WOLF CREEK
NUCLEAR OPERATING CORPORATION

Bart D. Withers
President and
Chief Executive Officer

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WM 92-0075

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station P1-137
Washington, D. C. 20555

Reference: 1) Letter dated July 24, 1990 from D. V. Pickett, NRC, to B. D. Withers, WCNOC
2) Letter WM 90-0194 dated November 30, 1990 from B. D. Withers, WCNOC to NRC
3) Letter WM 91-0004 dated January 15, 1991 from B. D. Withers, WCNOC to NRC

Subject: Docket No. 50-482: Results of Additional Demonstrations of Steam Generator Tube Rupture Operator Action Times for Wolf Creek Generating Station

Gentlemen:

The purpose of this letter is to provide the results of additional demonstrations of Steam Generator Tube Rupture (SGTR) operator action times for Wolf Creek Generating Station (WCGS). Reference 1 requested Wolf Creek Nuclear Operating Corporation (WCNOC) provide further information that demonstrated that the operator response times assumed in WCGS analysis were representative of the current operator population at the plant and that the maximum response times fell within the bounds of the analysis. Reference 2 provided demonstrated operator response times from five simulated SGTR scenarios. In subsequent discussion between WCNOC and the NRC staff, the staff requested that WCNOC demonstrate the simulated SGTR scenarios on a minimum of 80 percent of the current operator population. Reference 3 provided WCNOC's commitment to a one-time performance of additional simulated SGTR scenarios to demonstrate the action times assumed in the analysis. WCNOC committed to perform additional design basis steam generator overfill simulator scenarios on a minimum of 80 percent of the current operator population by March 31, 1992 and submit the results to the NRC staff.

Attached is WCNOC's response to the staff's request that WCNOC demonstrate the simulated SGTR scenarios on a minimum of 80 percent of the current operator population. During the time period of November 29, 1991 through February 4, 1992, WCNOC performed a total of ten additional simulated SGTR overfill scenarios representing more the 90 percent of the current operator population at WCGS. The attachment to this letter provides information on operator response times from the ten additional simulated SGTR scenarios.

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The operator action times obtained from the simulated SGTR scenarios have demonstrated that the action times assumed in the analysis are realistic and are representative of the current operator population at WCGS.

If you have any questions concerning this matter, please contact me or Mr. S. G. Wideman of my staff.

Very truly yours,

BDWithers

Bart D. Withers
President and
Chief Executive Officer

BDW/jra

Attachment

cc: A. T. Howell (NRC), w/a
R. D. Martin (NRC), w/a
G. A. Pick (NRC), w/a
W. D. Reckley (NRC), w/a

Results of Additional Demonstrations of Steam Generator Tube Rupture Operator Action Times for Wolf Creek Generating Station

1.0 Introduction

Wolf Creek Nuclear Operating Corporation (WCNOC) committed to the Nuclear Regulatory Commission (NRC), by letter dated January 15, 1991 [Reference 3] to perform additional simulator runs for the Steam Generator Tube Rupture (SGTR) overfill scenario in order to demonstrate that the operator action times assumed in the SGTR submittals are realistic and achievable for Wolf Creek Generating Station (WCGS) operators.

To fulfill this commitment, WCNOC performed a total of ten simulated SGTR overfill scenarios representing more than 90 percent of its current operator population on the WCGS simulator during the period of 11/19/91 through 2/4/92. The following sections briefly describe the SGTR overfill scenario and the associated operator actions to mitigate the consequences of the event, and present the results of the simulation. These sections are prepared to be consistent with the previous response to the NRC Request for Additional Information (RAI) on operator action times [Reference 2].

2.0 SGTR Overfill Scenario Description

The worst case single failure with respect to steam generator overfill is a failure in the open position of the auxiliary feedwater (AFW) control valve on the discharge side of the motor driven AFW pump feeding the ruptured steam generator. Failure of the control valve coupled with the flow contribution from the turbine driven AFW pump can supply initial AFW flow to the ruptured steam generator to a value near 723 gpm. In addition, realizing that AFW flow is delivered as a function of steam generator pressure and that pressure decreases as a result of relief valve actuation after trip, AFW flow can increase until AFW flow to the ruptured steam generator is terminated. Isolation of the ruptured steam generator is accomplished when AFW flow is terminated to the ruptured steam generator.

2.1 Initial Conditions

The initial conditions assumed for the steam generator overfill case are detailed in Table 3-1 of Reference 4. In the analysis of the design basis overfill scenario, initial values of plant parameters are determined by adding or subtracting parameter uncertainties as appropriate to maximize the resultant overfill potential. For example, the initial steam generator level is equal to the nominal level plus error allowance to maximize steam generator water and thus result in greater potential for steam generator overfill.

The initial simulated plant conditions were set up to be as close as possible to the conditions assumed in the analysis. The key parameters that lead to the potential steam generator overfill are the break flow and the AFW flow to the ruptured steam generator. For the SGTR overfill simulation, the initial values of these parameters were manually "dialed-in" to reflect the values assumed in the analysis.

2.2 Availability of Offsite Power

The potential for steam generator overfill is not strongly dependent on the availability of offsite power. However, the potential for overfill is slightly greater if offsite power is assumed lost at reactor trip as AFW flow is initiated earlier. Furthermore, if overfill should occur, then subsequent offsite doses are greater if offsite power is assumed lost.

2.3 Operator Response to Steam Generator Overfill

The potential for overfill of the ruptured steam generator is largely negated when AFW flow to the ruptured steam generator is terminated. Control Room operators are aware of AFW control valve malfunction when the ruptured steam generator's narrow range level is increasing significantly coupled with the indication that the AFW control valve is wide-open. Isolation of the AFW flow, and thus the steam generator, is accomplished by the operator's action to either deenergize the appropriate motor driven AFW pump or isolate the AFW pump discharge valve.

3.0 Operator Responses to Mitigate an SGTR

As identified in Reference 1, there are nine operator responses which must be performed in a timely manner to mitigate the consequences of an SGTR. These responses are irrespective of the scenario assumed and are:

1) Identify the ruptured steam generator,

2) Isolate the ruptured steam generator,

3) Initiate Reactor Coolant System (RCS) cooldown,

4) Terminate RCS cooldown,

5) Initiate RCS depressurization,

6) Terminate RCS depressurization,

7) Initiate safety injection,

8) Terminate safety injection, and

9) Equalize primary and secondary pressures.

These individual operator responses have previously been extensively described in References 4 through 7. However, discussion of the responses will be reiterated below.

3.1 Identification of the Ruptured Steam Generator

Reference 7 details the numerous indications available to the control room operators to alert them to the occurrence of a SGTR. In the simulated SGTR scenarios, identification of the ruptured steam generator can occur at any time the operators can state unequivocally that a tube rupture is in progress. This determination can be made either before or while in Emergency Operating Procedure, EMG E-3, "Steam Generator Tube Rupture".

Given a reactor trip or safety injection signal as a result of an SGTR, the control room operators would enter Emergency Operating Procedure, EMG E-0, "Safety Injection", which governs their actions to verify the proper response of the automatic protection system following manual or automatic actuation of safety injection. Through symptom-based diagnosis, the operator is directed to transfer to EMG E-3 when indications are such that a tube rupture is in progress.

Since the operator is in EMG E-3, procedural guidance in Step 2 requires identification of the ruptured steam generator. This is accomplished by observing at least one of the following:

1) Unexpected rise in any steam generator's narrow range level, or

2) High turbine driven AFW pump exhaust radiation, or

3) High radiation from any steam generator steamline radiation monitor, or by

4) Steam generator blowdown samples.

Items 1 through 3 above can be observed in the control room. Item 4 allows a manual sampling of the suspected ruptured steam generator as well as a sampling of the intact steam generator blowdown lines for verification of the ruptured steam generator. Reference 7 provided information regarding the capabilities for manual sampling of the steam generator.

Table 1 provides both the time of identification and isolation of the ruptured steam generator on the same line. This is due to the fact that it is not possible to precisely document when in fact the control room operators identify that a SGTR has occurred. For example, for all of the scenarios presented, the operators were aware that a potential SGTR was in progress early in the transient. Statements made by the operators ranged from "Looks like a rupture in Steam Generator A", "We've got a rupture in Steam Generator A", to "Steam Generator A is showing signs of a tube rupture". Rather than attempt to assign an observed time value to statements like these, the identification and isolation of the ruptured steam generator was combined and then compared to the assumed response time as recorded in Reference 8.

3.2 <u>Isolation of the Ruptured Steam Generator</u>

Steps 3 and 4 of EMG E-3 requires isolation of the ruptured steam generator by performance of the following:

1) Adjusting the ruptured steam generator's Atmospheric Relief Valves (ARV) controller to a high setpoint of 1125 psig and verify it closed.

2) Close steam supply valves from ruptured steam generator to the inlet of the turbine driven AFW pump.

3) Ensure blowdown lines have isolated,

4) Close ruptured steam generator's main steamline isolation, bypass, and drain valves, and

5) Stopping AFW flow to the ruptured steam generator when narrow range level has reached 4 percent.

For the steam generator overfill scenario, the ruptured steam generator is considered isolated when its AFW is terminated. The overfill concern is negated at this point as no other feedwater enters the steam generator from this point on. As is indicated in Table 1, the steam generator is isolated, on average, about 4 minutes prior to the assumed response time of 16 minutes.

3.3 Initiation of RCS Cooldown

Cooldown is begun when the intact steam generator's ARV are opened to allow steam dump to the atmosphere. This assumes that offsite power is lost. Should offsite power be available, then steam could be dumped to the main condenser until vacuum is lost in the condenser. Table 1 shows that cooldown was initiated, on the average, about 4 minutes prior to the assumed response time.

3.4 Termination of RCS Cooldown

Cooldown is terminated upon closure of the intact ARVs and is done when the appropriate core exit thermocouples/RCS wide range temperatures are reached. Table 1 indicates that for the overfill scenario cooldown was terminated, on average, about 7 minutes prior to the assumed response time.

3.5 Initiation of RCS Depressurization

Depressurization of the RCS is initiated shortly after cooldown has terminated. EMG E-3 specifies the use of one pressurizer pilot operated relief valve (PORV) to depressurize should normal spray not be available, as is the case when offsite power is lost. Table 1 indicates that for the overfill scenario depressurization was initiated, on average, about 6 minutes ahead of the assumed response time.

3.6 Termination of RCS Depressurization

The decision to terminate depressurization is based on either the difference between RCS and ruptured steam generator pressure, pressurizer level, or the amount of the RCS subcooling. Depressurization is terminated upon closure of the pressurizer PORV. Table 1 details that for the overfill scenario, depressurization was terminated, on average, about 3 minutes prior to the assumed response time.

3.7 Initiation of Safety Injection

Safety injection (SI) is not an operator action for the analyzed SGTR scenario. SI is automatically actuated upon receipt of a low pressurizer pressure SI signal fo] wing reactor trip.

3.8 Termination of Safety Injection

Following completion of RCS depressurization, EMG E-3 requires several conditions to be met prior to termination of SI. These are: a minimum amount of RCS subcooling, a secondary heat sink via at least one intact steam generator, a minimum RCS pressure, and a stable or increasing RCS pressure. Table 1 indicates that for the overfill scenario, the control room operator response was, on the average, about 2 minutes earlier than the assumed response time.

3.9 Equalization of Primary and Secondary Pressures

The immediate situation after the termination of SI is that RCS pressure is a few hundred pounds per square inch (psi) greater than the ruptured steam generator and break flow, though reduced, still continues. The operators are required to take available actions to equalize pressures between the RCS and the ruptured steam generator to stop break flow. The pressure equalization is terminated when break flow stops or reverses into the primary system.

As indicated in Table 1, the average pressure equalization time was about five minutes longer than the assumed response time (39.9 minutes) used in the analysis. This was due to the fact that the operators, in equalizing the primary and secondary pressure, followed Step 30 in EMG E-3 to use the letdown and auxiliary spray to slowly bring the primary pressure down when the ruptured steam generator water level was offscale high rather than opening one pressurizer PORV for a quick pressure release as in the previous demonstrations [Reference 2]. Depending upon the situation, there is more than one option the operators can take in order to achieve the pressure equalization. Per EMG E-3 Step 30, when the ruptured steam generator water level was offscale high, the only option the operators can take is to use auxiliary spray to depressurize the primary system when letdown is in service. Under this option, it normally would take 10 to 15 minutes for the primary and secondary pressure to equalize compared to just a few minutes when using pressurizer PORV. During the 12/10/91 demonstration (run number 6), the simulator crew decided to use one PORV for pressure equalization. As a result, it took less than 4 minutes to equalize the pressures and stop the break flow. Using one pressurizer PORV for pressure equalization is quick but the pressure response would be sawtoothed as the operator continuously cycles the PORV. The use of auxiliary spray for pressure equalization is preferred as it gives a much smoother control of the final conditions.

It should be noted that failure to achieve pressure equalization within the assumed response time would not invalidate the previous conclusions on the radiological consequences of a design-basis SGTR overfill event. By letter dated May 15, 1987 [Reference 8], WCNOC has demonstrated that the calculated offsite radiological consequences for a forced overfill SGTR event with a stuck-open safety valve remain well within the guidance values of 10CFR100 and Standard Review Plan 15.6.3. The offsite radiological consequences calculations for this worst-case overfill scenario was continued until Residual Heat Removal System cut-in conditions were reached; i.e., three hours forty minutes after initiation of the accident.

In addition, if the break flow continued for an additional five minutes, the water volume in the ruptured steam line would increase to approximately 155 cubic feet which would fill to only 23 percent of total capacity of the steam line (i.e., 682 cubic feet). The effects of a steam line filled with water due to steam generator overfill has been addressed in the original analysis [Reference 4] which confirms that the steam line integrity will be maintained during the worst case potential overfill.

4.0 Simulator Crew Composition

A total of 46 licensed individuals participated in the ten simulator scenarios. This constitutes 92 percent of the licensed operator population at the WCGS. The simulator crews were composed of licensed individuals filling the positions of shift supervisor, supervising operator, reactor operator and balance-of-plant operator. In five cases, one additional licensed operator participated as an extra crew member/shift technical advisor. Five operating crews (25 individuals) were included. The others were made up of 21 licensed individuals from the Operations and Training organizations. The crews provided a representative cross-section of licensed operators and the results shown in Table 1, except the last response time (to equalize the primary and secondary pressures) which was discussed in the previous section, indicates the ability to respond to a SGTR scenario in a timely manner.

As part of the Licensed Operator Requalification Training Program licensed operators are required to review the emergency operating procedures on an annual basis. Additionally, the operators practice implementing the emergency operating procedures on the simulator every six weeks as part of the requalification training program.

5.0 Conclusions

The results in Table 1 indicate that, on the average, the WCGS operators response times are well within the response times assumed in the SGTR analysis with the exception of the last step (equalization of primary and secondary pressures), where the average operator response time was about five minutes longer than the response time (39.9 minutes) assumed in the transient analysis.

It should be noted that failure to achieve pressure equalization within the assumed response time is insignificant as far as the contributions to offsite release are concerned. The major contribution to offsite radiation doses is from the break flow that flashes immediately and escapes to the outside atmosphere prior to termination of the safety injection. This is due to the fact that the break flow and the fraction of break flow flashed into vapor are significantly higher during this time period. After the termination of the safety injection, the contribution due to the water-vapor mixture released through the partially open safety valve is relatively insignificant. WCNOC has demonstrated that the calculated offsite radiological consequences for a forced overfill SGTR event with a stuck-open safety valve remain well within the guidance of 10CFR 100 and Standard Review Plan 15.6.3 [Reference 8]. The offsite radiological consequences calculations for this worst-case overfill scenario was continued to about three hours forty minutes after the accident has occurred.

On the basis of the SGTR simulator scenarios performed at WCGS and the previous discussion, WCNOC believes that it has sufficiently demonstrated that the operator response times assumed in the analysis are realistic and are representatives of the current operator population at WCGS. WCNOC also recognizes that the operator response time for the last step (pressure equalization) could be a few minutes longer than the assumed response time used in the analysis depending upon the options available at the time the operator takes the action.

6.0 References

1. Letter dated July 24, 1990 from D. V. Pickett, NRC, to B. D. Withers, WCNOC, "Request For Additional Information Concerning Steam Generator Tube Rupture Operator Action Times for the Wolf Creek Generating Station".

2. Letter dated November 30, 1990 from Bart D. Withers, WCNOC, to NRC, "Response to Request for Additional Information Concerning Steam Generator Tube Rupture Operator Action Times for the Wolf Creek Generating Station".

3. Letter dated January 15, 1991 from B. D. Withers (WCNOC) to NRC, "Steam Generator Tube Rupture Operator Action Times For Wolf Creek Generating Station".

4. SLNRC 86-01, January 8, 1986, "Steam Generator Tube Rupture Analysis - SNUPPS".

5. SLNRC 86-05, April 1, 1986, "Steam Generator Tube Rupture Analysis - SNUPPS".

6. SLNRC 86-08, September 4, 1986, "Steam Generator Tube Rupture Analysis - SNUPPS".

7. WM 87-0029, February 4, 1987, "Response to RAI Regarding SGTR Analysis".

8. WM 87-0145, May 15, 1987, "Response to RAI Regarding the SGTR Analysis".

TABLE I

SGTR OVERFILL SCENARIO

OPERATOR ACTIONS	ASSUMED RESPONSE TIME (min) + / *	SIMULATOR RUN (Response Time In Minutes) (1) 11/19/91	(2) 11/19/91	(3) 12/05/91	(4) 12/05/91	(5) 12/10/91	(6) 12/10/91	(7) 12/17/91	(8) 12/17/91	(9) 1/28/92	(10) 2/4/92	AVERAGE
Tube Rupture Begins	0.00/0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Identify/Isolate Ruptured SG	16.0/16.0	11.45	9.15	14.05	17.03	12.25	13.55	10.47	9.67	9.25	9.75	11.66
Initiate Cooldown	24.0/24.0	17.00	21.43	21.58	21.28	18.25	22.48	21.47	17.90	13.50	22.75	19.76
Terminate Cooldown	32.9/32.3	23.30	24.43	25.10	27.78	24.75	29.30	27.38	23.17	20.25	28.25	25.37
Initiate Depressurization	33.9/33.3	25.40	26.93	32.20	29.28	27.42	31.15	28.50	24.42	21.25	30.00	27.66
Terminate Depressurization	35.2/34.7	29.45	30.46	35.30	34.28	30.42	36.30	33.00	27.50	25.75	35.75	31.82
Terminate SI	36.2/47.3	32.00	32.13	36.80	35.03	31.42	40.45	35.33	31.20	28.00	36.50	33.89
Pressure Equalization	39.9/ **	42.30	42.85	46.80	48.53	46.18	44.00***	42.00	40.16	41.28	48.00	44.21

+ The response times assumed in the transient analysis for SGTR overfill case.

* The response times assumed in the analysis for a forced overfill SGTR with stuck-open safety valve, which the calculation of radiological consequences of the worst case overfill SGTR scenario was based upon.

** RHR cut-in conditions (219.9 minutes).

*** Operator chose to open 1 PORV.