effective 10-5-99

# CALCULATION TITLE PAGE

**Total Number of Pages: 9**

**MP3 FHAIC - DOSE TO PERSONNEL HATCH OPERATOR**

**TITLE**

| M3FHA-01836R3 | 0 | CONTAINMENT STRUCTURE | |
|---|---|---|---|
| **CALCULATION No.** | **Revision No.** | **System Name** | |
| NA | CB | 3312 | HATCH |
| **VENDOR CALCULATION No.** | **Structure** | **System Number** | **Component** |

NA
**VENDOR NAME**

1

| NUCLEAR INDICATOR: | Safety Evaluation or Screen Attached | Calc. Supports DCR/MMOD? | Calc. Supports Other Process? |
|---|---|---|---|
| ☒CAT1 ☐RWQA ☐SBOQA ☐FPQA ☐ATWSQA ☐NON-QA | ☐YES ☒NO | ☐YES ☒NO | ☒YES ☐NO |

| | |
|---|---|
| NA | TSCR 3-6-00 |
| **DCR/MMOD No.** | **Reference** |

**INCORPORATES:**

| CCN NO: | AGAINST REV. |
|---|---|
| NA | NA |

**Executive Summary**

This calculation provides an estimate of whole body dose to an individual operating the personnel access hatch at MP3 following a fuel handling accident in containment. This supports a TSCR allowing the personnel access hatch to remain open during fuel movement provided the hatch can be closed within 10 minutes. The whole body dose to the individual is 2.2 rem.

| **Approvals** (Print & Sign Name) | | |
|---|---|---|
| Preparer: Stu Torf | | Date: 1/15/01 |
| Interdiscipline Reviewer: NA | Discipline: | Date: |
| Interdiscipline Reviewer : NA | Discipline: | Date: |
| Independent Reviewer: Jim Wheeler | | Date: 1/16/01 |
| Supervisor: Bill Eakin | | Date: 1/16/01 |

**Installation Verification**

☐ Calculation represents the installed configuration and approved licensing condition (Calculation of Record)

☒ N/A does not affect plant configuration (e.g., study, hypothetical analysis, etc.)

Preparer/Designer Engineer: (Print and Sign) Stu Torf Date: 1/16/01

Approved 11/27/00 Effective 12/12/00

**PassPort DATABASE INPUTs**



Calculation Number: M3FHA-01836R3 Revision: 0

Vendor Calculation Number/Other: NA Revision: NA

CCN # NA QA ☒ Yes ☐ No Calc Voided: ☐ Yes ☒ No

Superseded By: NA Supersedes Calc: NA

Discipline (Up to 10) Z

| Unit (M1, M2, M3) | Project Reference (EWA, DCR or MMOD) | Component Id | Computer Code | Rev. No./ Level No. |
|---|---|---|---|---|
| M3 | NA | NA | NA | NA |
| | | | | |
| | | | | |
| | | | | |
| | | | | |

| PMMS CODES* | | | | | |
|---|---|---|---|---|---|
| Structure | System | Component | Reference Calculation | Rev No. | CCN |
| CB | CMT | HATCH | M3FHA-01791R3 | 0 | NA |
| | | | | | |
| | | | | | |
| | | | | | |
| | | | | | |

*The codes required must be alpha codes designed for structure, system and component.

NOTE: Avoid multiple item references on a line, e.g., LT 1210 A-D requires four separate lines.

| Reference Drawing | Sheet | Rev. No. |
|---|---|---|
| 25212-11133 | 1 | 8 |
| 25212-11140 | 1 | 6 |
| | | |
| | | |
| | | |

Comments:

NA

| Referenced By Calculation | Impact Y | Impact N | AR Reference/Calc Change Ref. |
|---|---|---|---|
| NA | NA | NA | NA |
| | | | |
| | | | |
| | | | |
| | | | |

## TABLE OF CONTENTS



TOTAL NUMBER OF PAGES: 9

# 1. PURPOSE

The purpose of this calculation is to evaluate the whole body dose to an individual assigned to the task of closing the MP3 personnel access hatch in the event of a Fuel Handling Accident in Containment (FHAIC).

# 2. SUMMARY OF RESULTS

The whole body dose to an individual assigned to the task of closing the personnel access hatch door after a FHAIC is 2.2 rem.

# 3. REFERENCES

1. Calc. # M3FHA-01791R3, Rev. 0, "MP3 Fuel Handling Accident in Containment - 10 Minute Closure Time"
2. 13th AEC Air Cleaning Conference, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Design Criterion 19", Murphy and Campe, 3/18/76
3. Regulatory Guide 1.4, Rev. 2, "Assumptions Used for Evaluating the Potential Radiological consequences of a Loss of coolant Accident for Pressurized water Reactors", 6/74
4. Radiological Health Handbook, 1/1970
5. Dwg # 25212-11133, Rev. 8, "Slab El 24'6" & el 28'6" Outline Aux. Bldg
6. Dwg # 25212-11140, Rev. 6, "Plan & Dets - el. 43'6" & 45'6" Reinf - Aux Bldg"

# 4. BASIC DATA AND ASSUMPTIONS

1. The personnel hatch remains open for 10 minutes and the individual assigned for closure remains outside Containment, but next to the hatch.
2. Unless otherwise specified, all data and assumptions associated with fuel damage and ventilation are provided in Reference 1.
3. The area outside of the personnel access hatch is essentially a room with approximate dimensions of 18'(w) x 43'(l) x 19'(h). These dimensions were estimated from References 5 & 6. Although this room is not isolated from the rest of the Auxiliary building, the configuration is such that airborne radioactivity outside this area does not impact the whole body dose to the individual operating the door. The total room volume is 1.5E+04 $ft^3$.
4. Based on actual walkdown of the transit path from the hatch to the Auxiliary Building exit, total transit time is 30 seconds. This will be added to the 10 minute residence time for the individual.
5. 1 roentgen is approximately 1 rem

# 5. METHOD OF ANALYSIS

Dose to the individual standing outside the personnel hatch for 10 minutes, post-FHAIC will be calculated as described below.

This approach will assume that the release from the refuel pool is dispersed in 10% of the Containment atmosphere. This airborne concentration in containment is assumed to be present in the room outside the personnel access hatch. This is a very conservative approach to the source term because it defines the worst case source term which will then be held constant for the duration of exposure. From this concentration, the finite volume whole body (gamma) dose will be determined for the residence time (10 minutes) and transit time (30 seconds). Whole body dose calculations will be consistent with the gamma dose calculation methods in Reference 3. Finite volume correction factors for gamma dose will be determined based on Reference 2.

The semi-infinite gamma dose rate equation from Reference 2 is:

semi-infinite gamma dose rate, rad/sec = $0.25 * \overline{E}_\gamma * \chi$

where:

E is the average gamma energy per disintegration, Mev/dis
X is the concentration of the isotope, Ci/m3.

The semi-infinite dose rate will be adusted for finite volume per Reference 2.

The finite volume dose rate is multiplied by the residence and transit time to get the whole body dose for the hatch operator.

# 6. BODY OF CALCULATION

## 1. DOSE DETERMINATION ASSUMING MIXING IN 10% CONTAINMENT VOLUME

It is assumed that the release from the refuel pool mixes instantaneously and homogeneously with 10% of the containment atmosphere. The use of 10% of containment volume is an assumption based upon engineering judgement. This is a reasonable assumption when considering the path that must be taken to get from the refuel pool to the personnel hatch. This path requires changes in elevation from the refuel floor (51' el.) to the personnel hatch (24'6" el.) as well as traversal over and around the crane wall. It should also be noted that based on Reference 1, containment purge is not in operation so there is no driving force to exhaust the activity from containment via the purge lines nor is there any appreciable ventilation to circulate air from the pool directly to the hatch.

In order to use the dose equation from Reference 3, the average gamma energy per disintegration must be determined. Table 1 below lists those averages. Gamma energies are extracted from page 13 of Reference 1.

TABLE 1 - AVERAGE ENERGY PER DECAY

| Isotope | Average Gamma Energy per disintegration, Mev/dis |
| --- | --- |
| I-131 | 3.53E-01 |
| I-132 | 0.00E+00 |
| I-133 | 0.00E+00 |
| I-134 | 0.00E+00 |
| I-135 | 1.56E+00 |
| Kr-83m | 0.00E+00 |
| Kr-85 | 2.11E-03 |
| Kr-85m | 1.51E-01 |
| Kr-87 | 0.00E+00 |
| Kr-88 | 1.65E+00 |
| Kr-89 | 2.00E+00 |
| Xe-131m | 3.28E-03 |
| Xe-133m | 3.26E-02 |
| Xe-133 | 3.00E-02 |
| Xe-135m | 0.00E+00 |
| Xe-135 | 2.46E-01 |
| Xe-137 | 0.00E+00 |
| Xe-138 | 0.00E+00 |

Also, the nuclide concentration in air must be determined. This is simply the released activity, decayed for 100 hours post-shutdown (from pg 13 of Reference 1) divided by 10% of containment volume (2.32E+06 ft3 per Reference 1) for a dilution volume of 2.32E+05 ft3. This value represents the activity released from 1 fuel bundle and must be multiplied by 1.25 to compensate for the additional 50 rods that are assumed to fail. It should be noted that the multiplier of 1.25 is conservative in that a more accurate multiplier of 1.19 could be used (based on 264 rods in a bundle and failure of 1 bundle plus 50 rods, (264 + 50)/ 264 = 1.19). The activity is listed in Table 2.

TABLE 2 - 100 HOUR DECAYED ACTIVITY FOR 1 FUEL BUNDLE + 50 RODS

| Isotope | 100 hour Decayed activity, Ci |
| --- | --- |
| I-131 | 8.41E+02 |
| I-132 | 1.41E-10 |
| I-133 | 9.78E+01 |
| I-134 | 5.89E-32 |
| I-135 | 8.00E-02 |
| Kr-83m | 2.52E-12 |
| Kr-85 | 2.92E+03 |
| Kr-85m | 6.30E-03 |
| Kr-87 | 1.47E-19 |
| Kr-88 | 1.63E-06 |
| Kr-89 | 0.00E+00 |
| Xe-131m | 6.94E+01 |
| Xe-133m | 1.50E+03 |
| Xe-133 | 1.29E+05 |
| Xe-135m | 0.00E+00 |
| Xe-135 | 3.02E+01 |
| Xe-137 | 0.00E+00 |
| Xe-138 | 0.00E+00 |

Applying the semi-infinite cloud formulas to the above information results in the information in Table 3.

TABLE 3 - SEMI-INFINITE CLOUD GAMMA DOSE RATE SUMMARY

| Isotope | Average Gamma Energy per disintegration, Mev/dis | 100 hour Decayed activity, Ci | Semi-infinite gamma dose rate, r/hr |
| --- | --- | --- | --- |
| I-131 | 3.53E-01 | 8.41E+02 | 4.07E+01 |
| I-132 | 0.00E+00 | 1.41E-10 | 0.00E+00 |
| I-133 | 0.00E+00 | 9.78E+01 | 0.00E+00 |
| I-134 | 0.00E+00 | 5.89E-32 | 0.00E+00 |
| I-135 | 1.56E+00 | 8.00E-02 | 1.71E-02 |
| Kr-83m | 0.00E+00 | 2.52E-12 | 0.00E+00 |
| Kr-85 | 2.11E-03 | 2.92E+03 | 8.44E-01 |
| Kr-85m | 1.51E-01 | 6.30E-03 | 1.30E-04 |
| Kr-87 | 0.00E+00 | 1.47E-19 | 0.00E+00 |
| Kr-88 | 1.65E+00 | 1.63E-06 | 3.68E-07 |
| Kr-89 | 2.00E+00 | 0.00E+00 | 0.00E+00 |
| Xe-131m | 3.28E-03 | 6.94E+01 | 3.12E-02 |
| Xe-133m | 3.26E-02 | 1.50E+03 | 6.70E+00 |
| Xe-133 | 3.00E-02 | 1.29E+05 | 5.30E+02 |
| Xe-135m | 0.00E+00 | 0.00E+00 | 0.00E+00 |
| Xe-135 | 2.46E-01 | 3.02E+01 | 1.02E+00 |
| Xe-137 | 0.00E+00 | 0.00E+00 | 0.00E+00 |
| Xe-138 | 0.00E+00 | 0.00E+00 | 0.00E+00 |
| | | Total | 5.79E+02 |

The semi-infinite cloud gamma dose rate in containment is 579 R/hr. The gamma dose rate needs to be adjusted for the finite volume of the cloud. Reference 2 provides a correction factor method which is listed below.

GF = 1173/ ($V^{0.338}$)

where GF = correction factor for infinite cloud dose rate/ finite cloud dose rate

so
with the volume affecting the hatch operator from Section 4.3 of 1.5E+04 ft3 (GF = 1173/ $(1.5E+04)^{0.338}$ = 45.5), the resultant gamma dose rate is:

outside of personnel hatch, r/hr = 579 R/hr / 45.5 = 12.7 R/hr

So, for a 10 minute and 30 second exposure period, the hatch operator standing outside the personnel hatch would receive a whole body dose of 2.2 R = 2.2 rem.

# 7. DESIGN VERIFICATION

Evaluating the activity outside the personnel hatch using the 3 and 4 air changes per hour should provide a reasonable basis for evaluating the conservatism of the method used in Section 6 (10% mixing in containment). Ultimately, the air concentration outside the equipment hatch ~~will be~~ were based on 10% mixing will be compared to that from 3 and 4 air changes per hour. It will be assumed that there is no buildup or holdup of activity in the Auxiliary building, because the activity exiting containment will continuously enter and exit the room outside the personnel hatch. This is a reasonable assumption because:

1. without the negative pressure in the Auxiliary Building to draw out the air from containment, there is no driving force to exhaust airborne activity in containment, and
2. to have the negative pressure in the Auxiliary Building capable of exhausting containment air, there must be a tremendous (albeit theoretical) exhaust rate from the Aux. Building which would support the assumption of no holdup or buildup in the Aux. Bldg.

Only 1 isotope need be evaluated for comparison purposes. XE-133 will be used. From Section 6, the XE-133 activity released to containment is 1.29E+05 Ci. This is divided by 10% of the containment volume to provide the concentration in air and is 1.29E+05 Ci/ 2.32E+05 ft3 = 5.6E-01 Ci/ft3. This is the value that is used in the dose calculation in Section 6 and will be used for comparison to an alternate calculation method.

When using 3 air changes per hour as the release method, the following approach is taken. The initial air concentration from XE-133 outside the hatch can be calculated as a function of time. The XE-133 source term outside the hatch at T=0 is equal to the the activity discharged in the volume of air exiting containment at 3 and 4 air changes per hour and does represent the maximum concentration due to the exponential nature of the release rate. This initial activity is dispersed in 1.5E+04 $ft^3$ (Section 4.3). At 3 containment air changes per hour, this is a flow rate of 6.96E+06 CFH (= (2.32E+06 ft3 /containment volume)* (3 containment volumes/ hour)) which is 1.16E+05 CFM, 1.5E+04 $ft^3$ would be displaced in 0.13 minutes. The initial amount of XE-133 exhausted from containment to the Aux. Bldg volume of 1.5E+04 $ft^3$ in 0.13 minutes is 0.65% (based on $1- e^{-3*(0.13/60)}$) of the initial XE-133 activity in containment which is 0.0065 * 1.29E+05 Ci Xe-133 = 839 Ci-XE-133 in the volume outside the access hatch. This results in an initial, and maximum XE-133 air concentration of 839 Ci /1.5E+04 $ft^3$ = 5.6E-02 Ci/ft3. Since there is no buildup or holdup of activity in the Aux. Bldg then 5.6E-02 Ci/ft3 is the maximum concentration and will only decrease with time.

Using the same approach for 4 containment volumes per hour results in a lower fraction of containment XE-133 initially filling the 1.5E+04 $ft^3$ volume. This occurs in 0.097 minutes which results in 0.65% or 832 Ci XE-133 resulting in a concentration of 832 Ci/ 1.5E+04 $ft^3$ = 5.6E-02 Ci/ ft3.

It is apparent that the initial concentration of XE-133 outside the personnel access hatch based on 10% mixing in containment (5.6E-01 Ci/ft3) results in a much higher source term than from a release rate of 3 (5.6E-02 Ci/ft3) or 4 (5.6E-02 Ci/ft3) containment volumes per hour. Please note that the concentrations associated with the two release rate do differ but because of round-off, are the same. Since a constant source term based on 10% mixing was assumed for 10 minutes and 30 seconds in the Section 6 determination, it is extremely conservative in its application towards dose evaluation.

In conclusion, the method used in Section 6 provides conservative results and is aceptable.

# 8. ATTACHMENTS

ATTACHMENT A - REVIEWER COMMENTS

Attachment A
Page 9

# Calculation Review Comment and Resolution Form

(Sheet 1 of 1)

Calculation Number: MP3FHA-01836R3 Revision: 0 CCN NA

Calculation Title: MP3 FHAIC - DOSE TO PERSONNEL HATCH OPERATOR

Calc. Originator: S. M. TORF Reviewer (PRINT): J. L. WHEELER

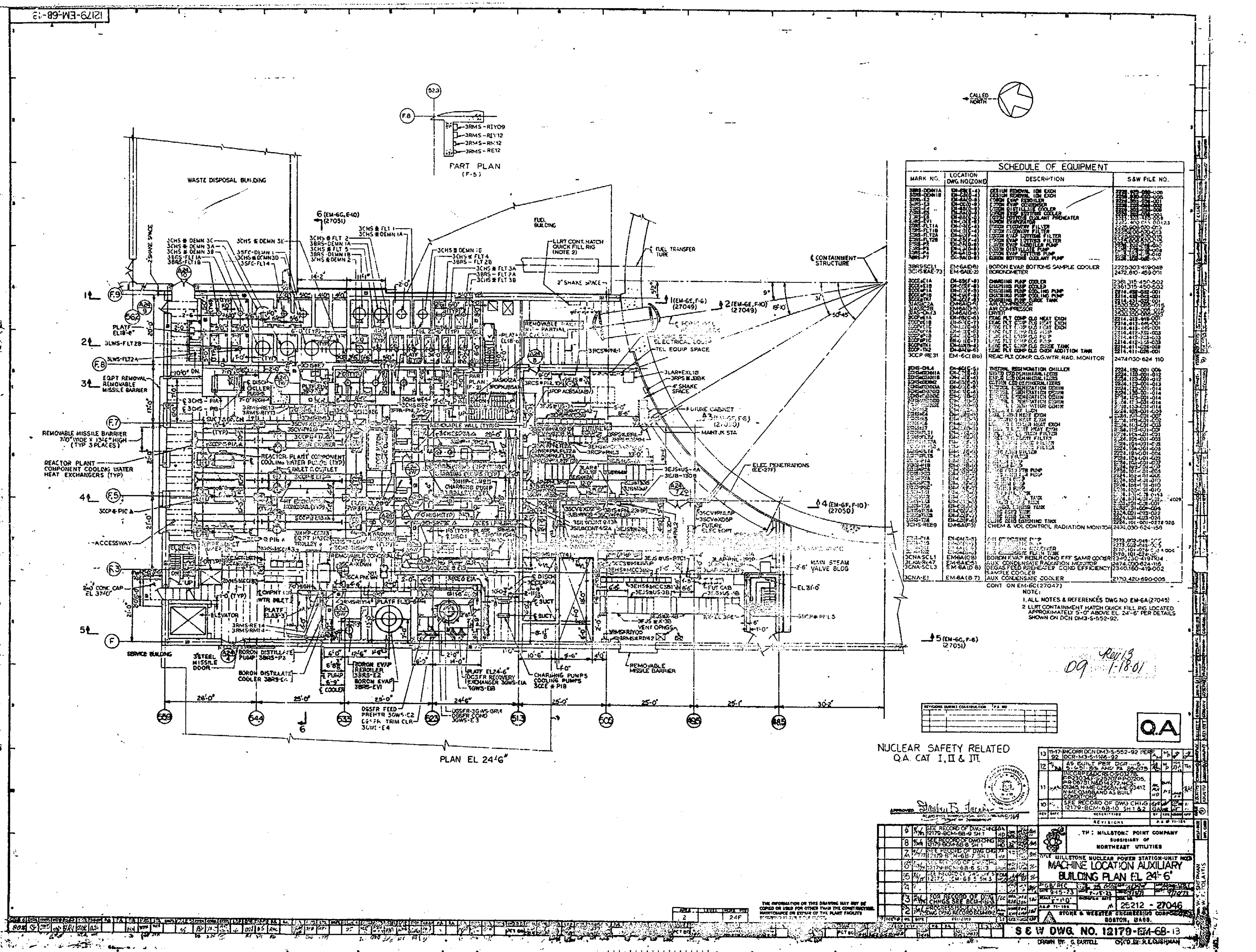
**This form is intended to document significant comments and their resolutions. Typographical errors and other editorial recommendations may be marked up in the calculation text and presented to the originator**

**Review Type** ☞ ☐ **Interdiscipline** ☒ **Independent**

Reviewer (SIGN) [signature] Date: 1/16/01

(signature signifies all comments have been resolved to your satisfaction)

| Item | Page/Section | Comments | Response |
|---|---|---|---|
| 1 | SEC 4.3 | NEED BASIS FOR ROOM DIMENSIONS | DRAWINGS ADDED AS REFERENCES |
| 2 | SEC 6 | NEED CLARIFICATION ON 1.25 MULTIPLIER | DONE |
| 3 | SEC 7 | CONTAINMENT AIR CHANGE RATE WAS CONVERTED TO INCORRECT FLOW RATE | CORRECTED |



12179-EM-6B-13
PART PLAN
(F-5)
3RMS-RIY09
3RMS-RIY12
3RMS-RM12
3RMS-RE12
CALLED NORTH
WASTE DISPOSAL BUILDING
FUEL BUILDING
LLRT CONT. HATCH QUICK FILL RIG (NOTE 2)
FUEL TRANSFER TUBE
CONTAINMENT STRUCTURE
2" SHAKE SPACE
ELECTRICAL LOOP
TEL EQUIP SPACE
FUTURE CABINET
MAINT JK STA
ELEC PENETRATIONS
MAIN STEAM VALVE BLDG
REMOVABLE MISSILE BARRIER
SERVICE BUILDING
3" STEEL MISSILE DOOR
BORON DISTILLATE PUMP 3BRS-P3
BORON DISTILLATE COOLER 3BRS-E3
BORON EVAP REBOILER 3BRS-E2
BORON EVAP 3BRS-EV1
CHARGING PUMPS COOLING PUMPS 3CCE＊P1B
REACTOR PLANT COMPONENT COOLING WATER HEAT EXCHANGERS (TYP)
REMOVABLE MISSILE BARRIER 7'-0" WIDE x 13'-6" HIGH (TYP 3 PLACES)
EQPT REMOVAL REMOVABLE MISSILE BARRIER
ACCESSWAY
ELEVATOR
PLAN EL 24'-6"
SCHEDULE OF EQUIPMENT
MARK NO.
LOCATION DWG. NO.(ZONE)
DESCRIPTION
S&W FILE NO.
3BRS-SCL1 EM-6A(D8) BORON EVAP BOTTOMS SAMPLE COOLER 2225.303-419-048
3CHS-RAE-73 EM-6A(E2) BORONOMETER 2472.610-459-011
3CCP-RE31 EM-6C(B6) REAC PLT COMP CLG WTR RAD MONITOR 2474.030-624 110
3CHS-RE69 EM-6A(F3) CHEM & VOL CONTROL RADIATION MONITOR 2474.030-624-158
3CNA-SCL1 EM-6A(C8) BORON EVAP REBLR COND EFF SAMP COOLER
3CNA-RE47 EM-6A(C8) AUX CONDENSATE RADIATION MONITOR 2474.030-624-116
3CNA-SCL3 EM-6A(D8) DEGASIFIER FEED PREHEATER COND EFFICIENCY SAMPLE COOLER 2340.160-419-002
3CNA-E1 EM-6A(B7) AUX CONDENSATE COOLER 2170.420-690-005
CONT ON EM-6C(27047)
NOTE:
1. ALL NOTES & REFERENCES DWG NO EM-6A(27045)
2. LLRT CONTAINMENT HATCH QUICK FILL RIG LOCATED APPROXIMATELY 5'-0" ABOVE EL. 24'-6" PER DETAILS SHOWN ON DCN DM3-S-552-92.
Rev 13
09 1-18-01
NUCLEAR SAFETY RELATED
Q.A. CAT I, II & III
Q.A
THE MILLSTONE POINT COMPANY
SUBSIDIARY OF
NORTHEAST UTILITIES
MILLSTONE NUCLEAR POWER STATION-UNIT NO.3
MACHINE LOCATION AUXILIARY BUILDING PLAN EL 24'-6"
25212 - 27046
STONE & WEBSTER ENGINEERING CORPORATION
BOSTON, MASS.
S & W DWG. NO. 12179-EM-6B-13

# PACKAGE DIVIDER