State University of New York at Buffalo

NUCLEAR SCIENCE AND TECHNOLOGY FACILITY

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FILE RE:

Docket 50-57

License R-77

Cecil O. Thomas, Chief
Standardization and Special Projects Branch
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Thomas:

In support of renewal of State University of Buffalo/Nuclear Science and Technology Facility operating license R-77, enclosed please find two documents which answer the final questions which are outstanding in our response to your letter dated January 21, 1983. As we have previously discussed with Bob Carter, the differential pressure across the NSTF heat exchanger is not always such that the secondary pressure is higher than the primary pressure. We have therefore analyzed the consequences of a postulated leak, and have demonstrated that airborne and water-borne concentrates of radioactive material do not exceed applicable limits. I believe the updated LOCA analysis is self-explanatory.

If you have any questions, please let me know.

Sincerely,

Louis G. Henry

Louis G. Henry
Acting Director

amf
encs.
cc: Charles C. Thomas

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LOSS OF COOLANT ACCIDENT

As per your request, we have reevaluated the loss of coolant accident (LOCA) analysis presented in our SAR (Reference 1 and 2). The results of this reevaluation are presented in the following.

LOCA mechanisms have been considered. The only credible accident would be one involving a beam port, as discussed below. Although such an accident would not result in totally uncovering the core, this has been assumed in our accident analysis. The time required to uncover the core, assuming shearing off of an unplugged, unsealed 6-inch beam tube is about 10.2 minutes. The total volume lost is about 13,000 gallons. The damaged beam port can be sealed with a cover plate when the rate of flow out of the tube decreases sufficiently to permit access to it. Once the beam port is sealed, the tank can be reflooded with the manual emergency pool filling system. It should be noted that the emergency pool filling system is a spray system directed onto the top of the core, thereby providing efficient cooling of the partially exposed core until the refilling can be accomplished.

A. Accident Mechanisms

The current reactor configuration has no reactor tank penetrations below the bottom of the grid plate. Such penetrations that are described in the original SAR (Reference 1) were welded shut during the 1978 repair program that re-routed the primary cooling piping. During the repair, the tank wall thickness was monitored. There was indication of possible decrease in tank wall thickness in sections covering the step in the biological shield, which is well above the top of the reactor core. New plate was welded in place to correct this situation.

1. Beam Tube Accidents

There are five beam tubes. These are all 6-inch diameter tubes. Four of these are located behind the hot cell on the neutron deck, and are not readily available for insertion of samples or extraction of a beam for use outside the biological shield, but are accessible for repair. Because of the limited accessibility, these tubes are plugged, sealed (the outside covers are bolted in place), and are water flooded. The only credible leaks involving these tubes would be minor leakage around the coverplates. Such leakage would be detected by visual observation during hot cell operations (daily) and, if it is pool water leakage, by the daily tank water-level measurements.

The fifth beam tube conceivably could be sheared off in an accident involving a heavy cask. Such an accident, although highly unlikely, could result in loss of pool water down to the bottom of the beam tube, which is several inches above the top of the reactor grid plate if the beam tube is unplugged and unsealed.

2. Coolant Piping Accidents

The core outlet is an 8-inch diameter pipe that exits the reactor tank and biological shield through the former 12-inch square beam port. The pipe is welded to a coverplate, which is bolted to the outside of the biological shield. An isolation valve is located in the pipe, approximately 24 inches from the cover plate.

The return line is a 6-inch diameter pipe that returns to the reactor tank through a former 6-inch diameter beam tube. The pipe is welded to a cover plate that is bolted to the outside of the biological shield. An isolation valve is located in the pipe about 24 inches in from the cover plate.

An accident involving a heavy cask inside the tank would force the outlet pipe against the opening (approximately 12-inches square) in the tank. This would probably result in crushing the pipe at the point of contact, but would probably not shear the pipe. The accident might distort coverplate, resulting in a slow leak that could be repaired, or by activating the emergency pool filling system. An accident that would result in rupture of the pipe between the biological shield and the valve is unlikely because of the concrete shielding surrounding the pipe and the valve.

An accident involving a heavy cask and the return line could result in guillotine type action between the pipe and the opening in the tank, because there is only a small difference in the diameters of the pipe and the opening. Closure of the isolation valve would prevent draining of the pool. Distortion of the coverplate in the accident would have the same results as described above for the outlet line. An accident that would result in rupture of the pipe between the coverplate and isolation valve is unlikely because of the concrete shielding, but if it did occur, would be equivalent to an accident involving a 6-inch beam tube.

B. Accident Analysis

Two accident analyses are discussed in the following sections. The first is based on comparison to a study of a LOCA in the National Bureau of Standards (NBS) reactor (Reference 2). The second is based on experimental studies at LITR and is identical with that discussed in our SAR (References 1 and 2). Both accidents assume complete uncovering of the reactor core, which will not occur.

1. Maximum Fuel Temperature Prediction

The method used in Reference 3 to predict the maximum fuel temperature in NBS reactor fuel uses a two-dimensional finite-element computer code. The code "homogenizes" the core by combining the structure, coolant channels, and fuel into two composite materials according to their respective volume

fractions. The materials used in the model are the fuel, the surrounding box material, and the air. In the NBS model, the fuel and surrounding box material are assumed to be aluminum. Thus, the effect of the uranium on the thermal conductivity is neglected. In our comparison, we have homogenized the UO_2, the zircalloy clad, and the zircalloy box material into one material.

The effective thermal conductivity of the homogenized fuel is based on the assumption that the dispersed phase (air) can be represented by lumped parallel or series resistances. For example, in the vertical direction (parallel to the fuel plates or pins), the effective thermal conductivity is

$$\lambda_E = \lambda_1 e_1 + \lambda_2 e_2$$

where λ_1 = conductivity for the gaps (air spaces)

λ_2 = conductivity for fuel composite

e_1 = void (air) fraction

e_2 = fuel fraction

In the horizontal direction where the void spaces and the fuel plates (pins) are in series, it is

$$\lambda_E = \lambda_1 \lambda_2 / (e_2 \lambda_1 + e_2 \lambda_2)$$

We have calculated the effective thermal conductivities for the NBS case and for our case. The results of these are given below with the input data.

	e_1	e_2	λ_1^a	λ_2^a	λ_E^a Vertical	λ_E^a Horizontal
NBS	0.709	0.291	5.75×10^{-5}	0.54	0.157	8.11×10^{-5}
NSTF	0.417	0.579	5.75×10^{-5}	0.0517	0.030	1.38×10^{-4}

a g-cal/cm-s-^{o}C al 100^{o}C

As can be seen from the above, the primary heat loss is in the vertical direction. The vertical direction thermal conductivity for the NBS fuel is a factor about 5 greater than that of the NSTF fuel. If we ignore the horizontal thermal conductivity, which favors the NSTF fuel, then the maximum fuel temperature for the NSTF fuel following extended operation at 2Mw should be equivalent to that resulting from operation of the NBS reactor at 10 Mw. Figure 3 of Reference 3 plots maximum fuel temperatures for several power levels for loss of coolant inside and outside the fuel for various decay times where the decay time is the interval between when the reactor is shut down from fuel power until the water is lost.

For our accident scenario, the time from reactor scram until the core is uncovered is 600 s (10 min). A linear extrapolation of the 10 Mw curve in Figure 3 (Reference 3) to a 600 sec. decay time gives a maximum fuel temperature of about 1400°C (approximately 1670 K). The decay heating rate after 600 s is about 100 watts/pin or 1.6 watts per cm^3 of active pin volume. This heating rate results in an insignificant temperature difference between the centerline of the fuel and the outer surface of the clad. Thus the estimated maximum clad temperature and the maximum fuel temperature is estimated to be 1400°C compared to a clad and fuel melting temperature of about 1815°C and 2760°C, respectively.

The assumptions used in the NBS model are inherent in our use of the results to estimate our maximum fuel temperatures. These assumptions are conservative as applied to NBS fuel and are, likewise, conservative in the case of our fuel. It should be noted that heat sinks such as a grid plate, and ECCS "fog" cooling have been neglected in the model.

2. Experimental

The LOCA presented in our SAR (Reference 1) is based on the results obtained from experiments with the LITR at Oak Ridge National Laboratory. In the LITR experiments, loss of water was used as a shutdown mechanism, and the fuel plate temperatures were measured. At the LITR, the maximum fuel temperature was reached approximately one hour after uncovering the core.

The fuel plate temperature data from the LITR experiments at 1.0 and 1.5 Mw were extrapolated to 2 Mw, and then were adjusted to reflect differences in hydraulic and heat transfer characteristics. The maximum clad surface temperature obtained is 707°C, which is well below the zircalloy melting point of 1815°C. We note that it is also about a factor of 2 lower than the conservative value calculated in Section B.1, above.

C. Dose Rate at Top of Tank

Because of the biological shield, the radiation emanating from the core after a LOCA will be highly collimated. We have estimated the dose rate at the top of the tank based on a total fission product inventory of 3×10^6 Ci with an average gamma energy release or 0.8 Mev per disintegration. The core was treated as a point source located at the midplane of the core and a source to top of tank distance of about 25 feet. The dose-rate directly above the core at the top of the tank would be 2.24×10^4 R/h. This would decrease rapidly due to radioactive decay. Dose rates at location other than above core would be substantially lower.

D. Summary

No credible LOCA would result in uncovering the core reactor. Rupture of a 6-inch beam tube would drain the pool to a point several inches above the top of the grid plate in about ten minutes. Two approaches to estimating the maximum fuel temperature following an accident that uncovers the core in ten minutes yield temperatures well below the melting point of the fuel clad. The more conservative method yields a maximum fuel temperature, following loss of coolant in 1.67 minutes (100 s) of about 1530°C as compared to the 1815°C melting point of zircalloy. Rupture of 8-inch primary coolant line would drain the pool to the bottom of the grid plate in about 5.7 minutes. Thus we conclude that any credible accident, if it could lead to uncovering the core, would not result in maximum fuel temperatures that would result in meltdown of the fuel and subsequent loss of fuel integrity. Because the postulated accidents do not lead to fuel melting, the time to reach maximum temperature is not significant. We note, however, that the time to reach maximum fuel temperature in the LITR experiments was about one hour. As noted in the discussion, a ruptured beam tube can be sealed. Breaks in the primary coolant pipes between the biological shield and the isolation valves are highly unlikely; however, if it should occur, sealing of the pipe with an inflatable mini-ball is feasible, and would allow reflooding of the core.

If a loss of coolant occurs during off-hours, the radiation level in the reactor building would be sufficient to activate the area alarms. These activate alarm lights outside the airlock doors. Security guards make rounds about every two hours, and have been instructed not to enter the containment building if the alarm lights are activated, and to contact specific NSTF personnel. Corrective action would then be taken.

As indicated in Reference 2, as a last resort in case of a loss of coolant, the entire lower floor of the reactor could be flooded.

We note that while a LOCA is not credible, and if one did occur it would not endanger the fuel or the health and safety of the public, a failure of the present automatic feature of the emergency pool filling system would result in overflowing of the reactor tank in about 5 minutes. Assuming this occurs during off-hours, it might go undetected for two hours. This would result in 27,500 gallons of contaminated water on the lower level of the containment building, and extensive damage to equipment. We have, therefore, proposed that the system be converted to a manually-activated system in our revised Technical Specifications.

Reference 1: WNYNRC, Inc., "Safety Analysis Report, Revision II", (September 23, 1963).

Reference 2: NSTF, "Safety Analysis Report Update, Docket 50-57 (1981).

Reference 3: M. T. Bencomo and G. E. Cort, "Parameters Study of the Effects of Loss of Coolant in the National Bureau of Standards Experimental Reactor", Transmitted by letter dated May 11, 1982 from G. E. Cort to Dr. Harold Bernard.

Accident Analysis: Heat Exchanger Leak

Scenario One: Short term (rapid) loss of H_2O from primary into secondary.

Assumptions:

1. Chronology:

   Operator does not respond to low water level annunciator, but responds to pool level scram. An inspection of the pool will show nothing unusual. The operator will close the pool isolation valves. It is assumed that the leak stops five minutes after scram.

2. Leak Rate:

   It is assumed that the leak rate is 60 gallons/minute. This corresponds to one inch of pool water per minute at the top of the tank.

3. Losses:

   100 per cent of gasses; 10 per cent others.

Data:

Volume of secondary system = 12,000 gallons

Make up rate = 750 gallons/hour

Flow rate = 700 GPM

Volume of loss = 12" x 60 gal/in
+ 5 min x 60 gal/min

1020 gallons

Conversions to C.G.S.

1. Water Loss
   1020 gal = 3.86×10^6 ml.

2. Volume of secondary
   12,000 gal = 4.54×10^7 ml

3. Flow rate
   700 GPM = 4.416×10^4 ml/sec

4. Make up rate
   750 gal/hr = 7.886×10^{2} ml/sec

5. Leak rate
   1020/10 gal/min = 3.785×10^{3} ml/sec

Calculations:

Results of calculations are presented in Table I.

The following relationships were used to generate Table I. The pool water concentrations represent a typical assay.

1. Total loss to secondary
   (Leak rate) x (1020 sec) x (conc. in 1^{o}) =(column 3)

2. Max concentration in secondary
   (Total loss)÷ (vol. of secondary) = (column 4)

3. Released concentration
   (Max conc. in secondary) x 1 for gases = (column 5)
   (Max conc. in secondary) x 0.1 for others =(column 5)

4. Release rate
   (Released concentration) x (Makeup rate) = (column 6)

Table II compares the postulated releases to regulatory guidelines. Table II was generated using the following correlations:

1. Column 2 = column 5 Table I

2. Column 3 = (column 2) x 17 min ÷ 5.2597×10^{5} min/year

3. Column 4 = 10CFR20 Appendix B, Table 2, column 1

4. Column 5 = (column 3) ÷ (column 4) x 100%

Conclusions:

None of the airborne concentrations, when averaged over one year, exceed 10 CFR 20 limits. These calculations do not take into account the volumetric expansion of the water due to evaporation and the forming of aerosols, and hence are extremely conservative.

TABLE I

Isotope	Pool H_2O Conc. µc/cc	Total loss to Secondary µc	Max. Conc. in Secondary µc/cc	Released Concentration µc/cc	Release Rate µc/sec
Ar-41	6.1×10^{-4}	2.36×10^{3}	5.23×10^{-5}	5.23×10^{-5}	4.12×10^{-2}
Xe-133	4.1×10^{-5}	1.58×10^{2}	3.51×10^{-6}	3.51×10^{-6}	2.77×10^{-3}
Xe-135	5.8×10^{-5}	2.24×10^{2}	4.97×10^{-6}	4.97×10^{-6}	3.92×10^{-3}
Ag-110m	1.1×10^{-5}	4.25×10^{1}	9.43×10^{-7}	9.43×10^{-8}	7.44×10^{-5}
Ba-133m	4.0×10^{-5}	1.54×10^{2}	3.43×10^{-6}	3.43×10^{-7}	2.70×10^{-4}
Ba-140	1.6×10^{-4}	6.18×10^{2}	1.37×10^{-5}	1.37×10^{-6}	1.08×10^{-2}
Co-60	3.8×10^{-7}	1.47×10^{0}	3.26×10^{-8}	3.26×10^{-9}	2.57×10^{-6}
Cs-138	1.9×10^{-4}	7.34×10^{2}	1.63×10^{-5}	1.63×10^{-6}	1.29×10^{-3}
I-131	5.5×10^{-6}	2.12×10^{1}	4.71×10^{-7}	4.71×10^{-8}	2.71×10^{-5}
I-132	2×10^{5}	7.72×10^{1}	1.71×10^{-6}	1.71×10^{-7}	1.35×10^{-4}
I-134	1.9×10^{-4}	5.41×10^{2}	1.20×10^{-5}	1.20×10^{-6}	9.46×10^{-4}
La-140	6.8×10^{-5}	2.63×10^{2}	5.83×10^{-6}	5.83×10^{-7}	4.60×10^{-4}
Na-24	9.2×10^{-4}	3.55×10^{3}	7.88×10^{-5}	7.88×10^{-6}	6.21×10^{-3}
Np-239	3.4×10^{-5}	1.31×10^{2}	2.91×10^{-5}	2.91×10^{-6}	2.29×10^{-3}
Sb-122	2.4×10^{-5}	8.11×10^{1}	1.80×10^{-6}	1.80×10^{-7}	1.42×10^{-4}
Sb-124	4.0×10^{-6}	1.54×10^{1}	3.43×10^{-7}	3.43×10^{-8}	2.70×10^{-5}
Tc-99m	3.3×10^{-5}	1.27×10^{2}	2.83×10^{-6}	2.83×10^{-7}	2.23×10^{-4}
H-3	2.2×10^{-4}	8.49×10^{2}	1.89×10^{-5}	1.89×10^{-6}	1.49×10^{-3}

TABLE II

Isotope	Released conc. µc/ml	Averaged conc. µc/ml	Limit µc/ml	% Limit %
Ar-41	5.23×10^{-5}	1.69×10^{-9}	4×10^{-8}	4.2
Xe-133	3.51×10^{-6}	1.13×10^{-10}	3×10^{-7}	0.04
Xe-135	4.97×10^{-6}	1.61×10^{-10}	1×10^{-7}	0.2
Ag-110m	9.43×10^{-8}	3.05×10^{-12}	7×10^{-9}	0.04
Ba-133m	3.43×10^{-7}	1.11×10^{-11}	1×10^{-10}	11
Ba-140	1.37×10^{-6}	4.43×10^{-11}	4×10^{-9}	1.1
Co-60	3.26×10^{-9}	1.01×10^{-13}	1×10^{-8}	.001
Cs-138	1.63×10^{-6}	5.27×10^{-11}	3×10^{-8}	0.2
I-131	4.71×10^{-8}	1.52×10^{-12}	1×10^{-10}	1.5
I-132	1.71×10^{-7}	5.53×10^{-12}	1×10^{-8}	0.06
I-134	1.20×10^{-6}	3.88×10^{-11}	6×10^{-9}	0.6
La-140	5.83×10^{-7}	1.88×10^{-11}	5×10^{-9}	0.4
Na-24	7.88×10^{-6}	2.55×10^{-10}	4×10^{-8}	0.6
Np-239	2.91×10^{-6}	9.41×10^{-11}	3×10^{-8}	0.3
Sb-122	1.80×10^{-7}	5.82×10^{-12}	6×10^{-9}	0.1
Sb-124	3.43 x 10-8	1.11×10^{-12}	5×10^{-9}	0.02
Tc-99m	2.83×10^{-7}	9.15×10^{-12}	1×10^{-6}	0.0009
H-3	1.89×10^{-6}	6.11×10^{-11}	2×10^{-7}	0.03

Table III presents calculations of concentrations of radionuclides retained in the cooling tower basin. Eventually these materials would drain into the sanitary sewer. No decay was calculated. When compared to 10 CFR Appendix B, Table II, column 2 limits, none of the materials would require decay in order to release. (Dilution = 100,000 gallons/day.)

TABLE III

Isotope	Total μc in Secondary	Total μc Retained	Concentration μc/ml	10 CFR Appendix B Table II, Column 2
Ag-110m	42.4	38.2	8.41×10^{-7}	3×10^{-5}
Ba-133m	154	139	3.06×10^{-6}	3×10^{-6}
Ba-140	618	556	1.22×10^{-5}	3×10^{-5}
Co-60	1.5	1.4	3.08×10^{-8}	5×10^{-5}
Cs-138	734	661	1.46×10^{-5}	----
I-131	21.2	19.1	4.21×10^{-7}	3×10^{-7}
I-132	77.2	69.5	1.53×10^{-6}	6×10^{-5}
I-134	541	487	1.07×10^{-6}	2×10^{-5}
La-140	263	237	5.22×10^{-6}	2×10^{-5}
Na-24	3552	3197	7.04×10^{-5}	2×10^{-4}
Np-239	1313	1182	2.60×10^{-5}	1×10^{-4}
Sb-122	81.1	73	1.61×10^{-6}	3×10^{-5}
Sb-124	15.4	13.9	3.06×10^{-7}	2×10^{-5}
Tc-99m	127	114	2.51×10^{-6}	6×10^{-3}
H-3	849	764	1.68×10^{-5}	3×10^{-3}
		7552 μc		