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January 4, 1982

Mr. Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Dresden 2
SEP Topics: II-3.A, Hydrologic Description
II-3.B, Flooding Potential and Protection Requirements.
II-3.B.1, Capability of Operating Plants to Cope with Design Basis Flooding Conditions.
III-3.C, Safety Related Water Supply - Ultimate Heat Sink

NRC DOCKET 50-237

Dear Mr. Crutchfield:

A meeting and field visit was held on October 29, 1981 at Dresden Station per request from Greg Cwalina (NRC Integrated Assessment Manager) and NRC consultants working on hydrological considerations of Dresden Unit 2. Those also present at the meeting were Dresden Station personnel, Station Nuclear Enginering personnel and Sargent & Lundy Engineering consultants for Commonwealth Edison.

As a result of this meeting the NRC requested additional information for the following three items to supplement other docketed information concerning hydrologic considerations.

Item 1: Design Basis Groundwater Level

The original calculations have been done using a groundwater level El. 514'-0". As part of the Mark I Long-Term Program, the reactor building structure has been reevaluated and found to be adequate for a groundwater level of 517'-0".

Item 2: Roof Loads Due to Probable Maximium Precepitation

The original design live load for the roof structures of the reactor building, turbine building, and crib house consist of a snow load of 30 lbs./sq. ft.

A re-evaluation of the structural capacity yielded the following results:

Turbine Building = 35 lbs./sq. ft.
Reactor Building = 70 lbs./sq. ft.
Crib House = 60 lbs./sq. ft.

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The height of the parapet of the Turbine Building and the Reactor Building is approiximately 3'-6" and 1'-6" for the Crib House. Assuming that the roof drains are inoperable, the weight of water which could accumulte behind the parapet would exceed the above mentioned roof load capacities. An inservice inspection of the roof is therefore indicated and will be addressed in SEP Topic III-3.C, Inservice Inspection of Water Control Structures.

Item 3: Ultimate Heat Sink

The FSAR states that in case of a postulated failure of the Dresden dam and locks 9 million gallons of water would be entrapped within the circulating water channels. The amount of water required per Unit 2 and 3 to remove decay heat through the isolation condenser is 2,500,000 gallons. This water would not be returned but would be boiled off. An additional small amount of water would be needed to cool whatever pumps and diesels are required to supply the water to the iso-condenser. This cooling water would, of course, be recirculated to the ultimate heat sink. The exact amount of water required for this purpose and its temperature condition is shown in appendix A.

Please address any questions you may have concerning this matter to this office.

One (1) signed original and thirty-nine (39) copies of this transmittal have been provided for your use.

Very truly yours,

T J Rausch

T.J. Rausch
Nuclear Licensing
Administrator
Boiling Water Reactors

SPPJ/rmr
1440D
cc: RIII Resident Inspector, Dresden

APPENDIX A

30-DAY WATER REQUIREMENTS FOR ULTIMATE HEAT SINK

Purpose

The purpose of this calculation is to determine the amount of water required to maintain Dresden 2 in a safe shutdown condition for 30 days.

Assumptions

The decay heat is removed by continuous operation of the isolation condenser as discussed in Reference 1. The ultimate heat sink consists of the approximately 9,000,000 gallons of water impounded in the intake/discharge canal network. The water necessary to remove the decay heat load is boiled off and vented to the atmosphere.

Analysis

The calculation of decay heat to be removed is performed as in Reference 2.

1. Fission Product Decay

For finite reactor operating time (t_o) the fraction of operating power, $\frac{P}{Po}$ (t_o, t_s), to be used for the fission product decay power at a time t_s after shutdown may be calculated as follows:

$$\frac{P}{Po}(\infty, t_s) = \frac{1}{200}\sum_{n=1}^{n=11} A_n \exp(-a_n t_s) \qquad (1)$$

$$\frac{P}{Po}(t_o, t_s) = (1 + K)\frac{P}{Po}(\infty, t_s) - \frac{P}{Po}(\infty, t_o + t_s) \qquad (2)$$

Where:

$\frac{P}{Po}$ = fraction of operating power

t_o = cumulative reactor operating time, seconds

t_s = time after shutdown, seconds

K = uncertainty factor; 0.2 for $0 \leq t_s < 10^3$ and 0.1 for $10^3 \leq t_s \leq 10^7$

A_n, a_n = fit coefficients having the following values:

n	A_n	a_n (sec^{-1})
1	0.5980	1.772×10^{0}
2	1.6500	5.774×10^{-1}
3	3.1000	6.743×10^{-2}
4	3.8700	6.214×10^{-3}
5	2.3300	4.739×10^{-4}
6	1.2900	4.810×10^{-5}
7	0.4620	5.344×10^{-6}
8	0.3280	5.716×10^{-7}
9	0.1700	1.036×10^{-7}
10	0.0865	2.959×10^{-8}
11	0.1140	7.585×10^{-10}

2. Heavy Element Decay Heat

The decay heat generation due to the heavy elements U-239 and N_p-239 may be calculated according to the following expressions:

$$\frac{P\ (U\text{-}239)}{Po} = 2.28 \times 10^{-3}\ C\ \frac{\sigma_{25}}{\sigma_{f25}}\ [1 - \exp(-4.91 \times 10^{-4}\ t_o)]\ [\exp(-4.91 \times 10^{-4}\ t_s)] \qquad (3)$$

$$\frac{P(N_p\text{-}239)}{Po} = 2.17 \times 10^{-3}\ C\ \frac{\sigma_{25}}{\sigma_{f25}} \qquad (4)$$

$$\Big\{ 1.007\ [1 - \exp(-3.41 \times 10^{-6}\ t_o)]\ \exp(-3.41 \times 10^{-6}\ t_s) - 0.007\ [1 - \exp(-4.91 \times 10^{-4}\ t_o)]\ \exp(-4.91 \times 10^{-4}\ t_s) \Big\}$$

Where:

$\frac{P\ (U\text{-}239)}{Po}$ = fraction of operating power due to U-239

$\frac{P\ (N_p\text{-}239)}{Po}$ = fraction of operating power due to N_p-239

t_o = cumulative reactor operating time, seconds

t_s = time after shutdown, seconds

C = conversion ratio , atoms of Pu-239 produced per atom of U-235 consumed

σ_{25} = effective neutron absorption cross section of U-235

σ_{f25} = effective neutron fission cross section of U-235

The product of the terms $C \cdot \frac{\sigma_{25}}{\sigma_{f25}}$ can be conservatively specified as 0.7.

3. <u>Total Integrated Decay Heat</u>

The total decay heat generated to time t_s is calculated as follows:

$$Q(t_s) = Po \int_o^{t_s} \left(\frac{P(\text{fission})}{Po} + \frac{P(\text{U-239})}{Po} + \frac{P(^{N}p\text{-239})}{Po} \right) dt$$

a. $$Q(\text{fission}) = Po \int_o^{t_s} (1+k) \frac{P}{Po} (\infty, t) - \frac{P}{Po} (\infty, t_o + t) \; dt$$

$$= \frac{Po}{200} \int_o^{t_s} (1+k) \sum_{n=1}^{n=11} A_n \exp(-a_n t) - \sum_{n=1}^{n=11} A_n \exp[-a_n (t_o + t)] dt$$

$$= \frac{Po}{200} \left[(1+k) \sum_{n=1}^{n=11} - \frac{A_n}{a_n} \exp(-a_n t) + \sum_{n=1}^{n=11} \frac{A_n}{a_n} \exp(-a_n t_o) \exp(-a_n t) \right]_o^{t_s}$$

$$= \frac{Po}{200} \left[(1+k) \sum_{n=1}^{n=11} \frac{A_n}{a_n} \left(1 - \exp(-a_n t_s) \right) + \sum_{n=1}^{n=11} \frac{A_n}{a_n} \exp(-a_n t_o) \left(\exp(-a_n t_s) - 1 \right) \right]$$

P_o = 2577 MW_t = 102% of rated power

t_o = 5 years of operation = 1.58×10^8 s (conservative assumption since approximately 1/3 of core is reloaded each refueling outage)

$t_s = 30 \text{ days} = 2.59 \times 10^{6} \text{ s}$

$K = 0.1 \text{ for } 10^3 \leq t_s \leq 10^7$

$Q(\text{fission}) = \frac{2577}{200} \text{ MW } (1.33 \times 10^6 \text{ s}) = 1.72 \times 10^7 \text{ MW-s}$

b. $$Q(\text{U-239}) = Po \int_{o}^{t_s} 2.28 \times 10^{-3} (0.7) \left[1-\exp(-4.91 \times 10^{-4} t_o)\right] \left[\exp(-4.91 \times 10^{-4} t)\right] dt$$

$$= Po\ (2.28 \times 10^{-3})(0.7) \left[1-\exp(4.91 \times 10^{-4} t_o)\right] \cdot \left[\frac{\exp(-4.91 \times 10^{-4} t)}{(-4.91 \times 10^{-4})}\right]_{o}^{t_s}$$

$$= Po\ (2.28 \times 10^{-3})(0.7) \left[1-\exp(-4.91 \times 10^{-4} t_o)\right] \left[\frac{\exp(-4.91 \times 10^{-4} t_s)-1}{(-4.91 \times 10^{-4})}\right]$$

$$= 2577 \text{MW } (3.25 \text{ s}) = 8.38 \times 10^3 \text{MW-s}$$

c. $$Q(\text{Np-239}) = Po \int_{o}^{t_s} 2.17 \times 10^{-3} (0.7) \left\{1.007 \cdot \left[1-\exp(-3.41 \times 10^{-6} t_o)\right] \cdot \exp(-3.41 \times 10^{-6} t) - 0.007 \left[1-\exp(-4.91 \times 10^{-4} t_o)\right] \cdot \exp(-4.91 \times 10^{-4} t)\right\} dt$$

$$= Po\ (2.17 \times 10^{-3})(0.7) \left\{1.007 \left[1-\exp(-3.41 \times 10^{-6} t_o)\right] \cdot \frac{\exp(-3.41 \times 10^{-6} t)}{(-3.41 \times 10^{-6})}\right.$$

$$-0.007 \left[1-\exp(-4.91 \times 10^{-4}\, t_o)\right] \left.\frac{\exp(-4.91 \times 10^{-4}\, t)}{(-4.91 \times 10^{-4})}\right\}_{o}^{t_s}$$

$$= 2577\text{MW } (448 \text{ s}) = 1.15 \times 10^6 \text{ MW-s}$$

d. $\text{Qtotal} = (1.72 \times 10^7 + 8.38 \times 10^3 + 1.15 \times 10^6)$ MW-s

$= 1.84 \times 10^7$ MW-s total decay heat generated for 30 days after shutdown

4. Water Requirements

Assume cooling water entering the isolation condenser is at 100°F, with enthalpy h_{in} of 68.1 Btu/lb, and specific volume V_{in} of 0.01613 ft^3/lb. Also conservatively assume that boiling takes place in the isolation condenser at the rated shell pressure of 25 psig (this assumption results in minimal heat removal per pound of water). The resulting enthalpy h_{shell} is 933.6 Btu/lb. The mass of water required to remove total decay heat may then be expressed

$$M = \frac{\text{Qtotal}}{(h_{shell} - h_{in})}$$

$$= \frac{(1.84 \times 10^7 \text{ MW-s})(9.486 \text{ Btu/kw-s})}{(933.6 - 68.1) \text{ Btu/lb}}$$

$$M = 2.02 \times 10^7 \text{ lb of water}$$

The resulting volume in gallons is

$$V = (2.02 \times 10^7 \text{ lb})(.01613 \text{ ft}^3/\text{lb})(7.481 \text{ gal/ft}^3)$$

$$V = 2.43 \times 10^6 \text{ gallons of water}$$

This result represents the water required to remove by boiling the decay heat produced by one 2577 MW_t reactor for 30 days. An additional small amount of cooling water to cool a diesel generator could be recirculated to the cribhouse forebay from the discharge canal after dissipating its heat to the environment.

References

1. Dresden 2&3 Safety Analysis Report, Amendment 9/10, Question I.F.

2. Branch Technical Position ASB 9-2, Residual Decay Energy for Light-Water Reactors for Long-Term Cooling, Rev. 2, July 1981.