

5.14 Steam Generator Tube Rupture Event

5.14.1 Definition of Event

A steam generator tube rupture event is defined as a loss of integrity of the barrier between the reactor coolant system and the main steam system as a result of the penetration of a steam generator tube. The most likely cause of such an event is the complete failure of a tube-to-sheet weld or the rapid propagation of a circumferential tube crack. The integrity of this barrier is significant from the radiological safety standpoint, because a leaking steam generator tube will allow the transportation of reactor coolant into the main steam system. Radioactivity contained in the reactor coolant mixes with shell side water in the affected steam generator. This radioactive contamination is transported by steam to the turbine and then to the condenser, or directly to the condenser via the steam dump and bypass system. Noncondensable radioactive materials will be discharged through the condenser vacuum pumps to the atmosphere.

As a result of the tube rupture the primary coolant transfer causes the water level in the affected steam generator to increase and the pressurizer level to decrease, provided that the tube leak rate exceeds the charging pump capacities. In the case of a double-ended tube rupture, the Design Basis event, the leak rate far exceeds the charging pump capacities, and consequently pressurizer level will decrease. The decrease in the pressurizer level and the inability of the heaters to maintain pressurizer pressure causes the RCS pressure to decrease. The rate of primary system depressurization is determined by the leak rate, charging flow rate and the pressurizer heater capacity. The decreasing level will eventually cause the pressurizer to empty, which will result in the primary pressure dropping to the hot leg saturation pressure. Since no credit is taken in the analysis for the manual trip upon observation of either an RCS leak rate in excess of the Technical Specification or excessive secondary side activity, a reactor trip will result from low pressurizer pressure.

The decrease in the primary pressure will also initiate a Safety Injection Actuation Signal (SIAS). If a loss of offsite power (i.e., the case for limiting radiological consequences) is not assumed, a manual trip of the reactor coolant pumps (RCPs) at 1350 psia will be performed in accordance with Fort Calhoun Station Emergency Operating Procedures. This will reduce the primary to secondary heat transfer rate and result in a higher hot leg temperature, which slows down the rate of primary system depressurization. As the pressure drops below the High Pressure Safety Injection (HPSI) pump shut-off head, Safety Injection (SI) flow is delivered to the core. The SI and charging flow partially offsets the mass loss from the ruptured tube, and results in slowing the depressurization of the primary system. Consequently, these two effects result in a higher primary pressure, which increases the primary-to-secondary leak rate.

Upon reactor trip the steam dump and bypass system will be actuated to reduce secondary pressure. The main feedwater flow will also ramp

5.14 Steam Generator Tube Rupture Event (Continued)

5.14.1 Definition of Event (Continued)

down to the 11 percent of full feed flow in 20 seconds and cooldown of the primary system will continue. After 1800 seconds, the operator is assumed to isolate the affected steam generator and continue the cooldown of the primary system using the unaffected steam generator.

5.14.2 Analysis Criteria

The steam generator tube rupture event is classified as a postulated accident for which the dose rates must be within the 10 CFR 100 guidelines.

5.14.3 Objective of Analysis

The objective of the analysis is to demonstrate that the 2-hour site boundary dose will not exceed the 10 CFR 100 guidelines in case of a steam generator tube rupture event.

5.14.4 Key Parameters and Analysis Assumptions

The key parameters used in the steam generator tube rupture event are given in Table 5.14.4-1. Assumptions used in the analysis to maximize the 2-hour site boundary dose include:

5.14.4.1 NSSS Response

1. Maximum charging flow and zero letdown flow are assumed.
2. A loss of offsite power is assumed concurrent with reactor trip. Manual operation of the Auxiliary Feedwater System is assumed upon the loss of the main feedwater system.
3. The steam dump and bypass system is assumed to be inoperable.
4. The leak rate, from the primary-to-secondary, is maximized using the combination of maximum primary pressure with minimum secondary pressure and coastdown of reactor coolant pumps following the loss of off-site power.

5.14.4.2 Radiological Consequences

Thyroid Doses

In order to calculate the site boundary thyroid dose, the total activity, I_{DEQ} , released to the environment must be first determined. The primary release paths for the

5.14 Steam Generator Tube Rupture Event (Continued)

5.14.4 Key Parameters and Analysis Assumptions (Continued)

5.14.4.2 Radiological Consequences (Continued)

Thyroid Doses (Continued)

activity released to the atmosphere are through the Main Steam Safety Valves (MSSV's), the Atmospheric Dump Valve, FW-10 vent or the condenser off-gas. The iodine transported from the steam generators is made up of two components, the iodine mechanically entrained in the moisture of the steam and the iodine which is volatile. A decontamination factor of 10 is used in the event when steam is released directly to the environment via MSSV's or the Atmospheric Dump Valve, and a decontamination factor of 2000 is used for release through the condenser off-gas or the vent on FW-10. The decontamination factor is defined as the ratio of the concentration of iodine in water to the concentration of iodine in steam. The primary coolant iodine activities, I_{DEQ} , are assumed to be the upper limits of the values permitted by the Technical Specifications.

Whole Body Dose

The whole body dose is due to beta and gamma radiation from a cloud of noble gases released from the plant. The noble gases are any Krypton and Xenon gases present in the RCS coolant. The inventory of noble gases in the coolant is assumed to be based on one (1) percent fuel failure as listed in Table 5.14.4-2. (All the noble gas activity contained in the primary coolant which is leaked through to the secondary side is assumed to be released to the environment.)

5.14.5 Analysis Method

The analysis methods used by OPPD to analyze the first 30 minutes in a steam generator tube rupture event consist of using the CESEC-III computer code to simulate the event, utilizing the analysis assumption listed in Section 5.14.4 (above) as input and extracting information regarding the amount of steam and its path leaving both steam generators. The affected generator is assumed to be isolated after 30 minutes.

Steam and associated radioactive nuclides released to the environment from the unaffected steam generator via the manually operated atmospheric dump valve, after the affected steam generator has been isolated, is estimated based on the sum of (1) the sensible heat released from the primary coolant (including the affected steam generator) utilizing the maximum Technical Specification cooldown rate, (2) the sensible heat released from the unaffected steam generator cooldown, (3) the decay power, after 30 minutes, is assumed constant, and (4) 10

5.14 Steam Generator Tube Rupture Event (Continued)

5.14.5 Analysis Method (Continued)

percent of the total heat released in an hour and one-half after the affected steam generator has been isolated is added to account for cooldown of structural material. The steam releases are extrapolated for an additional 1½ hours from the isolation of the affected generator to provide input into the radioactive release calculations to determine a 2 hour dose at the site boundary.

5.14.6 Analysis Results

The results of the steam generator tube rupture are contained in the Fort Calhoun Station Unit No. 1 USAR, Section 14.14, and dose rates are within the 10 CFR 100 guidelines.

5.14.7 Conservatism of Results

1. The pressurizer level control system is operated in the AUTO mode rather than the MANUAL mode as assumed in the analysis.
2. Actual primary pressure is lower and secondary pressure is higher which would result in a lower leak rate.
3. All three HPSI pumps are assumed to start. (For a loss of off-site power only two pumps are automatically loaded on the diesel, the third pump must be manually loaded by the operator).
4. Actual time delay after the SIAS until the HPSI pumps reach full speed is 30 seconds instead of one second as assumed in the analysis.
5. The Steam Dump and Bypass system is assumed unavailable concurrent with turbine trip which establishes a direct path for radioactive steam to the environment through the secondary safety valves, thus maximizing the dose rates.
6. The main feedwater system is assumed to be unavailable due to loss of offsite power concurrent with the turbine trip and manual operation of the auxiliary feedwater system is assumed. In case of a normal reactor trip, the main feedwater flow ramps down to 100 lbm/sec-SG compared to the AFW flow rate of 18 lbm/sec-SG. This lower flow rate results in higher primary and secondary temperatures and pressures.
7. The primary coolant activity is assumed to be the maximum permitted by the Technical Specification limits and constant throughout the event. No credit is taken for the reduction of specific fission product activity in the reactor coolant system resulting from the dilution with safety injection and charging flows. Similarly no credit is taken for the dilution of activity on the secondary side with the main or auxiliary feedwater. The ini-

5.14 Steam Generator Tube Rupture Event (Continued)

5.14.7 Conservatism of Results (Continued)

tial mass of the coolant in the affected steam generator is assumed to be constant for the purpose of determining radiological consequences.

8. No credit is taken for decay for the radioactive isotopes in transit to dose points.

Table 5.14.4-1

KEY PARAMETERS ASSUMED IN THE STEAM GENERATOR TUBE RUPTURE EVENT ANALYSIS

NSSS:

<u>Parameter</u>	<u>Units</u>	<u>Value</u>
Initial Core Power Level	%	102
Initial Core Inlet Temperature	°F	Maximum allowed by Tech. Specs.
Initial RCS Pressure	psia	Maximum allowed by Tech. Specs.
Initial Steam Generator Pressure	psia	Minimum value corresponding to core inlet temperature operating range.
High Pressure Safety Injection	No. Pumps Operable	Maximum number, i.e., 3
Charging Pump Flow	No. Pumps Operable	Maximum number, i.e., 3
Core Average H_{gap}	BTU/hr-Ft ² -°F	Minimum predicted during core lifetime.
CEA Drop Time	sec.	Maximum allowed by Tech. Specs.
Scram Reactivity Worth	% $\Delta\rho$	Minimum predicted during core lifetime.
<u>Primary Coolant Activities:</u>		
I_{DEQ}	$\mu\text{Ci/gm}$	Maximum allowed by Tech. Specs.
Noble Gases	$\mu\text{Ci/gm}$	Based on 1 percent fuel failure
<u>Radiological:</u>		
Decontamination Factor	release through MSSV's release through condenser	10 2000
Breathing Rate	m ³ /sec.	3.47×10^{-4}
Dose Conversion Factor (I_{DEQ})	Rem/Ci	1.48×10^6
Dispersion Factor	sec/m ³	6.3E-4

Table 5.14.4-2

NOBLE GAS INVENTORY IN THE REACTOR COOLANT
 BASED ON 1% FAILED FUEL

<u>Nuclide</u>	<u>Coolant (Curies Ci)</u>	<u>E_r (Mev/dis)</u>	<u>E_β (Mev/dis)</u>	<u>F_R (Fraction of Total Ci)</u>
Kr-85m	211	.284	.156	7.58E-3
Kr-85	136	.249	.0021	4.89E-3
Kr-87	110	1.334	.856	3.95E-3
Kr-88	299	.367	2.0	1.07E-2
Xe-133	2.591E4	.099	.0728	9.31E-1
Xe-135	1.116E3	.307	.248	4.01E-2
Xe-138	48.2	.961	.932	1.73E-3

Omaha Public Power District
1623 Harney Omaha, Nebraska 68102 2247
402/536-4000

September 30, 1987
LIC-87-598

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

- References:
1. Docket No. 50-285
 2. "OPPD Nuclear Analysis, Reload Core Analysis Methodology, Transient and Accident Methods and Verification,"
OPPD-NA-8303-P, Revision 1, dated November 1986

Gentlemen:

SUBJECT: Addition of Section 5.15, Steam Generator Tube Rupture Event

Enclosed for your review and approval is OPPD's proposed Steam Generator Tube Rupture Event Methodology which, if approved, will be incorporated into Reference 2.

The Updated Safety Analysis Report (USAR), Chapter 14, Section 14, Analysis of the Steam Generator Tube Rupture (SGTR) was analyzed by Combustion Engineering for the original license submittal, the FSAR. OPPD proposes that the USAR analysis be updated to the current Cycle 11 operating condition; after acceptance of this methodology change. A revised and retyped methodology report will be transmitted upon your approval of this submittal.

Please note that pursuant to 10 CFR 2.790(b)(1), the attached information has been deemed trade secrets and/or privileged commercial information by OPPD. Accordingly, please find attached OPPD's application for withholding this information from public disclosure. Enclosed herewith also is a check in the amount of \$150.00 per 10 CFR 170 for review of this document.

Sincerely,

R. L. Andrews

R. L. Andrews
Division Manager
Nuclear Production

RLA/rh

Attachments

- c:
- LeBoeuf, Lamb, Leiby & MacRae
1333 New Hampshire Avenue, N.W.
Washington, DC 20036
 - R. D. Martin, NRC Regional Administrator, Region IV
 - A. Bournia, NRC Project Manager
 - P. H. Harrell, NRC Senior Resident Inspector

PA01
w/ check \$150
**154085*
11