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TO: MR V STELLO

FROM: NORTHERN STATES POWER CO
MINNEAPOLIS, MINN.
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INPUT FORM

DESCRIPTION
LTR, RE THEIR 1-22-76 SUBMITTAL "AN ANALYSIS & SAFETY OF SPENT FUEL.....TRANS THE FOLLOWING........

ENCLOSURE
REPORT CONCERNING THE CONSEQUENCES OF A CASK DROP DOWN THE REACTOR BUILDING EQUIPMENT HATCH......

PLANT NAME: L MONTICELLO

FOR ACTION/INFORMATION

ENVIRO 2-20-76 RKB

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NORTHERN STATES POWER COMPANY

MINNEAPOLIS, MINNESOTA 55401

February 13, 1976

Mr Victor Stello, Director
Division of Operating Reactors
U S Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr Stello:

MONTICELLO NUCLEAR GENERATING PLANT
Docket No. 50-263 License No. DPR-22

Off-Site Shipment of Spent Fuel

On January 22, 1976 we transmitted to you a report entitled "An Analysis and Safety Evaluation of Spent Fuel Shipping Cask Handling at the Monticello Nuclear Generating Plant" dated January 13, 1976. Since then we have had several telephone conversations concerning this report with Mr R P Snaider of your staff. Mr Snaider had several questions concerning the consequences of a cask drop down the Reactor Building equipment hatch.

One area of concern was the potential for the cask to impinge upon the suppression pool if the cask should drop in other than a vertical attitude. Consideration was given to this situation in our review of cask handling operations and it was determined not to be a relevant consideration for several reasons. The cask travel path, shown on Figure 6-1 of our January 13 report, was selected such that the cask would be moving away from the suppression pool while it was still in the hatchway opening. If the cask should drop while in the hatchway, it would then land on the dividing wall or to the west of it, away from the suppression pool. It is not considered credible for the cask to fall in other than a vertical attitude unless it is forced into a tilted position before the drop occurs. This could occur only if the cask is moved along travel path A-B before it has cleared the hatchway opening at elevation 1027'-8". This event will be precluded by administrative controls and rehearsals of fuel shipping operations before the cask is lifted onto the 1027'-8" level.

Mr Snaider also asked for the offsite radiological consequences that could be expected should a loaded cask be dropped down the equipment hatch. This situation was analyzed and the potential consequences were found to be significantly lower than the limits specified in 10CFR100. The analysis that was performed is summarized in the attached report.

We are anxious for a resolution on this matter and suggest that a meeting with members of your staff be arranged at your earliest convenience should you have any further questions in regard to our fuel shipping plans.

Yours very truly,

L. O. Mayer

L O Mayer, PE
Manager of Nuclear Support Services

LOM/LLT/ak

cc: J G Keppler
G Charnoff
MPCA
Attn: J W Ferman

Attachment

POTENTIAL RADIOLOGICAL CONSEQUENCES OF CASK DROP DOWN THE REACTOR BUILDING EQUIPMENT HATCH

The radiological consequences at two offsite locations (the site boundary: 500 meters, and the low population zone: 2 miles) were determined for the drop of a two element shipping cask down the reactor building hatchway.

A. Assumptions

The following is a summary of the assumptions that were made in evaluating the off-site doses:

1. All of the assumptions in Regulatory Guide 1.25 were used with the following exception: Pasquill diffusion data given in Regulatory Guide 1.3 and 1.25 were used because equivalent site meteorological data was not available from the Monticello FSAR.

2. The accident is assumed to occur with the containment isolated at a negative containment pressure of 0.25" H_2O. As a result, all releases to the environment are assumed to exit via the Standby Gas Treatment System (SGTS) with a subsequent elevated release from the offgas stack which has a release point 125 meters above the surrounding grade.

3. Each train of the SGTS is capable of replacing the containment atmosphere at the rate of one air change per day (Reference 2). Only one train was assumed to be available at the time of the accident.

4. The gases released from the damaged rods are released to the containment immediately with no holdup in the fuel rods.

5. Radioactive inventory is proportional to fuel power level.

6. No credit is taken for decay of the gaseous radioactivity while in transit from the stack to the recipient.

7. It was assumed that a two element shipping cask is dropped down the reactor building hatchway and damaged such that all of the 98 rods within the cask release their gaseous inventory.

8. Plant release via SGTS, without mixing with the containment atmosphere, is made over a two-hour period (Reference 1)

9. The two bundles considered are of maximum burnup with a radial peaking factor of 1.5 for all 98 rods.

10. Fumigation conditions exist for the first 0.5 hours, followed by 1.5 hours of normal atmospheric dispersion. (Reference 1)

B. Radiological Model

The dose received at an offsite location due to the release of the gaseous radioactive inventory of a damaged fuel rod into the containment must be estimated. The dose was calculated as the product of the following series of terms:

1. (S_j) $\equiv$ average source strength of isotope j in each rod at the time of plant shutdown (Ci).

2. (P_i) $\equiv$ peaking factor for the rod being considered (rod i).

3. (GF_j) $\equiv$ fraction of the total rod inventory of isotope j in the plenum and available for release.

4. (DF_j) $\equiv$ fraction of the original activity of isotope j which remains after decay due to storage in the fuel pool. This term also takes into account any production of isotope j due to the decay of a parent during this time.

5. $(1-N_j)$ $\equiv$ fraction of isotope j which is not trapped in the SGTS filters and is allowed to escape via the offgas stack. (Reference 1)

6a. (RF_k) $\equiv$ fraction of the total radioactive containment inventory which is released to the environment during the time period under consideration.

6b. (BR_k) $\equiv$ breathing rate (for inhalation dose only) for the standard man during the time period under consideration. Breathing rates do not affect the Gamma and Beta (skin) doses and this factor is therefore taken as 1 for the calculation of skin doses.

6c. $(X/Q_k)_z$ $\equiv$ Pasquill diffusion coefficient at location (z) which is applicable during time period k (sec/m^3).

Since (RF_k), (BR_k), and $(X/Q_k)_z$ are all functions of the time period (k) under consideration, the sum of their products will give the required integrated term over the total period of the accident, i.e.,

Let $C_k = (RF_k)\ (BR_k)\ (X/Q_k)_z$, then the required term to be used in the calculation of the dose received at location z is $\sum_k C_k$, where

$$\sum_k C_k = C_1 + C_2 \ldots\ldots\ldots + C_k.$$

7. (DC_j) $\equiv$ dose conversion factor for the j^{th} isotope. This converts the integrated concentration of activity with respect to time into a dose in rems (rem/ci for inhalation or $rem\text{-}m^3/ci\text{-}sec$).

Therefore, the dose (in rems) received at location z due to isotope j, from rod i, is

$$(S_j)\ (P_i)\ (GF_j)\ (DF_j)\ (1-N_j)\ (DC_j) \sum_k C_k \qquad (1)$$

To find the total dose received from one rod we sum over all j isotopes involved. To find the total dose received from all rods, a sum over all i rods is performed. The overall dose, D in rems at offsite location z is:

$$D = \sum_{i} \sum_{j} (S_j)\ (P_i)\ (GF_j)\ (DF_j)\ (1-N_j)\ (DC_j) \sum_{k} C_k \qquad (2)$$

C. Input Data

1. Isotopic Source Terms (S_j) (Reference 3)

Isotope	Ci per Average Rod
I-131	5297
I-132	728
I-133	139
I-134	2.5E-5
I-135	171
Xe-131m	18.8
Xe-133	3655
Xe-133m	65.9
Xe-135	1075
Xe-135m	1.01
Kr-83m	.12
Kr-85	248
Kr-85m	3.45
Kr-87	2.5E-4
Kr-88	.71

2. Pasquill Diffusion Coefficients (Reference 1)
For 125 meter release height;

Time Period After Accident	(X/Q) 500 Meters	2 Miles
0 - 1/2 hr.	1.6×10^{-4}	3.2×10^{-5}
1/2 - 8 hrs.	1.4×10^{-5}	6.3×10^{-6}
8 - 24 hrs.	2.0×10^{-5}	3.2×10^{-6}
1 - 4 days	3.0×10^{-6}	8.1×10^{-7}

D. Radiological Analysis Results

The dose consequences for the equipment hatch drop case are based on equation (2) of Section B. The tabulated results as a function of shipping date for a cask drop in which all of the 98 rods contained in two fuel bundles are assumed failed is presented in the following Table.

Regulations in 10CFR100 require that offsite doses be limited for accident situations to:

1. 25 rems whole body
2. 300 rems thyroid

Based on the results, the conservatively calculated doses for this postulated accident fall well below 10CFR100 limits. Shipments of spent fuel in two element shipping casks may therefore be made at any time after January, 1976, with negligible impact to public health and safety should this postulated accident occur.

DOSE RESULTS FOR EQUIPMENT HATCH DROP CASE

Date of Shipment	Dose in Rems 500m		2mi	
	Thyroid	Whole Body	Thyroid	Whole Body
January 1976	88×10^{-3}	.028	23×10^{-3}	7.1×10^{-3}
April 1976	38×10^{-6}	.027	97×10^{-6}	7.0×10^{-3}
July 1976	17×10^{-9}	.027	4×10^{-9}	6.9×10^{-3}
October 1976	nil	.026	nil	6.8×10^{-3}
January 1977	nil	.026	nil	6.7×10^{-3}

REFERENCES

1. USNRC Regulatory Guide 1.25, Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors, March, 1972.

2. Monticello FSAR, Chapter XIV, Safety Analysis.

3. G.E. Standard Safety Analysis Report (GESSAR), Chapter 15.