Omaha Public Power District

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June 26, 1981

Mr. Robert A. Clark, Chief
U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Division of Licensing
Operating Reactors Branch No. 3
Washington, D.C. 20555

Reference: Docket No. 50-285

Dear Mr. Clark:

Omaha Public Power District's response to the Commission's letter dated April 23, 1981, requesting resubmittal of an application for amendment to Facility License DPR-40, regarding iodine spiking is attached. Please note that the District has addressed the specific concerns identified in the Commission's letter. As supported by the attached discussions, the District identified only two minor changes to the District's amendment application dated November 17, 1980. Accordingly, Attachments 5 and 6 are forwarded to replace pages 2-8 and 3-18 of the November 17, 1980, application.

Sincerely,

W. C. Jones
Division Manager
Production Operations

RECEIVED
JUN 29 1981
U.S. NUCLEAR REGULATORY COMMISSION

WCJ/KJM/TLP:jmm

Attachments

cc: LeBoeuf, Lamb, Leiby & MacRae
1333 New Hampshire Avenue, N.W.
Washington, D.C. 20036

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Concern 1

The equilibrium primary coolant activity limit was essentially removed (i.e., the absence of a limit on the fraction of time that the plant can operate with an iodine "spike", in effect, negated the equilibrium value and substituted the "spike" maximum as the only real limit).

Response

The equilibrium primary coolant activity limit of 2 μCi/gm Dose Equivalent I-131 was included in the District's application for "Amendment of Operating License", dated November 17, 1980. The District had previously agreed to adding to the Technical Specifications a cumulative operating time limit of 800 hours above the 2 μCi/gm limit for any 12 consecutive months. A revised page 2-8 of the Fort Calhoun Station Technical Specifications is attached (Attachment 5) for incorporation of this new requirement. It should be noted that the safety significance of this operating limit is not clear, since the protection of the public health and safety is insured following any design basis accident coincident with iodine spiking, as demonstrated in Attachments 2 and 3.

Concern 2

The proposed sampling frequencies would not assure detection of iodine spikes in the reactor coolant.

Response

The sampling requirement assuring the detection of iodine spikes in the reactor coolant is Item 4 b), Table 4-4-4 of the STS. The District, after reviewing the 7-year operating history of the plant, proposed a sampling requirement of one sample within 24 hours, following a thermal power change greater than 15 percent per hour. This sampling frequency has been changed in Attachment 6 to "within 2 to 8 hours". Using 8 hours as an upper limit, instead of 6 hours as called out in the STS, would provide operating flexibility to accommodate changes of personnel between shifts.

Concern 3

The proposed equilibrium value was twice the STS value.

Response

The radioactivity of the reactor coolant for DOSE EQUIVALENT I-131 is 2.0 μCi/gm instead of 1.0 μCi/gm based upon the following:

1. The 2.0 μCi/gm limit represents approximately 1% failed fuel as referenced in Table 11.1.5 of the FSAR,

2. The 2.0 μCi/gm limit was determined using the methodology presented in NUREG-0017 for 10 CFR Part 50, Appendix I evaluations,

3. The thyroid doses under accident conditions using 2.0 μCi/gm are less than 1% of 10 CFR Part 100 value, and

4. The proposed equilibrium value will not have any adverse effect on the transient limit of 60 μCi/gm, specifically under "Iodine Spiking" conditions. Please also see the response to Concern 5.

Concern 4

Your proposed Technical Specification as submitted could result in postulated off-site doses substantially in excess of 10 CFR Part 100 guidelines in the event of a steam generator tube rupture under iodine spiking conditions (i.e., with coolant activity up to 60 μCi/gm).

Response

The coolant concentration of 60 μCi/gm I-131 DOSE EQUIVALENT was not proposed by the District. This concentration was derived from Figure 3.4.1 of the STS. The District used the limiting value of 60 μCi/gm for 0-100 percent of rated thermal power instead of Figure 3.4.1 of the STS. Figure 3.4.1 of the STS allows the radioactivity to exceed 60 μCi/gm whenever the reactor thermal power is less than 80%; therefore, the District's approach is more conservative than the STS.

Two analyses, pertinent to the proposed Technical Specifications, were performed to determine the radiological consequences of a steam generator tube rupture and a main steam line break with the assumptions of coincident iodine spiking and an accident (initial activity of 2.0 μCi/gm I-131 DOSE EQUIVALENT and a final activity, following the accident, of 60 μCi/gm I-131 DOSE EQUIVALENT). The results of these analyses show that the thyroid and whole body doses would be well within the limits of 10 CFR Part 100.

The analysis and results of the main steam line break are provided as Attachment 2. The analysis and results of a steam generator tube rupture are provided as Attachment 3.

Concern 5

Your proposal to double the STS maximum (equilibrium*) value could lead to "spiked" coolant activity levels twice the transient limit of 60 μCi/gm.

Response

The reasons for proposing the equilibrium reactor coolant concentration of 2.0 μCi/gm instead of 1.0 μCi/gm are provided in response to Concern 3. After reviewing the plant operating data and the data presented in Table II of the NRC paper on "Iodine Spiking" and in Combustion Engineering's topical report, "CENPD-180", the District finds no evidence

*Added by the District.

leading to the conclusion that by doubling the equilibrium values the spiked coolant activity levels would be twice the transient limit of 60 μCi/gm. Specifically, the iodine spikes experienced by the Fort Calhoun Station have been in the range of 12 to 16 μCi/gm with variations in equilibrium levels between 0.09 and 1.9 μCi/gm. Therefore, the District has observed no relationship between equilibrium and spike activity levels.

Concern 6

If you still believe that operation above steady state limit for up to 100 hours is necessary and justified (versus 48 hours per the STS), we request that you provide justification beyond that offered in your November 17, 1980 submittal.

Response

The longer duration (100 hours or longer) to restore the coolant activity within acceptable values is based upon the specific design of the Fort Calhoun Station purification system. The coolant activity removal constant is dependent upon the purification system flow rates and the total volume of the reactor coolant system. Again, the bases for "48 hours operation", as per the STS, are not known but it is believed that "48 hours" are possibly based on a much more efficient purification system.

The justification for "up to 100 hours operation" is provided in the form of graphs in Attachment 4. The values for coolant concentrations for April 1 through 6, 1977, September 28 through October 3, 1977, June 4 through 15, 1979, and December 14 through 24, 1979 have been normalized to postulate the initial steady state concentrations of 2.0 μCi/gm I-131 DOSE EQUIVALENT. The data for August 15 through 18, 1976 were not changed as the coolant concentrations were already in the vicinity of 2.0 μCi/gm. The time intervals, shown on Attachment 4, ranging from 67 hours to 120 hours are the durations from the beginning of a spike till the activity returns to steady-state conditions. Therefore, the District requests the 100 hour limit be retained in the Amendment Application.

RADIOLOGICAL CONSEQUENCES OF A STEAM GENERATOR TUBE RUPTURE

Physical Model

The evaluation of the radiological consequences of a postulated steam generator tube rupture assumes a complete severance of a single steam generator tube while the reactor is operating at full rated power and a coincident loss of offsite power at the time of reactor trip. Occurrence of the accident leads to an increase in contamination of the secondary system due to reactor coolant leakage through the tube break. A reactor trip occurs automatically as a result of low pressurizer pressure after the tube rupture occurs. The reactor trip automatically trips the turbine.

The resulting increase in radioactivity in the secondary system is detected by radiation monitors. A coincident loss of offsite station power causes closure of the turbine bypass valves to protect the condenser. The steam generator pressure will increase rapidly, resulting in steam discharge as well as activity release through the main steam safety valves. Venting from the affected steam generator, i.e., the steam generator which experiences tube rupture, continues until the secondary system pressure is below the main steam safety valve setpoint. At this time, the affected steam generator is effectively isolated, and thereafter, no steam or activity is assumed to be released from the affected steam generator. The remaining unaffected steam generators remove core decay heat by venting steam through the main safety valves, atmospheric dump valve, and steam driven auxiliary turbine until cooldown can be accomplished with the shutdown cooling system.

The analysis of the radiological consequences of a steam generator tube rupture considers the most severe release of secondary activity as well as reactor activity leaked from the tube break. The inventory of iodine and noble gas fission product activity available for release to the environment is a function of the primary-to-secondary coolant leakage rate, the percentage of defective fuel in the core, and the mass of steam discharged to the environment. Conservative assumptions are made for all these parameters.

In this evaluation, a case with coincident iodine spike which already exists due to a previous power transient was considered. The mathematical models, assumptions, and parameters used in this analysis were identical with the design basis SGTR without an iodine spike as described in the above paragraphs with the following exception:

The reactor coolant system inventory was assumed to be 60 μC/g dose equivalent Iodine 131. This 60 μC/g is the Technical Specification limit for full power operation following an iodine spike for up to 48 hours.

Assumptions and Conditions

The major assumptions and parameters assumed in the analysis are itemized in Table I.

RADIOLOGICAL CONSEQUENCES OF A STEAM GENERATOR TUBE RUPTURE (Continued)

Assumptions and Conditions (Continued)

The following assumptions and parameters are used to calculate the activity releases and offsite doses for a steam generator tube rupture (SGTR):

1. The reactor coolant system equilibrium activity is 60 μC/cc DEC I-131.

2. The steam generator equilibrium activity for both steam generators is assumed to be 0.1 μC/g dose equivalent I-131 prior to the accident.

3. Offsite power is lost; the main condenser is not available for steam relief via the turbine bypass system.

4. Following the accident, no additional steam and radioactivity are released to the environment when the shutdown cooling system is placed in operation.

5. There is no main condenser evacuation system release and no steam generator blowdown during the accident.

6. Only 1 steam generator is affected.

7. The amount of noble gas activity released is equal to the amount present in the reactor coolant discharged into the secondary side following the tube rupture. The amount of noble gas activity contained in the secondary system is negligible in comparison.

8. Iodine activity released is based on the equilibrium activity present in the steam generators (0.1 μC/g dose equivalent I-131) and the amount of activity present in the reactor coolant discharged into the affected steam generator.

9. Thirty minutes after the accident, the affected unit is isolated. No steam and fission product activities are released from the affected steam generator thereafter.

10. The total amount of discharge of reactor coolant into the secondary system through the rupture is 42,300 pounds (in 30 minutes).

11. A post-accident DF of 10 was used in the steam generator between the water and steam phases.

12. The primary-to-secondary leakage of 8640 lbm/d (1.0 gal/min) is assumed to be applicable to the unaffected steam generator. The portion of the noble gas activity from the primary-to-secondary leakage attributed to the unaffected steam generator is assumed to be released during the course of the accident.

RADIOLOGICAL CONSEQUENCES OF A STEAM GENERATOR TUBE RUPTURE (Continued)

Assumptions and Conditions (Continued)

13. The amount of discharge of steam from the affected steam generator is assumed to be 3.55 x 10^6 pounds.

14. The activity released from the affected and unaffected steam generators is immediately vented to the atmosphere. No credit for radioactive decay for isotopes in transit to dose points.

Uncertainties and Conservatisms

The uncertainties and conservatisms in the assumptions used to evaluate the radiological consequences of a steam generator tube rupture are as follows:

1. Reactor coolant equilibrium activities are based on 1% failed fuel, which is greater by a factor of 2 to 8 than that normally observed in past PWR operation.

2. Steam generator equilibrium activity for both steam generators is assumed to be equal to the Technical Specification limit. The Technical Specification limits are conservatively derived based on accidents such as the SGTR.

3. Tube rupture of the steam generator is assumed to be a double-ended severance of a single steam generator tube. This is a conservative assumption since the steam generator tubes are constructed of highly ductile materials. The more probable mode of tube failure is 1 of minor leaks of undetermined origin. Activity in the secondary steam system is subject to continual surveillance, and the accumulation of activity from minor leaks that exceed the limits established in the Technical Specifications would lead to reactor shutdown. Therefore, it is unlikely that the total amount of activity considered available for release in this analysis would ever be realized.

4. The coincident loss of offsite power with the occurrence of the reactor trip following the steam generator tube rupture is a conservative assumption. In the event of availability of offsite power, the turbine bypass valves will open, relieving steam to the main condenser. This will reduce the amount of steam and entrained activity discharged directly to the environment from the unaffected steam generators.

5. The meteorological conditions assumed to be present at the site during the course of the accident are based on χ/Q values for the exclusion area boundary or LPZ outer boundary. Furthermore, no credit has been taken for the transit time required for activity to travel from the point of release to the exclusion area boundary or LPZ outer boundary. Hence, the radiological consequences evaluated under these conditions are conservative.

RADIOLOGICAL CONSEQUENCES OF A STEAM GENERATOR TUBE RUPTURE (Continued)

6. A conservative steam generator decontamination factor (DF) of 10 is used in the cooldown phase (release to atmospheric dump valve).

Results

The results of radiological consequences of a steam generator tube rupture are presented in Table III. The values for thyroid and whole body gamma dose show that the doses calculated using the above defined conservative assumptions are well within the limits of 10CFR100.

Table I
Parameters Used in Evaluating the Radiological
Consequences of a Steam Generator Tube Rupture

Parameter	Assumption	Reference
Source Data:		
Power level, Mwt	1,500	FSAR
Steam generator tube leakage, lb/d	8,640 (1 gal/min)	Assumption
Equilibrium reactor coolant activity		
Coincident (existing) iodine spike μc/g dose equivalent I-131	60	Assumption
Equilibrium secondary system activity	0.1 μc/g dose equivalent I-131	Assumption
Activity Release Data:		
Steam discharge, lb		
Affected steam generator		
Reactor coolant leakage to steam generator (0-30 min)	42,300	
Mass of steam released	3.551×10^{6}	
Dispersion Data:		
Atmospheric dispersion factors, sec/m^3		
EAB	4.4×10^{-4}	
LPZ (0-8 hours)	1.57×10^{-5}	
Iodine decontamination factors for steam generators (between water and steam phase)	10	Assumption

Table II
Activity Released from Steam Generator

Nuclide	Activity (Curies)
DEC I-131	2.7423 (2)
Kr-83m	1.74
Kr-85m	9.61
Kr-85	1.67 (3)
Kr-87	4.71
Kr-88	17.17
Xe-131m	13.64
Xe-133m	21.08
Xe-133	1.90 (3)
Xe-135m	6.5 (-1)
Xe-135	32.07
Xe-138	2.12

Table III
Radiological Consequences of a Postulated
Steam Generator Tube Rupture

Results	Value
Exclusion Area Boundary Dose (duration), Rem	
Coincident (existing) iodine spike	
Thyroid	62
Total Body Gamma	2.72×10^{-2}
LPZ Outer Boundary Dose (duration)*, rem	
Coincident (existing) iodine spike	
Thyroid	2.21
Total Whole Body	9.7×10^{-4}

* For conservatism, the LPZ dose calculations are based on a dispersion factor for 0-8 hours.

RADIOLOGICAL CONSEQUENCES OF A MAIN STEAM LINE BREAK

Physical Model

To evaluate the radiological consequences due to a postulated main steam line break (outside containment), it is assumed that there is a complete severance of a main steam line outside the containment with the plant in a hot zero power condition where the transient is initiated shortly after full power operation. It is also assumed that there is a simultaneous loss of offsite power. The hot zero power condition assures the maximum water inventory in the steam generators and the shutdown from full power (in conjunction with the loss of off-site power) assures the maximum decay heat which must be removed by manual control of the atmospheric dump valve associated with the intact steam generator.

The main steam isolation valves are installed in the main steam lines from each steam generator, downstream from the safety relief valves and atmospheric dump valves outside containment. The severance of the main steam line is assumed to be upstream of the main steam isolation valve. A reactor trip is actuated by a low steam generator pressure signal. A main steam isolation signal (MSIS) is actuated to shut the main steam isolation valves from both steam generators. The affected steam generator (steam generator connected to the severed steam line) blows down completely. The steam is vented directly to the atmosphere. The atmospheric dump valve of the unaffected steam generator is used to initiate a 75F/h cooldown of the reactor coolant system 1800 seconds after initiation of the accident. The steam is vented directly to the atmosphere. Mass release from the unaffected steam generator is terminated when the shutdown cooling system is initiated at a reactor coolant system temperature of 300°F.

In this evaluation, a case with an iodine spike caused by the main steam line break accident was evaluated for radiological consequences. The mathematical models, assumptions, and parameters used in this analysis were identical with the design basis main steam line break accident without an iodine spike and described in the above paragraphs with the following exception:

Prior to the main steam line break accident the reactor coolant system activity is based on 1% failed fuel.

However, at the initiation of the MSLB accident, the I-131 equivalent souce term (released from fuel) is assumed to increase. The iodine release rate is assumed to increase by a factor of 500.

Assumptions and Conditions

The major assumptions, parameters, and calculational methods used in the design basis analysis are presented in Table I. Additional clarification is provided as follows:

RADIOLOGICAL CONSEQUENCES OF A MAIN STEAM LINE BREAK (Continued)

Assumptions and Conditions (Continued)

1. Reactor Coolant Activity

   The reactor coolant equilibrium activity is based on long term operation at 100% of the ultimate core power level of 1500 Mwt and 1% failed fuel. Source terms are listed in Table I.

2. Secondary System Activity

   The activity in steam generators is conservatively assumed to be equal to 0.1 μCi/gm dose equivalent Iodine-131 (I-131).

3. Primary-to-Secondary Leakage

   The primary-to-secondary leakage of 1 gal/min (Technical Specification limit) was assumed to continue through the affected steam generator at a constant rate until shutdown cooling is initiated.

Uncertainties and Conservatisms

1. An 8640 lbm/d (1 gal/min) steam generator primary-to-secondary leakage is assumed, which is greater by a factor of 50 to 200 than that normally observed in past PWR operation.

2. The steam generator equilibrium activity for both steam generators is assumed to be equal to the Technical Specification limit (0.1 μCi/g dose equivalent I-131) for the duration of the accident.

3. The meteorological conditions assumed to be present at the site during the course of the accident are based on χ/Q values for the exclusion area boundary or LPZ outer boundary. Furthermore, no credit has been taken for the transit time required for activity to travel from the point of release to the exclusion area boundary or LPZ outer boundary. Hence, the radiological consequences evaluated under these conditions will be conservative.

Results

The results of radiological consequences due to a postulated Main Steam Line Break are presented in Table III. The values for thyroid and whole body gamma dose show that the doses calculated using the above defined conservative assumptions are well within the limits of 10CFR100.

Table I
Parameters Used in Evaluating the Radiological Consequences of a Main Steam Line Break Accident (MSLBA)

Parameter	Assumptions	Reference
Data and assumptions used to estimate radioactive source		
General		
Power level, Mwt	1500	FSAR
Burnup	End of Cycle	Assumption
Reactor coolant activity after accident Iodine spike caused by accident	60 μCi/gm DEC I-131	Assumption
Steam generator activity before accident	0.1 μCi/gm dose equiv. I-131	Assumption
General		
Loss of offsite power	yes	Assumption
Credit for radioactive decay in transit to dose point	no	Assumption
Affected steam generator		
Primary-to-secondary leakage rate, lb/d	8,640 (1 gal/min)	Assumption
Secondary mass release to atmosphere (through severed line), lb_m	233,498	
Mass of primary-to-secondary leakage, lb_m	491	Assumption
Steam generator decontamination factor between steam and water phase	1	Assumption
Unaffected steam generator		
Primary-to-secondary leakage rate, lb/d	0	Assumption
Dispersion Data:		
Atmospheric dispersion factors, sec/m^3		
EAB	4.4×10^{-4}	
LPZ (0-8 hours)	1.57×10^{-5}	

Table II
Activity Released from the Steam Generator

Nuclide	Activity (Curies)
DEC I-131	23.93
Kr-83m	1.86 (-2)
Kr-85m	1.08 (-1)
Kr-85	1.93
Kr-87	4.8 (-2)
Kr-88	2.12 (-1)
Xe-131m	1.61 (-1)
Xe-133m	2.44 (-1)
Xe-133	2.20 (1)
Xe-135m	4.83 (-3)
Xe-135	3.63 (-1)
Xe-138	1.54 (-2)

Table III
Radiological Consequences Due to a Postulated
Main Steam Line Break

Result	Iodine Spike Caused by Accident
Exclusion Area Boundary Dose (duration) rem:	
Thyroid	5.41
Total Body Gamma	1.19×10^{-3}
LPZ Outer Boundary Dose * (duration), rem:	
Thyroid	1.19×10^{-1}
Total Body Gamma	4.22×10^{-5}

* For conservatism, the LPZ dose calculations are based on a dispersion factor for 0-8 hours.

Attachment 4