NORTHEAST UTILITIES

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HOLYOKE WATER POWER COMPANY
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October 9, 1980

Docket Nos. 50-213
50-245
A01165

Mr. Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

References: (1) D. G. Eisenhut letter to W. G. Counsil dated September 19, 1980.
(2) D. L. Ziemann letter to W. G. Counsil dated February 15, 1980.

Gentlemen:

Haddam Neck Plant
Millstone Nuclear Power Station, Unit No. 1
Environmental Qualification of Electrical Equipment

In partial fulfillment of the license amendment issued by Reference (1), Connecticut Yankee Atomic Power Company (CYAPCO) and Northeast Nuclear Energy Company (NNECO) are hereby providing information regarding the issue of environmental qualification of electrical equipment.

During the site audits conducted in June and July of 1980, representatives from Franklin Research Center (FRC) were provided with preliminary information regarding the radiation service conditions to which the electrical equipment must be qualified. Since that time, this effort has been completed and the final data are provided as Attachments 1 and 2 for the Haddam Neck Plant and Millstone Unit No. 1, respectively. Included in the Attachments are summaries of the methodology used as well as the final radiation data in tabular form. The calculations performed conform to the applicable requirements transmitted by Reference (2).

There are some minor differences between the data provided during the audit and the attached values. Regarding the Haddam Neck Plant, the differences are primarily attributable to the inclusion of a time dependent iodine plate-out source on the containment walls. This calculations was in progress at the time of the audit and was, therefore, not included in the preliminary data provided at that time. Regarding Millstone Unit No. 1, the attached data now includes calculations performed for a main steam line break and for the dose in the control room due to a LOCA. These calculations were not completed at the time of the site audit.

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Also during the site audit, FRC representatives requested justification for maintaining ambient conditions in the following areas:

(1) Screenhouse (Intake Structure)
(2) Gas Turbine Building
(3) Diesel Generator Room
(4) Battery Rooms
(5) Control Room

Brief descriptions of the ventilation systems and relevant P&ID's and one-line diagrams are provided as Attachment 3 to facilitate Staff review of this material. This information is being docketed to support NNECO's position that these areas are classified as ambient, such that Class 1E electrical equipment need not be addressed in accordance with the provisions of Reference (1). A possible exception to this position concerns Item (4), the Battery Rooms. NNECO is currently re-evaluating the necessity to qualify equipment in these areas and will address this matter in subsequent correspondence.

This submittal is being provided to FRC by expedited mail to maximize available review time.

CYAPCO and NNECO are vigorously pursuing the remaining open items in an attempt to fully comply with the requirements of Reference (1). We will keep you advised of our progress.

Very truly yours,

CONNECTICUT YANKEE ATOMIC POWER COMPANY
NORTHEAST NUCLEAR ENERGY COMPANY

W. G. Counsil

W. G. Counsil
Senior Vice President

Attachments

DOCKET NO. 50-213

ATTACHMENT 1

HADDAM NECK PLANT

ENVIRONMENTAL QUALIFICATION OF ELECTRICAL EQUIPMENT

SEPTEMBER, 1980

Connecticut Yankee

Radiation Service Conditions

1. General

Per Reference 1, CYAPCo. was asked to review the radiation service conditions at the Haddam Neck Plant against the Guidelines of Reference 1, Appendix B. Since the procedures developed in Appendix B were based on a PWR containment with a containment spray system designed to enhance iodine removal, the procedures were not directly applicable to a PWR containment with Containment Air Recirculation (CAR) fans.

The procedures were also not applicable for equipment near ESF filters, near systems containing core cooling water and for equipment outside containment. CYAPCo., therefore performed detailed shielding calculations to ascertain the radiation service conditions under a DBA LOCA. A brief description is given below on the method used to calculate the radiation levels inside and outside containment.

2. Scenario

Consistent with the analysis performed in NUREG-0588, this analysis was performed assuming a Design Basis Accident.

3. Source Terms and Distribution of Activity

A. Containment

Consistent with guidance given in NUREG-0588, CYAPCo. assumed an initial release of 50% of the core iodines and 100% of the core noble gas to be uniformly mixed in the containment atmosphere. Removal of iodine in the containment atmosphere was accomplished by both plateout and CAR fan operation.

Iodine plateout on the containment walls was modeled as a competing rate process with the CAR system filters. The differential equation used to describe the activity accumulated on the containment walls is:

$$\frac{dA_s(t)}{dt} = A_c(t)\,\lambda_{pl} - A_s(t)\,\lambda_d$$

where: $A_s(t)$ = activity on the wall surface

$A_c(t)$ = activity in the containment atmosphere

$= A_c(0)\exp(-(\lambda_{pl} + \lambda_d + \lambda_{cf})t)$

λ_{pl} = rate of plateout = 1.27/HR (ref. 2)
λ_d = rate of decay (1/HR)
λ_{cf} = rate at which activity is taken out of containment atmosphere by the CAR system filters

λ_{cf} = 1.332/HR $t \leq$ 14 minutes (ref. 3)
= 3.995/HR $t \geq$ 14 minutes (ref. 3)

The results of the calculation showed that 27% of the initial activity released to the containment atmosphere would be plated out on the walls.

For purposes of conservatism and simplicity, the CAR fan filter source term was based on an initial release into the containment atmosphere of 50% of the core iodines. No plateout of activity on the containment walls was assumed for conservatism.

Core activity levels were based on a TID-14844 source term. In order to simulate the band of gamma energy spectra emitted by the nuclides, the source term was broken up into a seven energy group gamma source. The mean energies used were .4, .8, 1.3, 1.7, 2.2, 2.5 and 3.5 Mev.

B. Outside Containment

Primary Coolant Piping

Piping which carries primary coolant outside containment was conservatively assumed to contain 50% of the core iodine and 1% of the core solid fission product inventory. CYAPCo. has determined that this is a conservative estimate because most of the initial iodine activity in the containment atmosphere will be accumulated on the CAR fan filters and would therefore have no way of getting into the sump water. The only way the iodine could possibly get into the sump would be from the wash down of the plateout source on the walls by the containment spray system. That would mean at the most an iodine source in the sump of 14% of the core iodine inventory. To be consistent with Regulatory Guides and NUREG-0588, we have assumed that 50% of the core iodine would be in the sump water.

The iodine and solid fission product activity was assumed to be diluted by a total water inventory of 148,000 gallons. This includes water stored in the RWST and reactor coolant water.

4. Method of Analysis

A. Containment

i) Gamma

A modified version of the QAD-P5F (Ref. 4) computer code was used to evaluate the gamma dose rates in air at various locations inside the containment. The airborne source was broken up into 6 source regions and was represented by 34,560 point sources.

Credit was taken for the Control Rod Drive Mechanism Shield and the crane wall. The concrete density was conservatively taken to be $2.24 gm/cm^3$.

The plateout source was simulated in the QAD-P5F code by placing the source in a cylindrical volume of small thickness at the steel liner. The plateout source was simulated by a series of 4500 point sources.

The dose rate from activity contained in the sump water was included in the dose calculations at the electrical penetrations. This source was modeled by breaking up the water source volume into 10560 point sources. Attenuation and buildup in the water was taken into account in the calculations.

Twenty-two receptor points were placed at various locations inside the containment and the highest dose rate in each region was used to compute the integrated dose.

Table 1 summarizes the integrated dose at the various locations.

ii) Beta

Beta doses inside the containment were obtained using the semi-infinite cloud dose model. This method is appropriate because of the short range of beta particles in air. The beta doses indicated in Table 1 represent the beta dose in air, not the beta dose the equipment will actually receive. Actual beta doses to equipment are expected to be significantly less due to local shielding considerations (e.g., metal casing around electric motors will attenuate most of the betas).

B. Containment Air Recirculation Fan Motors (CAR)

i) Gamma

The CAR fan filters will accumulate iodine which is taken out of the containment atmosphere.

Three CAR fans were assumed to operate in this analysis and the activity was assumed to be uniformly distributed over the three. The filter was modeled in the QAD-P5F code by 1080 point sources. No self shielding of the source by the charcoal filters was assumed. The activity on the filters was placed in a cylindrical source volume of 4.9×10^6 cm^3 and was assumed to be uniformly distributed within this volume. The dose point was taken about 15 feet from this filter which is the approximate location of the motor.

In addition to the dose from the filter, the motor will receive a dose from the airborne source. The QAD computer code was also used to determine the dose rate from this source. The airborne annular source, and the containment airborne activity above the operating floor was taken into account in the calculations.

The plateout source also contributes to the dose rate at the CAR fan motor. The method used to determine this dose rate is as described in the above section.

The dose rate to the CAR fan motor from activity in the containment sump water was determined to be negligible.

ii) Beta

The beta dose rate in air at the CAR fan motors was determined using the semi-infinite cloud model for the airborne source. The plateout source used Loevingers formula for an infinite plane source. The Loevingers equation is shown below:

$$R(x) = 1.07 \times 10^6 v\, E_{av}\, \alpha\, C_a \left[C \left\{ 1 + \ln (C/vx) - \exp (1-vx/C) \right\} + \exp (1-vx) \right] \quad \text{ref(5)}$$

where:
- $R(X)$ = beta dose rate at distance x from plane
- v = the apparent absorption coefficient for air
- E_{av} = average energy of beta particles (Mev)
- C = a dimensionless parameter whose value depends upon the maximum beta particle energy
- $\alpha = \left[3\, C^2 - (C^2-1) \exp (1)\right]^{-1}$
- x = distance measured in gm/cm^2
- C_a = activity concentration (Ci/cm^2)

C. Outside Containment

i) Primary Auxiliary Building (PAB)

The LOCA dose in the PAB was determined in the cubicle which houses the Residual Heat Removal (RHR) System Heat Exchanger and associated piping.

The dose point was located about 1 foot from the heat exchanger. The dose rate determined at this point was applied to the entire PAB since it was determined to be the location of maximum potential dose rate under accident conditions.

The heat exchanger was simply modeled as a pipe of a cross sectional area equivalent to the primary coolant area flowing through the heat exchanger. No credit for shielding by secondary water or tubing was taken. This method is conservative and simplifies the modeling.

Dose rates from piping sources were determined using a series of parametric piping curves. The curves were generated using the QAD-P5F computer code and give dose rates at various receptor locations for piping sources of different pipe diameters, pipe lengths and decay times. Self shielding and buildup in the water source was accounted for in the calculations. The buildup factors and attenuation coefficients used are the same as those described in the Stone & Webster RP8A shielding manual (Reference 6).

Dose rate estimates were made by determining distances from receptor locations to the piping sources. Parametric curves were then used to determine dose rates.

A summary of the integrated doses from this source is given in Table 1.

ii) Service Building

Accident doses in the auxiliary building were determined from the HPSI lines which run along the outside wall of the PAB. The distance from these lines was determined to be approximately 88 feet and the thickness of the concrete service building wall is 1'-0". The parametric piping curves were used to determine the dose rates. The dose point was taken to be on the surface of the Service building wall.

Table 1 gives a summary of the computed maximum Service Building integrated doses.

5. Normal Operating Doses

The normal operating doses were calculated and added to the post LOCA doses. Table 1 lists the normal operating doses used in the analysis.

Doses in the containment are based upon the highest dose rates observed from several months of recent surveys at 100% power. All neutron doses were adjusted by assuming an 80% capacity factor while gamma doses assumed a 100% capacity factor to take into account crud buildup on piping, valves and equipment.

Doses in other areas are based upon the highest dose rates observed from recent surveys or other information. All gamma doses assumed a 100% capacity factor.

6. Summary

Table 1 summarizes the integrated LOCA doses and normal operating doses for the 40 year life of the plant.

In the PAB, the normal operating dose was taken in many areas. The highest normal operating dose, even though it does not correspond to the location where the LOCA dose was computed, was added to the LOCA dose result. The analysis was performed in this way in order to avoid having to map the LOCA dose in a larger number of locations in the PAB.

References

1) Letter from D. L. Zieman to W. G. Council, February 15, 1980.

2) Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment, Division of Systems Safety Office of Nuclear Reactor Regulation, U.S.N.R.C., Washington, D.C. NUREG-0588 - Date Published 12/79.

3) Letter from D. C. Switzer to D. L. Zieman, March 21, 1978.

4) E. A. Warman, D. R. Rogers, and B. A. Lindsey, AKERN Aerojet Point Kernel Integration Calculation System, RSIC Computer Code Collection CCC-190, Oak Ridge, National Laboratory, Oak Ridge, Tennessee, 1972.

5) J. J. Fitzgerald, G. L. Brownell, and F. J. Mahoney, 'Mathematical Theory of Radiation Dosimetry", Gordon and Breech Science Publishers, Inc., New York, 1967.

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

TABLE 1

CONNECTICUT YANKEE

RADIATION SERVICE CONDITIONS

Receptor Point Location	40 Year[1] Dose (Rads)	Post LOCA Doses[2] (Rads) 1 Hour (Gamma)	B(Beta)	24 Hours γ	β	720 Hours γ	β	6 Months γ	β	Total Dose (Rads) 1 Hour $\gamma+n$	β	24 Hours $\gamma+n$	β	720 Hours $\gamma+n$	β	6 Months $\gamma+n$	β
Containment[3] Reactor Cavity Area	1.8E+6	6.5E+5	2.0E+6	3.6E+6	1.2E+7	7.8E+6	4.8E+7	8.0E+6	5.7E+7	2.5E+6	2.0E+6	5.4E+6	1.2E+7	9.6E+6	4.8E+7	9.8E+6	5.7E+7
Containment[3] Refueling Floor Area	2.7E+5	9.1E+5	2.0E+6	5.0E+6	1.2E+7	1.1E+7	4.8E+7	1.1E+7	5.7E+7	1.2E+6	2.0E+6	5.3E+6	1.2E+7	1.1E+7	4.8E+7	1.1E+7	5.7E+7
Containment[3] Electrical Penetration Area	5.3E+5	7.4E+5	5.8E+6	5.8E+6	6.4E+7	1.2E+7	1.6E+8	1.5E+7	1.7E+8	1.3E+6	5.8E+6	6.3E+6	6.4E+7	1.3E+7	1.6E+8	1.6E+7	1.7E+8
Containment[3] CAR Fan Motor	9.2E+4	1.8E+6	5.1E+6	1.9E+7	6.4E+7	4.2E+7	1.6E+8	4.6E+7	1.7E+8	1.9E+6	5.1E+6	1.9E+7	6.4E+7	4.2E+7	1.6E+8	4.6E+7	1.7E+8
PAB[4]	9.8E+5	4.9E+5	-	3.1E+6	-	7.7E+6	-	1.1E+7	-	1.5E+6	-	4.1E+6	-	8.7E+6	-	1.2E+7	-
Service Bldg.	4.6E+3	< 1	-	4.8E+0	-	1.2E+1	-	1.8E+1	-	4.6E+3	-	4.6E+3	-	4.6E+3	-	4.6E+3	-
Turbine Bldg.	3.5E+1									3.5E+1	-	3.5E+1	-	3.5E+1	-	3.5E+1	-

POOR ORIGINAL

TABLE 1 (CONTINUED)

CONNECTICUT YANKEE

RADIATION SERVICE CONDITIONS

Receptor Point Location	40 Year[1] Dose (Rads)	Post LOCA Doses[2] (Rads) 1 Hour (Gamma)	1 Hour B(Beta)	24 Hours γ	24 Hours β	720 Hours γ	720 Hours β	6 Months γ	6 Months β	Total Dose (Rads) 1 Hour γ+n	1 Hour β	24 Hours γ+n	24 Hours β	720 Hours γ+n	720 Hours β	6 Months γ+n	6 Months β
Diesel Bldg.	2.1E+2	-	-	-	-	-	-	-	-	2.1E+2	-	2.1E+2	-	2.1E+2	-	2.1E+2	-
Screenwell	3.5E+2	-	-	-	-	-	-	-	-	3.5E+2	-	3.5E+2	-	3.5E+2	-	3.5E+2	-
Containment Loop Areas	2.8E+6	6.5E+5	-	3.6E+6	-	7.8E+6	-	8.0E+6	-	3.4E+6	-	6.4E+6	-	1.1E+7	-	1.1E+7	-
Outside Equipment Hatch	3.5E+2	4.6E+4	-	2.5E+5	-	5.6E+5	-	5.8E+5	-	4.7E+4	-	2.5E+5	-	5.6E+5	-	5.8E+5	-
Cable Vault	3.5E+2	7.2E+3	-	4.0E+4	-	8.8E+4	-	9.0E+4	-	7.6E+3	-	4.0E+4	-	8.8E+4	-	9.0E+4	-

Notes:

[1] Normal operating doses assumed a capacity factor of 100% for gammas and 80% for neutrons.

[2] LOCA doses assumed 50% of the core iodine and 100% of the core noble gas was initially released in the containment. 50% of the core iodine and 1% of the solid fission products were assumed to be uniformly mixed in the reactor H_2O + RWST H_2O. Doses are measured in air. Actual beta doses to equipment may be significantly less due to casings, etc. in which equipment is contained.

[3] Conta'nment doses do not apply the containment lower level (inside area by the generators).

[4] PAB doses do not apply to equipment which may be located in (1) PAB drum storage area, (2) PAB nonregenerative heat exchanger in metering pump room, (3) PAB volume control tank room, (4) PAB aerated drain tank room, (5) PAB primary drain tank room, because these areas are not normally surveyed. There is no electrical equipment required to be qualified in the above areas.

DOCKET NO. 50-245

ATTACHMENT 2

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 1

ENVIRONMENTAL QUALIFICATION OF ELECTRICAL EQUIPMENT

SEPTEMBER, 1980

MILLSTONE UNIT 1

RADIATION SERVICE CONDITIONS

1) General

Per reference 1, NNECO was asked to review the radiation service conditions for the Millstone Unit 1 plant against the Guidelines of ref. 1, Appendix B. Since the procedures developed in Appendix B were based on a PWR containment with a containment spray system designed to enhance iodine removal, the procedures were not directly applicable to BWR type containment systems.

The procedures were also not applicable for equipment outside containment, near filters and near systems containing core cooling water. NNECO therefore performed detailed shielding calculations to ascertain the radiation service conditions under a DBA LOCA and MSLB accident. A brief description is given below on the method used to calculate the radiation levels inside and outside containment.

2) Scenario

Consistent with the analysis performed in NUREG-0588, the LOCA analysis was performed assuming a Design Basis Accident.

The MSLB accident used assumptions consistent with those stated in NUREG-0588.

3) Source Terms and Distribution of Activity

A) LOCA

1) Containment

Consistent with guidance given in NUREG-0588, NNECO assumed an initial release of 50% of the core iodines and 100% of the core noble gas to be uniformly mixed in the MP-1 drywell. To be conservative, no plateout on the containment walls and no removal of iodine from the drywell atmosphere due to containment sprays was assumed. (Note: It should be emphasized that 50% of the core iodine activity has been assumed to be uniformly mixed in the drywell atmosphere (no plateout assumed) because it is conservative and simplifies the modeling by avoiding the necessity of calculating the plateout dose rate. NNECO does not believe that 50% of the core iodines would be available for release from the drywell in a DBA. For off-site dose calculations, to comply with 10CFR100, we assume that 25% of the iodines are available for release from the containment. This is consistent with criteria given in Regulatory Guide 1.3.) Decay of fission products was the only means of removal assumed in the analysis.

Because of the water interface between the MP-1 drywell and the torus air space, very little mixing between the drywell and torus air volume would occur. For this analysis no mixing between drywell and torus has been assumed.

Core activity levels were based on a TID-14844 source term. In order to simulate the band of gamma ray spectra emitted by the nuclides, the source was broken up into a seven energy group gamma source. The mean energies of this source are .4, .8, 1.3, 1.7, 2.2, 2.5 and 3.5 Mev.

2) Outside Containment

a) Primary Coolant Piping

Piping which carries primary coolant outside containment was assumed to contain 50% of the core iodine and 1% of the core solid fission product inventory The activity was diluted by a water inventory of 1.3×10^6 gallons which includes water from three sources: (1) reactor vessel, (2) torus and (3) CST.

b) SGTS

Activity accumulated on the SGTS filters was based on an initial release into the containment atmosphere of 25% of the core iodines and 100% of the core noble gases. This assumption is consistent with the assumptions stated in Regulatory Guide 1.3.

B) Main Steam Break Accident

The source term for a steam line break accident was consistent with the assumptions given in NUREG-0588. Ten percent of the core iodine and noble gas activity (except KR-85 which was 30%) was assumed to be released and uniformly distributed in the room where the break occurred. Consequently, 4.75×10^7 curies of iodine and 7.66×10^7 curies of noble gas would be released in the event.

4) Method of Analysis

A) LOCA

1) Containment

a) Gamma

A modified version of the QAD-P5F (ref. 2) computer code was used to evaluate the gamma dose rates at various locations inside the containment. The airborne source in the drywell was broken up into four major source regions. The drywell was broken up into over 150,000 source points to simulate the volume source.

Credit was taken for the biological shield to reduce the dose rates in the drywell. It was modeled as a hollow cylinder with a wall thickness of 2'-0" and a density of 2.4 gm/cm^3. The biological shield was the only shielding for which credit was taken in the drywell.

Twenty receptor points were placed throughout the drywell and the highest dose rate was used to compute the integrated dose at various times after the accident.

b) Beta

Beta doses inside the containment were obtained using the semi-infinite cloud dose model. This method is appropriate because of the short range of Beta particles in air. The beta doses indicated in Tables 1 and 2 represent the beta dose in air, not the beta dose the equipment will actually receive. Actual beta doses to equipment are expected to be significantly lower due to local shielding considerations (e.g., metal casing around electric motors will attenuate most of the betas).

2) Outside Containment

a) Reactor Building

The dose in the reactor building was determined in the corner room (elev. -26'0") in which the LPCI/containment spray heat exchanger and associated piping is located.

The dose point was located in the vicinity of the LPCI/containment spray pump, which is located about eight feet from the heat exchanger. The dose determined at this point was applied to the entire reactor building since this was determined to be the location of maximum potential dose under accident conditions.

The heat exchanger was simply modeled as a pipe of equivalent cross sectional area as the primary coolant flowing through the heat exchanger. No shielding by secondary water or tubing was taken. This method is conservative and simplifies the modeling.

Dose rates from piping sources were determined using a series of parametric piping curves. The curves were generated using a modified version of the QAD-P5F computer code (ref. 2) and give dose rates at various receptor locations for piping sources of different pipe diameters, pipe lengths and decay times. Self-shielding and buildup in the water source was accounted for in the calculations. The buildup factors and attenuation coefficients used are the same as those described in the Stone and Webster RP8A Shielding Manual (ref. 3).

Dose rate estimates were made by determining distances from receptor locations to the piping sources. Parametric curves were then used to determine dose rates.

The torus was also considered in the calculations. It was modeled as a cylindrical source and attenuation by a 2'-0" thick concrete wall was factored into the calculation. Dose rates as determined from this source were negligible compared to the heat exchanger and associated piping.

The computed integrated dose for various times after the accident is given in Table 1.

b) Standby Gas Treatment System (SGTS)

In the event of an accident, the SGTS will process the leakage from the drywell. As a result, activity will be deposited on the filters. A containment leakrate of 1.5%/day and instantaneous transport of activity was assumed. The iodine isotopes and the daughter products Cs^{138} and Rb^{88} produced by the decay of Xe^{138} and Ke^{88}, respectively, were the isotopes which were considered in the calculation.

The activity on the charcoal filters was conservatively assumed to be contained in a volume of $1.9x10^4$ cm^3. Although some distance separates the particulate filters from the charcoal, for simplicity, the iodines and Cs and Rb were all considered to be contained in the same volume. No self-shielding of the charcoal was assumed. This, in addition to the small volume assumed for the filters would conservatively predict the dose from the activity accumulated on the filter.

The QAD model broke up the source volume into 2,000 source points. Shielding by concrete walls was taken into account where applicable.

There were two areas where the dose from the SGTS filters would affect electrical equipment, they are:

(a) SGTS Room

(b) Switchgear Room

The dose point for the SGTS room was taken on the surface of the filter. For the switchgear room the dose point was assumed to be on the surface of the wall which separates the SGTS room from the switchgear room.

The results of the calculation for various times after the accident are given in Table 1.

3) Control Room

A 10" pipe carrying core spray water runs along the control room wall. The pipe is about 3' from the surface of the concrete wall and the concrete wall is 3'-0" thick. A density of 2.4 gm/cm^3 was assumed for concrete.

The dose receptor point was conservatively taken on the surface of the wall inside the control room.

The calculated integrated doses from this source are given in Table 1.

B) Main Steam Line Break (MSLB)

1) Steam Tunnel

To qualify the Main Steam Isolation Valves (MSIV's) outside containment a main steam line break was postulated to occur in the steam tunnel. The dose at the location of the MSIV's was calculated using conservative models. The activity released into the room was assumed to be uniformly distributed in the steam tunnel volume. Due to the presence of shielding in that area, the effective volume source would be approximately 37,000 $ft.^3$

A semi-infinite cloud gamma dose in air was calculated for a one-hour period after the steam line break. This dose was corrected using a finite cloud correction factor similar to that given in Ref. 5.

A semi-infinite cloud beta dose in air was also calculated for a one-hour period after the steam line break. The beta dose represents a dose in air and not necessarily the dose the equipment will receive due to metal casings, etc. which may surround the equipment.

The integrated beta and gamma dose is given in Table 1.

2) Condenser Bay

The dose due to a postulated MSLB in the area of the Condenser Bay was calculated by simply correcting the dose calculated at the MSIV's for concentration and finite cloud. A finite cloud correction volume of 8.041 x 10^5 ft.^{3} was used to correct the gamma dose in the analysis. The beta dose was obtained by simply rationing the concentrations.

Table 1 lists the results of the calculation from this source.

5) Normal Operating Doses

The normal operating doses were calculated and added to the post LOCA doses. Table 1 lists the normal operating doses used in the analysis.

Doses in the containment are based upon the highest dose rates observed from a special radiation measurement while at power. All neutron doses were adjusted by assuming an 80% capacity factor while gamma doses assumed a 100% capacity factor to take into account crud buildup on piping valves and equipment.

Doses in other areas are based upon the highest dose rates observed from recent surveys or other information. All gamma doses assumed a 100% capacity factor.

6) Summary

Table 1 summarizes the integrated accident and normal operating doses. Table 2 summarizes the total dose to which equipment should be qualified (accident dose + 40-year normal operating dose).

In the reactor building, the normal operating dose was taken in many areas. The highest normal operating dose, even though it does not correspond to the same location where the accident dose was computed, was added to the accident dose result. This was done in this way to avoid having to calculate the accident dose in a large number of locations in the reactor building.

REFERENCES

1) Letter from D. L. Zieman to W. G. Counsil, February 15, 1980.

2) E. A. Warman, D. R. Rogers, and B. A. Lindsey, AKERN Aerojet Point Kernel Integration Calculation System, RSIC Computer Code Collection CCC-190, Oak Ridge National Laboratory, Oak Ridge, Tennessee, 1972.

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

4) Millstone Nuclear Power Station Unit 1, Final Safety Analysis Report.

5) K. G. Murphy and K. M. Campe, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19," Presented at the 13th AEC Air Cleaning Conference.

Table 1

Millstone Point Unit 1

Radiation Service Conditions

Receptor Point Location	40 Year[1] Dose (Rads)	Post-Accident Doses[2] (Rads) 1 Hour γ(Gamma)	1 Hour B(Beta)	24 Hours γ	24 Hours B	720 Hours γ	720 Hours B	6 Months γ	6 Months B
Drywell	3.8E+6	7.1E+6	5.1E+7	3.9E+7	3.7E+8	8.8E+7	1.3E+9	9.0E+7	1.5E+9
Reactor Building									
1) Generic[3]	1.3E+6	3.1E+4	-	2.1E+5	-	5.3E+5	-	7.9E+5	-
2) Steam[4] Tunnel	1.1E+6	1.8E+6	2.5E+7	-	-	-	-	-	-
3) TIP Room	7.0E+5	1.6E+5	-	9.2E+5	-	2.1E+6	-	2.4E+6	-
Turbine Building									
1) 2 M Level	8.8E+2	-	-	-	-	-	-	-	-
2) Switchgear Area	3.5E+5	8.0E+2	-	4.3E+3	-	9.9E+3	-	1.0E+4	-
3) SGTS Room	7.0E+5	8.9E+5	-	4.0E+7	-	2.8E+9	-	3.7E+9	-
4) Turbine Ground Floor	2.8E+4	-	-	-	-	-	-	-	-
5) TBCCW, TBSCCW, Machine Shop, Boiler Room	1.1E+4	-	-	-	-	-	-	-	-
6) Condenser[6] Bay, 14.6 Elev.	3.5E+5	2.9E+5	1.4E+6	1.6E+6	9.8E+6	3.6E+6	3.2E+7	3.7E+6	3.6E+7
7) Condenser[6] Bay Floor	1.8E+6	2.9E+5	1.4E+6	1.6E+6	9.8E+6	3.6E+6	3.2E+7	3.7E+6	3.6E+7
Control Room	3.5E+1	<1	-	<1	-	<1	-	1.0E+0	-
Intake Structure & Gas Turbine Bldg.	1.8E+2	-	-	-	-	-	-	-	-

Table 1 Notes:

1. Normal operating doses assumed a capacity factor of 100% for gamma and 80% for neutrons.

2. Accident doses based on a DBA LOCA and assumed 50% of the core iodine and 100% of the core noble gases were intially released and uniformly mixed in the drywell. 50% of the core iodines and 1% of the solid fission products were assumed to be uniformly mixed in the available cooling water.

3. Generic LOCA dose in the reactor building was taken in the corner room in the general vicinity of the LPCI pump motor. The normal operating dose was taken at the ground floor. The generic reactor building dose listed in the table above cannot be applied to the (a) cleanup filter sludge tank and cleanup pump room (second floor), (b) cleanup filter tank, cleanup demineralizer tanks and fuel pool cleanup demineralizer rooms. There is no electrical equipment required to be qualified in the above areas.

4. Accident doses based on steamline break in pipe tunnel. 10% core activity of noble gas and halogens assumed to be released during break.

5. These doses do not apply to equipment in contact with the turbine or any equipment in the general vicinity of the condensate demineralizer regeneration room and SJAE room of the Condenser Bay 14.6' elevation. There is no electrical equipment required to be qualified in these areas.

6. Accident doses based on steamline break in condenser bay, 10%, core activity of noble gas and halogens assumed to be released during break.

Table 2

Millstone Point Unit 1

Radiation Service Conditions

Receptor Point Location	Total Dose[1] (Rads)							
	1 Hour		24 Hours		720 Hours		6 Months	
	γ+n	B(Beta)	γ+n	B	γ+n	B	γ+n	B
Drywell	1.1E+7	5.1E+7	4.3E+7	3.7E+8	9.2E+7	1.3E+9	9.4E+7	1.5E+9
Reactor Building								
1) Generic[2]	1.3E+6	-	1.5E+6	-	1.8E+6	-	2.1E+6	-
2) Steam Tunnel	2.9E+6	2.5E+7	-	-	-	-	-	-
3) TIP Room	8.6E+5	-	1.6E+6	-	2.8E+6	-	3.1E+6	-
Turbine Building[3]								
1) 2 M Level	8.8E+2	-	8.8E+2	-	8.8E+2	-	8.8E+2	-
2) Switchgear Area	3.5E+5	-	3.5E+5	-	3.6E+5	-	3.6E+5	-
3) SGTS Room	1.6E+5	-	4.1E+7	-	2.8E+9	-	3.7E+9	-
4) Turbine Ground Floor	2.8E+4	-	2.8E+4	-	2.8E+4	-	2.8E+4	-
5) TBCCW, TBSCCW, Machine Shop, Boiler Room	1.1E+4	-	1.1E+4	-	1.1E+4	-	1.1E+4	-
6) Condenser Bay, 14.6 Elev.	6.4E+5	1.4E+6	2.0E+6	9.8E+6	4.0E+6	3.2E+7	4.1E+6	3.6E+7
7) Condenser Bay Floor	2.1E+6	1.4E+6	3.4E+6	9.8E+6	5.4E+6	3.2E+7	5.5E+6	3.6E+7
Control Room	3.5E+1	-	3.5E+1	-	3.5E+1	-	3.6E+1	-
Intake Structure & Gas Turbine Bldg.	1.8E+2	-	1.8E+2	-	1.8E+2	-	1.8E+2	-

Table 2 Notes:

1. Total dose includes normal operating doses and post accident doses as given in Table 1.

2. Generic doses cannot be applied to the
(a) cleanup filter sludge tank and cleanup pump room (second floor), (b) cleanup filter tank, cleanup demineralizer tanks and fuel pool cleanup demineralizer room (third floor) and fuel pool cleanup demineralizer room (third floor). There is no electrical equipment to be qualified in the above areas.

3. These doses do not apply to equipment in contact with the turbine or any equipment in the general vicinity of the condensate demineralizer regeneration room and SJAE room of the Condenser Bay (14.6' elevation).

DOCKET NO. 50-245

ATTACHMENT 3

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 1

ENVIRONMENTAL QUALIFICATION OF ELECTRICAL EQUIPMENT

SEPTEMBER, 1980

SCREENHOUSE (ITEM 1)

The screenhouse has two roof mounted exhaust fans, HVR-8 and HVR-9, each with a rated capacity of 20,000 CFM. Outside air intake is provided through a louvered penthouse on the roof of the screenhouse (NUSCo Dwg. 25202 - 24001).

Additional ventilation can be provided if necessary by opening a single 3' x 7' door through the east wall, a double 3' x 7' door through the north wall and eleven various sized hatch covers in the roof (NUSCo Dwg. 25202 - 54013).

GAS TURBINE BUILDING (ITEM 2)

The Gas Turbine Building has two 2500 CFM motor operated roof exhaust fans which are manually controlled and normally off and the accessory package exhaust duct, which is an opening approximately 7' x 10' in the Gas Turbine Building roof with a louvered penthouse over it. Outside air intake is provided by four 10' x 5' louvered openings, two through the west wall, one through the east wall and one through the south wall.

Additional ventilation can be provided by opening a 14' x 22' overhead door through the north wall and a 14' x 17' , 6" overhead door through the east wall (NUSCo Dwg. 25202 - 27007).

Power for the two roof fans is supplied from 208/120 volt lighting panel LP-5 (NUSCo Dwg. 25202 - 35002) which is fed from motor control center F-4 (2A-4) (NUSCo Dwg. 25202 -30005) located in the Turbine Building at 14'6" elevation.

DIESEL GENERATOR ROOM (ITEM 3)

The Diesel Generator room is supplied, through ductwork, 2000 CFM of air from the Turbine building supply fans HVS-2A and HVS-2B, which are located in the Heating and Ventilation (H&V) equipment room at elevation 34' 6" in the Turbine building (NUSCo Dwg. 25202 -24001, 24005). Two thousand CFM of air is also exhausted from the Diesel Generator room by roof exhaust fan HVR-7, which is located on the roof of the House Heating Boiler room adjacent to the Diesel Generator room and connected to the Diesel Generator room by a short length of duct.

Located within the Diesel Generator room are Air Handling Units HVH-1 and HVH-2, which are cooled by the Secondary Closed Cooling Water System. These units recirculate and cool 23, 700 CFM of air each.

Additionally, there is a 3' 6" x 7' 0" door through both the north and south walls of the Diesel Generator room.

Power supply for the Turbine building supply fan HVS-2A is from 480 Volt Motor Control Center E-6(2-6), while Fan HVS-2B is supplied from 480 volt MCC F-6 (2A-6), both of which are located in the Heating and ventilation equipment room (NUSCo Dwg. 25202 - 30007). Roof exhaust fan HVR-7 receives its power through Utility panel UP-1E and lighting panel LP-15 (NUSCo Dwg. 25202 - 35002) from 480 volt MCC D3 (1A-3), located in the machine shop (NUSCo Dwg. 25202 - 30007). Air Handling Units HVH-1 and HVH-2 are both supplied from MCC EF7 (22A-1), located in the Diesel Generator room (NUSCO Dwg. 25202 - 30005). Power for the Turbine building secondary closed cooling water pumps is supplied from 480 volt BUS 1E (2) and BUS 12F (2A) (NUSCo Dwg. 25202 - 30003).

BATTERY ROOMS (ITEM 4)

Battery rooms 1 and 1A are supplied a total of 3000 CFM of air through Turbine building ductwork from Turbine building Supply fans HVS-2A and HVS-2B (NUSCo Dwg. 25202 - 24001, 24004). Air from battery room 1A is exhausted through a 48" x 20" transfer grill (with fire damper) in the east wall of battery room 1A into battery room 1. Air is exhausted from battery room 1 by roof exhaust fan HVE-6, rated for 3000 CFM, which is located on the roof of battery room 1. This fan is controlled by a timer set to allow operation 10 minutes out of every hour.

Additionally, there is a 36" x 12" transfer grill (with fire damper) through the south wall of battery room 1A, a 3' 6" x 7' door through the north wall of battery room 1, a 3' 6" x 7' door through the common wall between battery rooms and a double 3' 6" x 7' door through the south wall of battery room 1A.

Power supply for Turbine building supply fans HVS-2A and HVS-2B are described in Diesel Generator room ventilation description (ITEM 3). Power supply for battery room exhaust fan HVE-6 is from Motor Control Center F5 (2A-5), located on elevation 34' 6" of the Turbine building in the 4160 volt switchgear area.

CONTROL ROOM (ITEM 5)

The main control room ventillation is comprised of HVH-8, which recirculates 17,500 CFM of control room air and takes in 2600 CFM of outside air for make-up, transfer fan HVT-11, which exhausts 2500 CFM of control room air to the Heating and Ventillating equipment room to provide cooling there and control room toilet fan HVT-10, which exhausts 100 CFM of control room air to the outside (see NUSCo Dwg. 25202 - 24001).

Cooling for the control room ventillation system is provided by water cooled condensing units HVAC-1 and HVAC-2 (NUSCo Dwg. 25202 - 24001).

Power supplies for the ventillation and cooling system equipment are as follows:

HVH-8 Motor Control Center (MCC) F-6 (2A-6) located in the H&V equipment room, elevation 54' 6" in the Turbine building (NUSCo Dwg. 25202 - 30007)

HVT-10 Utility Panel UP-1P, Ckt. 19 (NUSCo Dwg. 25202 - 35002) which is fed from 480V MCC F5 (2A-5)

HVT-11 MCC F-6 (2A-6) same as HVH-8

HVAC-1 MCC E-6 (2A-6) located in the H&V equipment room

HVAC-2 MCC F-6 (2A-6) same as HVH-8