

LICENSEE EVENT REPORT (LER)

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

Title (4) The Potential Exists For Unacceptable Dilution Of The ECCS During A Large Break LOCA Due To Postulated Non-Borated Water Addition To Containment Sump

| EVENT DATE (5) | | | LER NUMBER (6) | | | REPORT DATE (7) | | | OTHER FACILITIES INVOLVED (8) | | | | | | | | | |
|----------------|-----|------|----------------|-------------------|-----------------|-----------------|-----|------|-------------------------------|---|------------------|---|---|---|---|---|------------------|-------------------------------|
| MONTH | DAY | YEAR | YEAR | SEQUENTIAL NUMBER | REVISION NUMBER | MONTH | DAY | YEAR | FACILITY NAMES | | DOCKET NUMBER(S) | | | | | | | |
| 0 | 7 | 2 | 2 | 8 | 7 | 0 | 4 | 6 | 0 | 1 | 0 | 3 | 0 | 5 | 8 | 8 | Sequoyah, Unit 2 | 0 5 0 0 0 3 2 8 |
| 0 | 7 | 2 | 2 | 8 | 7 | 0 | 4 | 6 | 0 | 1 | 0 | 3 | 0 | 5 | 8 | 8 | | 0 5 0 0 0 |

OPERATING MODE (9) 5

POWER LEVEL (10) 0 | 0 | 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR § (Check one or more of the following) (11)

| | | | |
|-------------------|------------------|---------------------|-------------------------------------------------------------------------------------------------|
| 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)(vi) | <input checked="" type="checkbox"/> OTHER (Specify in Abstract flow and in Text, NRC Form 366A) |
| 20.405(a)(1)(iii) | 50.73(a)(2)(i) | 50.73(a)(2)(vii)(A) | |
| 20.405(a)(1)(iv) | 50.73(a)(2)(ii) | 50.73(a)(2)(vii)(B) | |
| 20.405(a)(1)(v) | 50.73(a)(2)(iii) | 50.73(a)(2)(ix) | |

LICENSEE CONTACT FOR THIS LER (12)

| NAME | TELEPHONE NUMBER |
|--------------------------------------------|-----------------------------------------------|
| K. E. Meade, Plant Operations Review Staff | 6 1 5 8 7 0 - 1 6 2 5 0 |

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

| CAUSE | SYSTEM | COMPONENT | MANUFACTURER | REPORTABLE TO NRC | CAUSE | SYSTEM | COMPONENT | MANUFACTURER | REPORTABLE TO NRC |
|-------|--------|-----------|--------------|-------------------|-------|--------|-----------|--------------|-------------------|
| | | | | | | | | | |
| | | | | | | | | | |
| | | | | | | | | | |

SUPPLEMENTAL REPORT EXPECTED (14)

YES (If, yes, complete EXPECTED SUBMISSION DATE) NO

EXPECTED SUBMISSION DATE (15)

| MONTH | DAY | YEAR |
|-------|-----|------|
| | | |

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

This LER is being revised to inform NRC that TVA has completed its safety evaluation relating to the potential dilution of emergency core cooling system (ECCS) recirculation flow following a postulated large break loss of coolant accident (LOCA).

This condition is being voluntarily reported to inform NRC of a potential issue that has generic implications to all Westinghouse pressurized water reactors. TVA received a technical bulletin from Westinghouse concerning the post-LOCA boron concentration requirements during the long-term cooling mode of operation with all control rods out (ARO). The bulletin states that the recent trend towards longer reload cycle lengths and the introduction of a positive moderator temperature coefficient at lower power levels have resulted in the need to reconfirm that the reactor core remains subcritical during the post-LOCA, ARO, long-term cooling mode of operation. The concern is whether there is an acceptable boron concentration in the emergency core cooling system to maintain the reactor core subcritical during a large break LOCA with the addition of nonborated water to the primary containment sump. The essential raw cooling water, high pressure fire protection, primary water, and component cooling water piping systems are potential sources for nonborated water. An evaluation has been performed for the above described condition which concluded there are no safety concerns associated with the subject scenario.

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LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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| FACILITY NAME (1) Sequoyah, Unit 1 | DOCKET NUMBER (2) 0 5 0 0 0 3 2 7 8 7 | LER NUMBER (6) | | | PAGE (3) | | |
| | | YEAR | SEQUENTIAL NUMBER | REVISION NUMBER | | | |
| | | - | 0 4 6 | - | 0 1 | 0 2 | OF 0 3 |

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

DESCRIPTION OF CONDITION

This LER is being revised to inform NRC that TVA has completed its safety evaluation relating to the potential dilution of emergency core cooling system (ECCS) recirculation flow following a postulated large break loss of coolant accident (LOCA).

This condition is being voluntarily reported to inform NRC of a potential issue that has generic implications to all Westinghouse pressurized water reactors.

Westinghouse technical bulletin NSID-TB-86-08, "Post-LOCA Long-Term Cooling: Boron Requirements," described a potential safety concern relating to the post-LOCA boron requirements during the long-term cooling mode of operation with all control rods out (ARO). The bulletin states that the recent trend to longer reload cycle lengths and the introduction of a positive moderator temperature coefficient at lower power levels have resulted in the need to reconfirm that the reactor core remains subcritical for a post-LOCA, ARO, long-term cooling event. The boron concentration in the ECCS is required to remain at an adequate level to maintain the reactor core subcritical during a post-LOCA, ARO, long-term cooling condition. There is some concern that the boron concentration in the ECCS could be diluted by nonborated water entering the primary containment sump. The essential raw cooling water (EIIS Code BI), high pressure fire protection (EIIS Code KP), primary water (EIIS Code KJ), and the component cooling system (EIIS Code CC) are potential sources of nonborated water.

This potential for dilution of the ECCS applies to all Westinghouse pressurized water reactors (PWRs). It should be noted that Westinghouse believes that the ECCS will maintain an adequate boron concentration during a post-LOCA event such that the reactor core will remain subcritical. This issue is being tracked at Sequoyah Nuclear Plant (SQN) under Significant Condition Report (SCR) SQNMEB86120.

ANALYSIS OF CONDITION

The Westinghouse technical bulletin indicates that there is a need to reconfirm that the reactor core will remain subcritical during a post-LOCA, ARO, long-term cooling mode of operation due to trends to longer reload cycle lengths and the introduction of a positive moderator temperature coefficient at lower power levels. They recommend that an evaluation be made of all potential water sources (borated and nonborated) to the primary containment sump and a mass average boron concentration be determined. The mass average boron concentration must be greater than that required for the post-LOCA, ARO, long-term cooling boron concentration in order to keep the reactor core subcritical for post-LOCA, ARO, long-term cooling mode of operation.

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| | | YEAR | SEQUENTIAL NUMBER | REVISION NUMBER | | |
| | | | | | 0 3 | OF 0 3 |

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

CORRECTIVE ACTION

A safety evaluation has been performed for units 1 and 2 of the Sequoyah Nuclear Plant (SQN) to determine the safety implications the current plant configuration would have on a post-LOCA, ARO, long-term cooling condition. The evaluation identified the worst-case scenario and determined if the plant was adversely affected. The evaluation confirmed that the post-LOCA ECCS boron concentration would be adequate to ensure that the reactor core for SQN units 1 and 2 would remain subcritical during the worst-case scenario.

0521Q

TENNESSEE VALLEY AUTHORITY

Sequoyah Nuclear Plant
Post Office Box 2000
Soddy-Daisy, Tennessee 37379

March 5, 1988

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

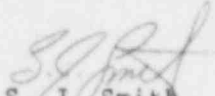
Gentlemen:

TENNESSEE VALLEY AUTHORITY - SEQUOYAH NUCLEAR PLANT UNIT 1 - DOCKET NO.
50-327 - FACILITY OPERATING LICENSE DPR-77 - REPORTABLE OCCURRENCE REPORT
SQRO-50-327/87046 REVISION 1

The enclosed licensee event report has been revised to provide additional information relating to TVA's safety evaluation of the potential for diluting the emergency core cooling system recirculation flow following a large break loss of coolant accident. This event was originally reported on July 22, 1987, for information only.

Very truly yours,

TENNESSEE VALLEY AUTHORITY


S. J. Smith
Plant Manager

Enclosure
cc (Enclosure):

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NRC Inspector, Sequoyah Nuclear Plant

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