NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES 8/31/88

# LICENSEE EVENT REPORT (LER)

| FACILITY NAME (1) | DOCKET NUMBER (2) | PAGE (3) |
|---|---|---|
| Sequoyah, Unit 2 | 0 5 0 0 0 3 2 8 | 1 OF 0 7 |

TITLE (4) The Failure To Identify The Effects Of Excessive Post-Trip Reactor Coolant System Cooldowns Could Have Caused Noncompliance With Shutdown Margin Requirements

| EVENT DATE (5) MONTH | DAY | YEAR | LER NUMBER (6) YEAR | SEQUENTIAL NUMBER | REVISION NUMBER | REPORT DATE (7) MONTH | DAY | YEAR | OTHER FACILITIES INVOLVED (8) FACILITY NAMES | DOCKET NUMBER(S) |
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| 0 6 | 2 1 | 8 8 | 8 8 | — 0 3 0 — | 0 1 | 0 9 | 0 8 | 8 8 | Sequoyah, Unit 1 | 0 5 0 0 0 3 2 7 |
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THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

OPERATING MODE (9): 2
POWER LEVEL (10): 0 0 2

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| 20.402(b) | 20.405(c) | 50.73(a)(2)(iv) | 73.71(b) |
| 20.405(a)(1)(i) | 50.36(c)(1) | 50.73(a)(2)(v) | 73.71(c) |
| 20.405(a)(1)(ii) | 50.36(c)(2) | 50.73(a)(2)(vii) | OTHER (Specify in Abstract below and in Text, NRC Form 366A) |
| 20.405(a)(1)(iii) | XX 50.73(a)(2)(i) | 50.73(a)(2)(viii)(A) | |
| 20.405(a)(1)(iv) | 50.73(a)(2)(ii) | 50.73(a)(2)(viii)(B) | |
| 20.405(a)(1)(v) | 50.73(a)(2)(iii) | 50.73(a)(2)(x) | |

LICENSEE CONTACT FOR THIS LER (12)

| NAME | TELEPHONE NUMBER (AREA CODE) |
|---|---|
| J. D. Smith, Plant Operations Review Staff | 6 1 5 8 7 0 - 6 6 7 2 |

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

| CAUSE | SYSTEM | COMPONENT | MANUFACTURER | REPORTABLE TO NPRDS | CAUSE | SYSTEM | COMPONENT | MANUFACTURER | REPORTABLE TO NPRDS |
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SUPPLEMENTAL REPORT EXPECTED (14)

| YES (If yes, complete EXPECTED SUBMISSION DATE) | XX NO | EXPECTED SUBMISSION DATE (15) | MONTH | DAY | YEAR |
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ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)


This LER is being revised to provide supplemental information on corrective action.

On June 21, 1988, with unit 2 in startup (mode 2), Sequoyah Nuclear Plant (SQN) personnel documented the potential for a loss of required shutdown margin (SDM) following an end-of-cycle reactor trip and subsequent reactor coolant system (RCS) cooldown below the nominal no-load average RCS temperature (T-avg) of 547 degrees F. SDM calculations performed following reactor trips have typically been performed approximately 30 to 60 minutes following the trip and, moreover, have been based on either the RCS temperature at that time or on the minimum planned RCS temperature for the specific SDM surveillance interval. As a result, these SDM calculations did not necessarily consider the impact of the RCS cooldown immediately following the trip.
The condition was the result of a failure to identify the effects of excessive post reactor trip cooldown on other plant parameters. The excessive cooldowns and subsequent addition of positive reactivity from the negative moderator temperature coefficient have been attributed to the overall secondary system response to a reactor trip. TVA, in conjunction with Westinghouse (the nuclear steam supply system vendor), has back-calculated the minimum SDM during past operation and verified that the Technical Specification SDM requirements have been satisfied at all times during cycle 3 operation of units 1 and 2. As immediate corrective action, TVA revised the emergency instruction used to guide operator action following a reactor trip to include a provision requiring plant operators to manually control auxiliary feedwater flow (AFW) if the RCS cooldowns become excessive. Subsequent corrective actions have deleted requirements for manual AFW control while requiring operator action to borate the RCS based upon Tavg.


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DESCRIPTION OF EVENT

This LER is being revised to provide supplemental information on corrective actions.

On June 21, 1988, with unit 2 in mode 2 (2 percent power, 2240 psig, 548 degrees F), Potential Reportable Occurrence (PRO) 2-88-178 was written to formally evaluate the potential reportability of Condition Adverse to Quality Report (CAQR) SQP 880375. The subject CAQR, which was initiated on June 14, 1988, documented the potential for a loss of required shutdown margin (SDM) following an end-of-cycle (EOC) reactor trip and subsequent reactor coolant system (RCS) (EIIS Code AB) cooldown below the nominal no-load average RCS temperature (T-avg) of 547 degrees F.

On June 13, 1988, personnel from the Technical Support Group at Sequoyah Nuclear Plant (SQN) were reviewing SDM requirements associated with post-trip RCS cooldowns (reference LERs SQRO-50-328/88023, 88024, 88027, and 88028). SDM calculations performed following reactor trips are typically performed 30 to 60 minutes following the trip and, moreover, are based on either the RCS temperature at that time or the minimum planned RCS temperature for the specific SDM surveillance interval. That is, the post-trip SDM calculations were typically based on an RCS temperature near the no-load T-avg of 547 degrees F and did not necessarily consider the impact of the RCS cooldown immediately following the reactor trip. For example, following the June 6, 1988 reactor trip from full power conditions (reference LER SQRO-50-328/88027), the SDM calculation was performed approximately 60 minutes after the trip when the RCS temperature had recovered to the no-load T-avg of approximately 547 degrees F. Following a review of the post-trip data, it was determined that the average RCS temperature (i.e., the average of the T-avg's from each of the four RCS loops) had actually decreased to approximately 527 degrees F before it began to recover. It was subsequently determined that if an SDM calculation had been performed at an RCS temperature of 527 degrees F (i.e., the minimum post-trip RCS temperature), the 1.6 percent delta k/k (1600 pcm) SDM requirement of Technical Specification (TS) 3.1.1.1 may not have been satisfied. Upon determining that SDM requirements may have been violated, Technical Support personnel contacted the nuclear steam supply system (NSSS) vendor (Westinghouse) and initiated an investigation to determine the actual value of SDM at the minimum RCS temperature following the June 6, 1988 reactor trip.

Following a review of post-trip RCS data by Westinghouse, it was determined that adequate SDM had existed at all times following the June 6, 1988 reactor trip. This calculation was performed by Westinghouse and documented in a transmittal to TVA dated June 17, 1988. Following the determination that adequate SDM had existed after the June 6, 1988 reactor trip, Westinghouse subsequently recalculated the minimum SDM available during all previous reactor trips that had occurred on unit 2 since restart.

Under all conditions, it was determined that adequate SDM had been available; however, because unit 2 was approaching end-of-cycle (EOC) conditions, it was also determined that SDM requirements would become more difficult to satisfy.

On June 17, 1988, TVA received a transmittal from Westinghouse that provided guidelines for ensuring adequate SDM following reactor trips from 70 percent power (or less) and subsequent RCS cooldowns to approximately 520 degrees F. Based on the subject guidelines, TVA calculated RCS boration volumes that are required in the event that the plant cooled to below 520 degrees F following a trip. These boration requirements were subsequently incorporated into Emergency Instruction ES-0.1, "Reactor Trip Response." On June 19, 1988, following the revision to ES-0.1, unit 2 was restarted and subsequently maintained at 70 percent power. On June 29, 1988, TVA received a transmittal from Westinghouse that provided additional guidelines for the maintenance of SDM requirements following reactor trips from 70, 80, 90, and 100 percent power and have incorporated these guidelines into ES-0.1 to allow unit 2 operation at greater than 70 percent power.

On June 30, 1988 Westinghouse confirmed that the SQN unit 2, cycle 3 all rods in (ARI) boron concentrations that had previously been transmitted to TVA were nonconservative because they did not contain an allowance for power redistribution effects. TVA had previously identified an inconsistency in the ARI boron concentrations for unit 2, cycle 3, and had requested Westinghouse to verify the adequacy of the data. Following confirmation that the ARI data were nonconservative, TVA initiated CAQR SQP880401 to document and track the issue. These ARI boron concentrations are incorporated in Technical Instruction (TI)-22, "Shutdown Margin Calculations," and are used whenever the unit is in mode 4 or 5 (and the residual heat removal (RHR) (EIIS Code BP) system is in operation) to provide protection against a postulated boron dilution event. The relationship between the neutron multiplication factor (k-eff) and the ARI boron concentration must be such that it allows plant operators at least 15 minutes from the initiation of an inadvertent boron dilution accident to take appropriate action (i.e., stop the dilution) before the reactor goes critical. Westinghouse subsequently provided a new set of ARI boron concentrations to ensure unit 2 had adequate protection against a boron dilution accident.

No immediate operator action was required for unit 2 as a result of the nonconservative ARI boron concentration because the unit was operating in mode 1 at approximately 70 percent power. However, Technical Support personnel immediately initiated a search of past unit 2 operation (in the applicable modes) to determine if the subject tables had been used to justify plant operation at a relatively low boron concentration. The results of this search indicated that during unit 2 operation in modes 4 and 5, adequate boron concentrations existed to ensure the 15-minute criteria for operator action was satisfied.

CAUSE OF EVENT

The immediate cause of the potential loss of SDM is the excessive cooldown of the RCS following a reactor trip and the associated addition of positive reactivity from the negative moderator temperature coefficient (MTC). These excessive post-trip cooldowns have been determined to be the result of the overall secondary system response to a reactor trip.

Following a reactor trip, the steam dump system is automatically actuated in an attempt to restore the primary system temperature to the no-load T-avg of 547 degrees F. The steam dump system can be controlled during steady-state operation in either a steam pressure mode or a T-avg mode, although typically the steam pressure mode is used only during plant startups and cooldowns. If the steam dump system is in the T-avg mode and a reactor trip occurs, the system controller will compare a T-avg signal with the no-load T-avg of 547 degrees F. Since the T-avg signal will initially be greater than 547 degrees F, a signal will be sent to open selected sets of steam dump valves to remove stored energy and decay heat to return T-avg to no-load conditions. The steam dump system does not anticipate the rapid RCS cooldown that can occur when the auxiliary feedwater (AFW) system (EIIS Code BA) begins to fill the steam generators with relatively cold water.

The AFW system automatically actuates following the trip of the main feedwater pumps to ensure the steam generators have an adequate supply of water for primary system heat removal. As the steam generator water levels decrease below approximately 33 percent narrow-range (NR) level (initially from the collapsing voids and operation of the steam dump system, and secondarily, from the lack of main feedwater), AFW from the two motor-driven AFW pumps (each with a 440 gallon per minute (GPM) capacity) and from the turbine-driven AFW pump (880 GPM capacity) begins to enter the steam generators to automatically maintain levels to 33 percent. Approximately 440 GPM of AFW flow is required to ensure that the primary system has an adequate heat sink. The AFW flow, which is supplied from the condensate storage tank (CST) and is therefore much colder than main feedwater, can cause a rapid decrease in RCS temperature. Thus, the combination of steam dump operation and AFW flow rates from the CST have resulted in post-trip RCS cooldowns below the designed no-load T-avg of 547 degrees F.

The root cause of this event was failure of TVA to identify the effects of excessive post reactor trip RCS cooldown on other plant parameters. The Westinghouse core design for SQN units 1 and 2 did not account for more than a nominal 2- to 4-degree cooldown (from no-load T-avg conditions) following a reactor trip. However, following a review of post-trip data, it was determined that actual RCS cooldowns were approximately 20 to 30 degrees below the nominal no-load T-avg conditions.

The root cause of the nonconservative ARI boron concentrations contained in TI-22 has been attributed to a personnel error by Westinghouse. The Westinghouse method for calculating ARI boron concentrations required hand manipulation of several sets of data. Among these data sets are corrections to account for the effects of power redistribution in the core. During the manipulation of the subject data, Westinghouse personnel inadvertently omitted the data set used to account for power redistribution effects. As a result, the ARI boron concentrations transmitted to TVA and incorporated into TI-22 were nonconservative.

ANALYSIS OF EVENT

The potential for a loss of SDM following a reactor trip and subsequent RCS cooldown and the nonconservative ARI boron concentrations in TI-22 did not result in a violation of the SQN TS. However, these conditions are being reported in accordance with 10 CFR 50.73, paragraph a.2.i.b, as a condition that could have resulted in noncompliance with TS 3.1.1.1. TVA has also evaluated this event with respect to potential reportability under 10 CFR 21; however, because the above described conditions did not result in a significant safety hazard, TVA does not believe this event satisfies the criteria of 10 CFR 21.

The design of the SQN core includes 1600 pcm of SDM to ensure (1) the reactor core can be made subcritical from all operating conditions, (2) reactivity transients associated with possible accident conditions are controllable within acceptable limits, and (3) the reactor will be maintained sufficiently subcritical to allow time for the operator action necessary to prevent inadvertent criticality from a shutdown condition. The most restrictive conditions with respect to SDM occur at the end of core life and are associated with an MSLB accident from no-load conditions. An MSLB initiated from no-load conditions is more severe than an MSLB initiated from power because the initial steam generator water mass is greatest at no-load conditions; hence, the magnitude and duration of the RCS cooldown are more severe.

Westinghouse has back-calculated the minimum SDM for the more severe reactor trips (i.e., those trips that may have approached the SDM limit) for both unit 1 and unit 2 during cycle 3 operation. In general, only cycle 3 trips were investigated because in previous core designs, control and shutdown rods had higher reactivity worths and the reactor core had more SDM (over and above the 1600 pcm SDM required by TS 3.1.1.1). The results of this investigation verified that over 1600 pcm of SDM had been available at all times during cycle 3 operation of units 1 and 2.

Technical Support personnel have investigated previous unit 2 operation in modes 4 and mode 5 (unit 1 was unaffected) while the unit was on RHR to determine if the noncorservative ARI boron concentrations supplied by Westinghouse and incorporated into TI-22 could have caused a plant condition in which a postulated boron dilution accident may have resulted in the reactor core attaining criticality before operator action could be credited. The results of this investigation verified that SQN operators had sufficient time (i.e., greater than 15 minutes) to take the actions necessary to stop a postulated boron dilution accident before criticality occurred. That is, the actual RCS boron concentrations were higher than those required by TI-22, even when the TI-22 values were corrected to include the effects of power redistribution. Thus, since adequate SDM was available at all times and since actual RCS boron concentrations in unit 2 ensured that adequate time was available to mitigate a potential boron dilution accident, there were no safety consequences associated with this event.

CORRECTIVE ACTION

As immediate corrective action (taken on June 18, 1988, before unit 2 reentered mode 2 conditions), TVA revised ES-0.1 to require plant operators to manually control AFW flow whenever the RCS temperature decreases to less than 530 degrees F in an attempt to maintain temperature above 520 degrees F. This revision to ES-0.1 also requires plant operators to initiate an RCS boration if the RCS temperature decreases below 520 degrees F. To ensure the RCS is sufficiently borated, a table was incorporated into ES-0.1 that provides the specific boration volumes necessary to maintain adequate SDM to temperatures as low as 490 degrees F (in 5 degree F increments). As a result of this action, SQN management allowed unit 2 to restart and increase power to a nominal 70 percent.

To allow unit operation above 70 percent power, ES-0.1 was again revised (revision 4) to provide operator action to initiate RCS boration as necessary based upon post-trip RCS temperature. This revision deletes the previous ES-0.1 requirement to take manual control of the AFW system and, although post-trip cooldowns may continue until further corrections are made, ensures adequate SDM will be available in the event of a post-trip RCS cooldown.

In addition, a corrective action plan has been developed to pursue long term corrective actions. Options currently being considered are: (1) Operation of the Steam Dump Control System in Pressure mode, (2) Implementing a modification to optimize Steam Dump Control System in T-avg mode and/or (3) Implementation manual control of the AFW system to limit the post-trip cooldown. Also, evaluations are being made on the thermal effects of the cooldown on the Reactor Vessel. Following completion of the review of these options/evaluations, TVA will determine the appropriate long-term resolution. A plan for this long term resolution will be submitted as a follow-up to TVA's August 31, 1988 letter to NRC addressing this event.

| FACILITY NAME (1) | DOCKET NUMBER (2) | LER NUMBER (6) YEAR | SEQUENTIAL NUMBER | REVISION NUMBER | PAGE (3) |
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| Sequoyah, Unit 2 | 0 5 0 0 0 3 2 8 | 8 8 | — 0 3 0 | — 0 1 | 0 7 OF 0 7 |

TEXT (If more space is required, use additional NRC Form 366A's) (17)

To ensure unit 2 can operate in modes 4 and 5 at boron concentrations less than the refueling boron concentrations and still provide adequate protection against a postulated boron dilution accident, TVA implemented Instruction Change Form (ICF)-88-0918 on July 1, 1988. This ICF completely replaced the nonconservative ARI boron concentration in TI-22 with newly calculated ARI boron concentrations that included an allowance for power redistribution effects. Thus, plant operators will have at least 15 minutes from the initiation of an inadvertent dilution to take appropriate actions before the reactor core attains criticality.

ADDITIONAL INFORMATION

There have been no previously reported events that resulted from the failure to identify plant characteristics during the core design process.

COMMITMENTS

None.

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TENNESSEE VALLEY AUTHORITY
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September 8, 1988

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

Gentlemen:

TENNESSEE VALLEY AUTHORITY - SEQUOYAH NUCLEAR PLANT UNIT 2 - DOCKET NO. 50-328 - FACILITY OPERATING LICENSE DPR-79 - REPORTABLE OCCURRENCE REPORT SQRO-50-328/88030 REVISION 1

The enclosed licensee event report has been revised to provide additional information regarding the corrective action TVA is considering to prevent excessive post-trip RCS cooldowns. These cooldowns could have caused noncompliance with shutdown margin requirements. This event was originally reported in accordance with 10 CFR 50.73, paragraph a.2.i.b on July 14, 1988.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

S. J. Smith
Plant Manager

Enclosure
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