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November 5, 1984

Docket No. 50-320

Mr. F. R. Standerfer, Director
Three Mile Island Unit 2
GPU Nuclear Corporation
P.O. Box 480
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Dear Mr. Standerfer:

The purpose of this letter is to clarify the staff's intent in issuing the July 17, 1984 Exemption from the requirements of 10 CFR 50, Appendix A, Criteria 2, 50 and 51. This intent was previously discussed with GPU licensing personnel prior to the issuance of the above document.

Because of the unique status of TMI-2, the staff did not believe that the requirements of Criteria 2, 50 and 51 were applicable in all instances and therefore issued the Exemption. However, the staff does believe that requirements similar to Criteria 2, 50 and 51 may be applicable for some structural and component designs at TMI-2. The staff's action merely allowed for a case-by-case application of natural phenomenon design criteria and should not be interpreted as a permanent deletion of associated design requirements.

The staff's analysis to support the Exemption was bounded by estimated source terms, release pathway area and air flow parameters as discussed in the Enclosure. You should review the staff's analysis and provide the staff with a safety evaluation if a postulated failure of present or future penetrations would exceed the NRC's offsite dose consequence estimates. All documents forwarded to the NRC for review or approval that discuss penetration modifications or containment integrity should also address natural phenomenon effects if applicable.

You should note that the staff's analysis primarily considered radiological releases that were filtered (via the auxiliary and fuel handling buildings) prior to release to the environment. The only unfiltered release pathway considered by the staff was penetration 401, which for the scenarios considered, would release a maximum of 20% of its activity directly to the environment.



| OFFICE | TMIPO:NRR | SL:TMIPO:NRR | PD:TMIPO:NRR | | | |
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| SURNAME | TCPoindexter:ba | RAWeller | BJSnyder | | | |
| DATE | 10/31/84 | 10/31/84 | 11/1/84 | | | |

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Therefore, notwithstanding the July 17, 1984 Exemption to Criteria 2, 50 and 51, the staff will also apply appropriate natural phenomenon design criteria on a case-by-case basis to procedures and design changes reviewed by the NRC in accordance with Section 6.8.2 of the Proposed Technical Specifications for penetrations and structures.

Sincerely,

Original signed by
S. J. Snyder

Bernard J. Snyder, Program Director
Three Mile Island Program Office
Office of Nuclear Reactor Regulation

Enclosure:
As stated

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# TMI PROGRAM OFFICE FAILURE ANALYSIS FOR PENETRATIONS MODIFIED DURING THE RECOVERY PERIOD IN SUPPORT OF 10 CFR 50, APPENDIX A, CRITERIA 2, 50, AND 51 EXEMPTIONS

## INTRODUCTION AND ASSUMPTIONS

Calculations were performed to estimate the offsite dose consequences of various accident scenarios involving breach of non-seismic containment penetrations. The scenarios were selected to be representative of the types and conditions which could occur at TMI-2 during defueling activities. The scenarios were chosen to be at the severe end of the spectrum, i.e., minor fires and cracks in the penetrations were not considered.

A limited number of representative isotopes and critical organs were used to simplify calculations. This simplification will set limiting conditions and account for greater than 90% of the dose. A more complete source term which will account for greater than 98% of the offsite dose is being sent to RAB for dose assessment. Dose conversion factors are from Regulatory Guide (RG) 1.109 except for transuranics. Since RG 1.109 does not list values for Pu and Am, NUREG/CR-1972 was used for these isotopes. Previously published (by NRC) values for short term (accident) atmospheric dispersion were used.

In general the results are ratioable, i.e., one may double the fraction assumed to go airborne and it will double the dose. There are exceptions to this. Lengthening the duration of the events beyond two hours will not increase exposures in direct proportion due to meteorological sector averaging beyond two hours. Increasing the reactor coolant system leak will not increase doses in a linear fashion in that a stream of water does not produce droplets as efficiently as a spraying leak. The assumptions used regarding release fractions are conservative, probably by more than one order of magnitude. A list of references is provided following the last scenario. Additional release fraction models and experimental data from Battelle-Pacific Northwest Laboratories was utilized.

## DOMINANT FACTORS

Several factors dominate the offsite dose consequences at the following accident scenarios. Compared to scenarios for an operating reactor the isotopic mix is smaller and different. In an operating reactor, volatiles (iodine) and noble gases are the primary dose contributors. They are essentially absent at TMI-2. The particulates which dominate dose calculations at TMI-2 have a tendency to settle which iodines and noble gases don't. They are essentially absent at TMI-2. In an operating plant one can usually generate peak containment pressures of 50-60 psig to provide a driving force to propel the isotopes out of containment. At TMI-2 this force is largely absent. HEPA filters are effective at removing particulates, even when assumed to operate at one thirtieth their design efficiency. One must assume much lower decontamination factors for iodine and one for noble gases.

Plutonium 239, and other transuranics become the limiting isotopes due to the fact that their dose conversion factors are orders of magnitude higher than the fission products considered. The actual curie releases are somewhat smaller. In operating reactors plutonium contributes relatively little offsite dose due to physical characteristics (i.e., solid) and the relative abundance of other isotopes. The whole body doses listed for Pu-239 and transuranics are actually organ doses which are "equivalent whole body doses."

The dose consequences only consider the initial releases. The scenarios considered would cause continual releases; however, after the first hour or two (which were factored in) the rates drop by orders of magnitude. The licensee has several methods available to terminate the release, during the initial period it was assumed that these actions are not taken. These actions include plugging the penetration and/or starting a train of RB purge to put the building under negative pressure and provide a filtered releases path. Most of the scenarios including the limiting scenario are self-extinguishing. Any long-term releases were considered improbable and small, and therefore were neglected.

SCENARIO I - FIRE AND PENETRATION FAILURE

In this scenario a seismic event fails various penetrations and also knocks over a temporary lighting device which starts a fire in the reactor building. Several fires were considered including contaminated surfaces (i.e., cable trays), a fire in the "D" ring supported by reactor coolant pump lubricating oil and a fire in the radioactive materials storage area. The fire in the radioactive materials storage area produced the highest dose consequences.

The stroage area is assumed to contain a variety of materials including tools, equipment (i.e., TV cameras, hoses, etc.). The materials would be enclosed in polyethylene (PE) or polyvinyl chloride (PVC) wraps or bags. A total of 8 Ci of Cs-137 and 4 Ci of Sr-90 is assumed in the storage area. This number is somewhat conservative in that personnel exposure considerations would preclude accumulation of that much activity in one location. The isotopic distribution is representative of the average expected over the defueling. Conservatism in the total curie content covers the expected shift from low initial Sr fractions to perhaps more than 50% as defueling activities progress.

The PE, PVC, rags, paper and other wiping and wrapping materials are assumed to contain 10% of the total activity. The release fraction due to the fire is 5 E-2 for these materials. The remainder of the activity is on tools and components. The release fraction for these materials is 1 E-2.

The total airborne activity generated in the reactor building by the fire is:

(0.8 Ci x .05) + (7.2 Ci x .01) = .112 Ci Cs-137
(0.4 Ci x .05) + (3.6 Ci x .01) = .056 Ci Sr-90

During the event the fire creates a 2 PSI RB overpressure (16.7 psia), contaminated air escapes through open penetrations until equilibrium is reached with outside air. This release represents 12% (2 ÷ 16.7) of the containment air or 240,000 cubic feet.

If a penetration in the vicinity of the fire (i.e., 561 or 565) failed, essentially all of the airborne activity could escape in the 240,000 cubic feet.

If the penetration, which fails is remote from the fire location, the maximum fraction of airborne activity leaving the RB would approximate 12% due to mixing by the RB recirculation system.

Assuming that penetration 561, 565, or both fail, if the auxiliary building ventilation system is operating it will remove 99% of the activity (accident assumptions for 99.97% efficient HEPA filters). If the ventilation system is not operating, 90% of the activity will fall out in the auxiliary building (a large dead air volume) and 10% will exfiltrate from the building. The worst case is with the ventilation inoperable and results in the release of 11 mCi of Cs-137 and 6 mCi of Sr-90.

If penetration 401 fails (remote from radioactive material storage area) 12% of the airborne activity (13 mCi of Cs-137 and 7 mCi of Sr-90) could be transferred to the basement of the service building. No fallout was assumed since this would not be a large dead air space. The 13 mCi of Cs and 7 mCi of Sr are assumed to be released directly to the environment. This represents the limiting case since a combination of failed penetrations would lower the activity escaping through penetration 401.

Dose calculations:

The activity is all released within a 2 hour period ($X/Q = 6.8_3E\text{-}4$ sec/$m^3$). An adult at the exclusion boundary breathes at 1.2 $m^3$/hr.

For Cs-137

(1 hr/3600 sec) (1.2 $m^3$/hr) 6.8 E-4 sec/$m^3$) (0.0065 Ci/hr) (2 hr)
(1 E12 pCi/Ci) (5.35 E-5 mrem/pCi) = .16 mrem whole body dose

(1 hr/3600 sec) (1.2 $m^3$/hr) (6.8 E-4 sec/$m^3$) (0.0065 Ci/hr) (2 hr)
(1 E12 pCi/Ci) (5.98 E-5 mrem/pCi) = .13 mrem bone dose

For Sr-90

(1 hr/3600 sec) (1.2 $m^3$/hr) (6.8 E-4 sec/$m^3$) (0.0035 Ci/hr) (2 hr)
(1 E12 pCi/Ci) (7.62 E-4 mrem/pCi) = 1.21 mrem whole body dose

(1 hr/3600 sec) (1.2 $m^3$/hr) (6.8 E-4 sec/$m^3$) (0.0035 Ci/hr) (2 hr)
(1 E12 pCi/Ci) (.124 E-2 mrem/pCi) = 19.7 mrem bone dose

In Scenario I all the airborne activity from the reactor building escaped to the auxiliary building. Therefore, change in the number and size of the penetrations can't increase the potential release.

SCENARIO II - LEAKS AND SPILLS

In this scenario a seismic event causes the failure of penetration(s) and a leak in the reactor coolant system (RCS) or an RCS cleanup system. At the time of the leak the RCS activity concentrations are assumed to be elevated due to defueling activities. The following concentrations are assumed 15 uCi/ml Cs-137, 7.5 uCi/ml Sr-90, 1 uCi/ml Ce-144, and 5 E-8 uCi/ml Pu-239.

In the leak of the processing system the leak is assumed to be at the pump outlet prior to demineralizers. The leak rate is 25 gpm and the system is turned off (isolating the leak) after 1 hr. The fraction becoming airborne due to spraying, splashing, and free fall is $10^{-3}$. The resulting airborne activity is (25 gal/min) (60 min) (3785 ml/gal) (.001) (activity conc Ci/ml). The results are .085 Ci Cs-137, .042 Ci Sr-90, .005 Ci Ce-144, and 2.8 E-10 Ci Pu-239.

If the leak occurs in the RCS it would be unpressurized (other than static head). The leak is assumed to continue until the water drains to the level of the reactor vessel nozzles the total volume is assumed to be 20,000 gallons. Due to the large volume and lack of pressurization the fraction becoming airborne is $10^{-4}$. For this case .11 Ci Cs-137, .056 Ci Sr-90, .008 Ci Ce-144 and 3.7 E-10 Ci of Pu-239 become airborne. Assuming simultaneous failure of several penetrations air could be drawn into the reactor building through penetration 401 and out penetrations 561 and 565 by the auxiliary building (AB) and fuel handling building (FHB) exhaust fans. If the highly contaminated air remained below the RB 347 ft. elevation, total airflow through the penetrations would not become a limiting factor. The air would pass through HEPA filters (accident DF of 100) and 1% of the activity would be discharged through the vent stack.

A single failure of penetration 401 coupled simultaneously with the passage of a 1 psig low pressure front could result in a direct release pathway. This would result in the release of 6.8% of the containment air prior to reaching pressure equilibrium. Due to the location of penetration 401 in the lower portion of the RB it is assumed that 20% of the activity is entrained in the air (6.8% RB volume) which is released.

The resultant offsite doses (assuming short term release) are given by (1 hr/3600 sec) 1.2 m-3/hr) (6.8 E-4 sec/m-3) (1 E12 pCi/Ci) (Ci released) (dose conversion factor mrem/pCi) = mrem.

(2.27 E5) (Ci) (DCF) = mrem

For Cs-137

(2.27 E5) (0.22 Ci) (5.35 E-5 mrem/pCi) = 0.27 mrem whole body dose
(2.27 E5) (0.22 Ci) (5.98 E-5) = .30 mrem bone dose

For Sr-90

(2.27 E5) (.011 Ci) (7.62 E-4 mrem/pCi) = 1.9 mrem whole body
(2.27 E5) (.011 Ci) (1.24 E-2 mrem/pCi) = 31 mrem bone dose

For Ce-144

(2.27 E5) (0.0016) (2.3 E-5) = .01 mrem whole body
(2.27 E5) (0.0016) (4.29 E-4) = .16 bone dose

For Pu-239

(2.27 E5) (7.4 E-11) (.514 mrem/pCi) = 9 E-6 mrem whole body equivalent
(2.27 E5) (7.4 E-11) (9.139 mrem/pCi) = $1.5 \times 10^{-4}$ mrem bone surface dose

The sum of other transuranics which could potentially cause significant dose contributions is 84% of the Pu-239 dose. This is based upon their ratios in the fuel and the ratios of dose conversion factors. These isotopes are Pu-238, Pu-240, Pu-241 and Am-241.

In Scenario II the assumptions for penetration 401 assumed 20% of the reactor building airborne escaped to the environment. The worst case for multiple failures on any size in the auxiliary and fuel handling buildings would be 10%. Therefore, they cannot become limiting in this scenario.

SCENARIO III - DROPS OF CANISTERS

The licensee's RCS cleanup and defueling methodologies have not been finalized. The RCS cleanup system will probably be located outside the RB; however, it is included inside since this is a potential alternative. A limit of 800 kg of fuel and rubble is assumed for fuel transfer canisters.

The limiting activity in RCS cleanup canisters is placed at 42,500 Ci Cs-137, 21,000 Ci Sr-90, 100 Ci Ce-144 and Pu-239 5 Ci. Cs and Sr would be limited by water throughout (media depletion); Ce and Pu would be limited by solids accumulation (plugging). In the canister drop $10^{-4}$ is assumed to become airborne resulting in an airborne source in the RB of 4.25 Ci Cs-137, 2.1 Ci Sr-90, 0.01 Ci Ce-144 and .0005 Ci Pu-239.

The canister drop would occur on the upper elevations, 10% of the activity is assumed to pass out penetration 401 (in the basement) due to storm front passage. This results in a release to the environment of .425 Ci Cs-137, .21 Ci Sr-90, .001 Ci Ce-144 and 5 E-5 Ci Pu-239.

Resulting offsite doses are:

| | Whole Body (mrem) | Bone (Bone Surface) (mrem) |
|---|---|---|
| Cs-137 | 5.10 | 5.70 |
| Sr-90 | 36.02 | 591.00 |
| Ce-144 | .01 | 0.01 |
| Pu-239 | 58.50 equiv. | 103.00 |
| Other TRU | 49.00 equiv. | 87.00 |
| | 149.00 equivalent | 787.00 equivalent |

With a limit of 800 kg of fuel in a canister, the activity per canister would be 4350 Ci Cs-137, 5650 Ci Sr-90, 1570 Ci Ce-144, and 83 Ci Pu-239. With $10^{-4}$ airborne in the fuel canister drop accident the airborne source becomes .435 Ci Cs-137, .565 Ci Sr-90, .157 Ci Ce-144, and .0083 Ci Pu-239.

This canister drop would also occur in the upper elevations of the RB with 10% of the airborne activity escaping through penetration 401. The release to the environment becomes .0435 Ci Cs-137, .0565 Ci Sr-90, .0157 Ci Ce-144, and .00083 Ci Pu-239.

Resulting offsite doses are:

| | Whole Body (mrem) | Bone (Bone Surface) (mrem) |
|---|---|---|
| Cs-137 | .50 | 0.60 |
| Sr-90 | 10.00 | 160.00 |
| Ce-144 | 0.10 | 0.20 |
| Pu-239 | 97.00 equiv. | 1718.00 |
| Other TRU | 81.00 equiv. | 1443.00 |
| | 189 mrem | 3161 mrem (312 Rem) |

In Scenario III 10% of the reactor building airborne passed out penetration 401 this is greater than the upper limit for a series of large penetration failures on the auxiliary and fuel handling building.

SCENARIO IV - PYROPHORIC EVENT AND PENETRATION FAILURE

In this scenario the seismic event simultaneously fails penetration 401 and many of the incore instrument guide tubes. Core rubble spills out into the reactor vessel cavity, the rubble is assumed to contain finely divided zirconium. The moist zirconium undergoes a pyrophoric reaction when exposed to air. The total quantity of rubble is assumed to be 1000 Kg including 100 Kg of finely divided zircalloy (predominantly zirconium). The fuel which is mixed with the zircalloy is already in an oxide form and will not react. However, due to the zirconium reaction, it is assumed that 10-4 of the fuel will become airborne particulates.

The release to the RB atmosphere is 0.5 Ci Cs-137, 0.65 Ci Sr-90, 0.18 Ci Ce-144 and 0.0091 Ci Pu-239. The following release mechanism is assumed; half is captured by the water from the leak and deposition due to the high density of the particles and that 20% of that remaining airborne escapes through the open penetration.

Resulting offsite doses are:

| | Whole Body (mrem) | Bone (Bone Surface) (mrem) |
|---|---|---|
| Cs-137 | 0.6 | 0.7 |
| Sr-90 | 11 | 186 |
| Ce-144 | 0.1 | 0.2 |
| Pu-239 | 113 equiv. | 2002 equiv. |
| Other TRU | 95 equiv. | 1682 equiv. |
| | 220 mrem | 3871 mrem (3.9 Rem) |

In Scenario IV 20% of the activity building airborne passed out penetration 401 which exceeds the limiting case for multiple large breaks in the auxiliary and fuel handling building penetrations.

## CONCLUSIONS AND RECOMMENDATIONS

The results of the scenarios show the worst case offsite dose commitments exceeding, but within a factor of 10 of 10 CFR 20 limits. The dose to the most exposed organ is well within (i.e., less than 20%) the exposure guidelines for whole body dose in 10 CFR 100 (using ICRP 30 methodology). The equivalent whole body dose is on the order of 10 CFR 20 annual limits. The dose commitments are all less than those for which evacuation would be recommended per NUREG-0654.

In general, if the auxiliary and fuel handling building ventilation systems are operating, the activity released would be less than 1%. If the ventilation is down the lack of driving head and building plateout will each limit releases to 10%. In aggregate they would limit releases to <5%.

This analysis is valid for up to 20 $ft^2$ of penetrations in the auxiliary or fuel handling buildings.

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