NRC FORM 464 Part I
(6-1998)

U.S. NUCLEAR REGULATORY COMMISSION

RESPONSE TO FREEDOM OF INFORMATION ACT (FOIA) / PRIVACY ACT (PA) REQUEST

FOIA/PA: 2000-0234

RESPONSE NUMBER: 2

RESPONSE TYPE: ☑ FINAL ☐ PARTIAL

REQUESTER: Ms. Theresa Sutter

DATE: AUG 24 2000

PART I. -- INFORMATION RELEASED

☐ No additional agency records subject to the request have been located.

☐ Requested records are available through another public distribution program. See Comments section.

☑ APPENDICES C — Agency records subject to the request that are identified in the listed appendices are already available for public inspection and copying at the NRC Public Document Room.

☑ APPENDICES D — Agency records subject to the request that are identified in the listed appendices are being made available for public inspection and copying at the NRC Public Document Room.

☐ Enclosed is information on how you may obtain access to and the charges for copying records located at the NRC Public Document Room, 2120 L Street, NW, Washington, DC.

☑ APPENDICES D — Agency records subject to the request are enclosed.

☐ Records subject to the request that contain information originated by or of interest to another Federal agency have been referred to that agency (see comments section) for a disclosure determination and direct response to you.

☐ We are continuing to process your request.

☐ See Comments.

PART I.A -- FEES

AMOUNT * $ 279.60

* See comments for details

☑ You will be billed by NRC for the amount listed.

☐ You will receive a refund for the amount listed.

☐ None. Minimum fee threshold not met.

☐ Fees waived.

PART I.B -- INFORMATION NOT LOCATED OR WITHHELD FROM DISCLOSURE

☐ No agency records subject to the request have been located.

☐ Certain information in the requested records is being withheld from disclosure pursuant to the exemptions described in and for the reasons stated in Part II.

☐ This determination may be appealed within 30 days by writing to the FOIA/PA Officer, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001. Clearly state on the envelope and in the letter that it is a "FOIA/PA Appeal."

PART I.C COMMENTS (Use attached Comments continuation page if required)

Copies of Appendix D records are enclosed. The records identified on Appendix C with a ML accession number are publicly available in the NRC's Public Electronic Reading Room at http://www/mrc/gpv/NRC/ADAMS/index.html. If you need assistance in obtaining these records, please contact the PDR at (202)634-3273, or 1-800-397-4209, or by e-mail to pdr@nrc.gov.

The actual fees for the processing of your request, which you agreed to pay are noted below:

Professional Search	-	**5 hrs. @ $39.00 per hr.**	=	**$195.00**
Professional Review	-	**2 hrs. @ $39.00 per hr.**	=	**78.00**
Duplication	-	**33 pgs. @ $0.20 per pg.**	=	**6.60**
Total				**$279.60**

SIGNATURE - FREEDOM OF INFORMATION ACT AND PRIVACY ACT OFFICER

Mary Jean Bradford
Carol Ann Reed

APPENDIX C
RECORD IN ADAMS
RECORDS BEING RELEASED IN THEIR ENTIRETY
(If copyrighted identify with *)

NO.	DATE	ACCESSION #	DESCRIPTION/(PAGE COUNT)
1.	09/24/92	ML003724974	Task Interface Agreement (TIA 92-03) Concerning Crack in Oconee Decay Heat Removal (DHR) Drop Line (TAC NO. M83247) (3 pages)
2.	11/24/92	ML003725040	Close Out of Task Interface Agreement (TIA) TIA 92-28, Turkey Point Unit 4 Restart Following Hurricane i' , Kidrew (TAC NO. M84370 & M84371) w/attachments (49 pages) **(PACKAGE #ML003725023)**
3.	10/28/97	ML003725279	NRR Response to TIA 94-021, Sequoyah Nuclear Plant, Units 1 and 2 - Offsite Power Technical Specifications (TAC Nos. M93319 & M93320) (5 pages)
4.	11/06/96	ML003725301	Memo to E. W. Merschoff from H. N. Berkow; re: Catawba Nuclear Station - TIA 95-10, Standby Nuclear Service Water Pond Analysis Model (TAC M95256 and M95257) (18 pages)
5.	06/07/96	ML003725362	Memo to E. W. Merschoff from F. J. Hebdon; re: TIA 96-001, Request for Review Assistance of Sequoyah JCO for Potential Degradation of ECCS Throttle Valves During a LOCA (TAC NOS. M94780 and M94781) w/attachment (4 pages)
6.	03/11/98	ML003725492	Memo to L. Plisco from H. N. Berkow; re: Catawba Nuclear Station - Response

			to TIA 97-14, Frequency Requirements for Quality Assurance Audits (TAC NOS. M98929 and M98930) w/attachment (3 pages)
7.	06/04/97	ML003725479	Memo to F. J. Hebdon from J. R. Johnson; re: TIA 97-015 Request for Review Assistance - Maintenance Rule Implementation for Browns Ferry, Unit 1 w/enclosures (12 pages)
8.	03/31/99	ML003725472	Memo to L. R. Plisco from C. O. Thomas; re: Response to Technical Assistance (TIA 97-015) Regarding the Implementation of 10 CFR 50.65 - Browns Ferry Nuclear Plant Unit 1 (TAC NO. M98931) (2 pages)
9.	12/11/98	ML003725509	Memo to L. R. Plisco from F. J. Hebdon; re: Task Interface Agreement (TIA 98-003) Crystal River Unit 3 Low Pressure Injection System Valve Configuration (TAC NO. MA2125) w/attachment (4 pages)

APPENDIX D

RECORDS BEING RELEASED IN THEIR ENTIRETY
(If copyrighted identify with*)

NUMBER	DATE	DESCRIPTION/PAGES
1.	8/11/95	Memo to J. Zwolinski from E. Merschoff, subject: Request for Technical Assistance - Sequoyah Offsite Power Technical Specifications, (3 pgs.).
2.	2/12/96	Memo to F. Hebdon from E. Merschoff, subject: Request for Review Assistance of Sequoyah JCO for Potential Degradation of ECCS Throttle Valves During a LOCA, (20 pgs.).
3.	8/6/96	Memo to S. Varga from R. Cooper, subject: Proposed Task Interface Agreement Regarding Oyster Creek Dry Fuel Movement with the Plant in Cold Shutdown, (3 pgs.).

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W., SUITE 2900
ATLANTA, GEORGIA 30323-0199

August 11, 1995

MEMORANDUM TO: John A. Zwolinski, Deputy Director
Division of Reactor Projects, I/II
Office of Nuclear Reactor Regulation

FROM: Ellis W. Merschoff, Director
Division of Reactor Project

SUBJECT: REQUEST FOR TECHNICAL ASSISTANCE - SEQUOYAH OFFSITE POWER TECHNICAL SPECIFICATIONS (TIA 94-021)

Sequoyah Unit 1 is connected to the 500 KV switchyard and Unit 2 is connected to the 161 KV Switchyard. The two switchyards electrically are connected by a 500 KV to 161 KV intertie transformer bank. Preferred offsite power for both Sequoyah units is supplied via Common Station Service Transformers (CSSTs) from the 161 KV switchyard.

During an inspection at the Sequoyah facility, documented by NRC Inspection Report 50-327,328/93-02, concerns were identified regarding the adequacy of the 161 KV offsite power grid voltage when the Sequoyah 500 KV to 161 KV intertie transformer was not available.

Based on these concerns, the licensee agreed to enter TS LCO 3.8.1.1, Action C, and to assure grid stability with good lines of communication between the plant and dispatcher, when the intertie transformer is out of service. This action is noted in Sequoyah FSAR Revision 11, page 8.2-21. The licensee also agreed to submit to the NRC their understanding of TS restrictions associated with the intertie transformer along with their current grid load study and design calculations associated with the CSST modification. Sequoyah has replaced the original CSSTs with new CSSTs equipped with automatic load tap changers.

The region is satisfied with the licensee's actions regarding the offsite power technical specification interpretation as described in the FSAR but believes that the Sequoyah Technical Specifications are not conservative relative to operation with the intertie transformer bank out of service. The Office of Nuclear Reactive Regulation (NRR) should review this technical specification for possible change.

The licensee has performed an analysis which demonstrates that the 161 KV grid remains a reliable offsite power supply to ensure safe shutdown of the Sequoyah units in the event of loss of the intertie transformer bank. The region requests NRR review of the licensee's analyses. The documents provided by the licensee have been listed in the attachment and are provided as enclosures to this Request for Technical Assistance.

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Items requested for review include the following:

1. Based on the new Transmission System Study (Enclosures 2 and 4) and the new Common Station Service Transformers (Enclosures 5 and 6), does the plant have an acceptable immediate preferred offsite power source if the 500 KV to 161 KV Intertie Transformer Bank is not operable? Does the 161 KV analysis demonstrate that the plant can achieve safe shutdown without the intertie transformer bank?

2. Should the plant's TS be amended to require that LCO 3.8.1.1, Action C, be entered following a loss of the Intertie Transformer Bank?

These issues were discussed between D. LaBarge, NRR and M. Shymlock and G. MacDonald of Region II. If additional information is required, please contact G. MacDonald at (404) 331-5576 or M. Shymlock at (404) 331-5596.

Attachment: List of Enclosed Documents

cc w/o att: S. Vias, DRP/RII
R. Cooper, DRP/RI
W. Axelson, DRP/RIII
J. Dyer, DRP/RIV
K. Perkins, RIV/WCFO
B. Holland, RII/RI
S. Sparks, DRP/RII

DOCUMENTS PROVIDED FOR REVIEW

ENCLOSURE	DESCRIPTION
1.	PLANT VOLTAGE SCHEDULE (SWYD-18, REVISION 7)
2.	TVA ENGINEERING CALCULATION - OFFSITE POWER SUPPLY - (E31930907200)
3.	TVA MEMO - SEQUOYAH NUCLEAR PLANT 161 AND 500 KV GRID VOLTAGE SCHEDULES AND OPERATING INSTRUCTIONS (7-30-93)-(E31930730 230)
4.	TVA MEMO - TRANSMISSION SYSTEM STUDY - SEQUOYAH NUCLEAR PLANT REVISION OF THE GRID VOLTAGE SCHEDULES AND OPERATING INSTRUCTIONS - (E31930730231)
5.	TVA ENGINEERING CALCULATION - AUXILIARY POWER SYSTEM ANALYSIS SQN-EEB-MS-TI06-0002
6.	TVA ENGINEERING CALCULATION - COMMON STATION SERVICE TRANSFORMER (CSST) LOAD TAP CHANGER STUDY - SQN-EEB-MS-TI06-0007

Attachment

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W., SUITE 2900
ATLANTA, GEORGIA 30323-0199

February 12, 1996

MEMORANDUM TO: Frederick J. Hebdon, Director
Project Directorate II-4
Office of Nuclear Reactor Regulation

FROM: Ellis W. Merschoff, Director
Division of Reactor Projects

SUBJECT: TIA 96-001, REQUEST FOR REVIEW ASSISTANCE OF SEQUOYAH JCO FOR POTENTIAL DEGRADATION OF ECCS THROTTLE VALVES DURING A LOCA

Attachment 1 is a Justification for Continued Operation (JCO) supplied to TVA by Westinghouse on September 7, 1995. The JCO was used at Sequoyah to justify continued operation with a potential failure of the throttle valves located in the discharge lines of the Charging Pumps and Safety Injection pumps under post-LOCA conditions. Attachment 2 provides a description of this condition for Westinghouse plants. The failure of the throttle valves due to erosion could cause pump damage from run-out after two days. The following technical issue needs to be addressed in order to disposition Sequoyah's denial of a violation issued in Inspection Report 327, 328/95-18. The licensee denied the NOV (Attachment 3), in part, because they currently consider the potential ECCS throttle valve degradation **NOT** to be a condition adverse to quality as described in 10 CFR 50, Appendix B, Criterion XVI.

We request that NRR evaluate the technical adequacy of the JCO relating to the licensee's position that no condition adverse to quality exists in this case. Specific questions relating to the issue are:

- From an EOP perspective, is there an accident scenario that could require the use of the Saftey Injection or Charging Pump flowpaths longer than two days? If so, could the use of these flowpaths cause degradation of the ECCS throttle valves to the point that pump damage could occur?

- Is there a requirement for ECCS components to be available for a specified period of time following a LOCA to satisfy the requirements of 10 CFR 50.46?

- Is the Westinghouse NASL-94-016 (Reference 3 in the JCO) methodology an acceptable method to accomplish the intended objective of hot leg recirculation?

- If a LOCA occurs (large cold leg break with failure of the RHR hot leg MOV) then this RHR flowpath is not available for hot leg recirculation. Based on this condition and the ECCS throttle valve potential degradation issue, is it necessary for the licensee to provide operators

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additional procedural guidance to assure design basis events are adequately addressed?

- Is the flow path from the Safety Injection pumps, operating on piggy back mode, to the hot legs considered the redundant method to perform hot leg recirculation? Would the inability to perform hot leg recirculation dictate that the ECCS be considered inoperable?

- Is the potential degradation of the ECCS throttle valves as discussed in the JCO a condition adverse to quality as defined by 10 CFR 50, Appendix B, Criterion XVI?

If you have any questions regarding this request, please contact Mr. William Holland at (423) 842-8001 or Mr. Mark Lesser at (404) 331-0342.

Docket Nos. 50-327 and 50-328
License Nos. DPR-77 and DPR-79

Attachments:
1. JCO supplied to TVA by Westinghouse dated September 7, 1995
2. Nuclear Safety Advisory Letter dated January 11, 1996
3. Licensee response to the NOV

cc w/atts:
R. W. Cooper, RI
W. L. Axelson, RIII
J. E. Dyer, RIV
K. E. Perkins, WCFO
M. S. Lesser, RII
[redacted]
W. E. Holland, RII
S. E. Sparks, RII
J. C. Barnes, RII

QA Record

B38 950925 801

Westinghouse Electric Corporation

Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

NAR

TVA-95-169
September 7, 1995

Ref: TVA-94-158

Contract No.
91NNP-86305B
Task N94-023

Mr. Mark Burzynski, Manager
Department of Nuclear Engineering
Tennessee Valley Authority
P.O. Box 2000
Soddy Daisy, TN 37379

TENNESSEE VALLEY AUTHORITY
SEQUOYAH NUCLEAR PLANT
Revised ICO for 100 Days Post-LOCA for CCP/SI
Balancing Valves Potentially Failing

Dear Mr. Burzynski:

In response to your request, the attached revised Justification for Continued Operation (JCO) addresses the potential failure of the balancing valves located in the discharge lines of the Centrifugal Charging Pumps (CCPs) and the Safety Injection Pumps (SIPs) under post-LOCA conditions for 100 days.

If you have questions, please contact the undersigned.

References:
1. SQ930800PER
2. TVA-94-158 (B38 940913 801)
3. N9222 (B38 950214 802)

Very truly yours,

Nelli Agostino for
D.W. Salak
Sequoyah Project Manager
TVA Projects

cc: D. Lafever

JMG 9/25

MJB:DML CC:	DRAFT REPLY	COM COPY	INFO COPY	DUE DATE
SE				
~~SE~~ J F Burrow				
~~MLS~~ GG Bair				
~~SSE~~ J Attaway h				
M/N DML				
JA Vogel				
WM Justice				
JD Smith				
DL Chapman				

DM Lafever, x8377

NZN-048
MASTER FILE

RIMS, WT 3B-K, w/Attachment

1307e

Attachment 1

TVA Sequoyah
Reference N94-023

JCO For 100 Days Post LOCA
For CCP/SI Balancing Valves Failing

Revised 9/7/95

Summary

This justification for continued operation addresses the potential failure of the balancing valves located in the discharge lines of the CCP pumps and the SI pumps under post-LOCA conditions for 100 days. During this time period, recirculation flow is needed to satisfy the FSAR LOCA and Long Term LOCA Mass & Energy Release Containment Integrity accidents. The recirculation flow requirements would be satisfied based on the following assessment.

Introduction

The calculations performed to evaluate the balancing valves show that the valves do not fail instantaneously. They are expected to initially perform their intended function and to deteriorate over a time period such that the CCP and SI pumps would also initially perform their intended function and deteriorate over a time period such that the performance of the pumps would be detected (flow monitoring for runnout conditions) and operator action taken to avoid pump failure and provide either hot leg recirculation or cold leg recirculation appropriately.

This assessment is based on the post accident EOPs bringing the plant to cold shutdown conditions, the operator monitoring the post accident conditions of a small break LOCA resulting in relatively low flows at relatively high pressures being provided by the CCP/SI pumps and experienced by the balancing valves, and the operator monitoring the large break LOCA resulting in relatively high flows at relatively low pressure being provided by the CCP/SI pumps and experienced by the balancing valves.

This assessment takes into consideration that the hot leg recirculation configuration would not have a direct line from the RHR to the hot leg due to the postulated failure of the motor operated valve in the line. The cold leg recirculation configuration would always have a direct line from the RHR pump to the cold leg.

It is noted that, for a Small Break LOCA, the pressure drop across the balancing valves is not sufficient to lead to a pump runout condition, until the RCS has been depressurized. Following depressurization, the CCP/SI termination criteria in the Emergency Operating Procedures will result in shut off of the CCP/SI pumps before the valves would erode to the point where pump runout would occur. As a result, it is assumed that CCP/SI termination will be performed in accordance with the plant Emergency Operating Procedures, which ensure that sufficient RHR flow will be available for core cooling and to preclude boron precipitation.

Balancing Valves And CCP/SI Pumps

Based on the current system configuration and alignment following a LOCA, it has been determined that all balancing valves would erode by cavitation (PER TVA-94-118) and the throttling capability would be significantly compromised in approximately 12 days.

The intensity of the cavitation damage was determined to be a function of the pump flow and the material erosion of the valve's plug and seat areas. A functional relationship between the seat/plug material erosion vs. the valve's Cv value was derived from historical data obtained from field recorded data and from consultation with the valve vendor. This functional relationship predicted that over a 12 day period, erosion of the seat and plug would gradually reduce the throttling capability of the valve by the 12th day. The cavitation damage would be limited to the valve's seat and plug, however the structural and pressure boundary (body) integrity should be maintained during the next 100 days. It is still our recommendation that the valves be replaced at the end of the 100 day period.

SI Pump and Centrifugal Charging Pump Runout

The Sequoyah SI pumps and centrifugal charging pumps were originally designed for runout flow rates of 650 GPM and 550 GPM, respectively. The allowable runout flow rates were later increased by Westinghouse letter TVA-91-309 to 675 GPM and 560 GPM, respectively. The Sequoyah pumps are currently balanced to maintain runout flows within these limits.

It has been determined that the ECCS balancing valves will gradually erode due to the large pressure drop across the valve following a LOCA, especially after a large break. The increase in the valve Cv will result in an increased pump flow. However, the increase in pump flow will occur at a more critical rate than

the valve degradation due to the relationship of pump flow and the valve Cv. Thus, pump operation is more limiting than the valve degradation. The pump operation has been evaluated for both large break and small break LOCA conditions.

Large Break LOCA

A parametric study was conducted of the increase in pump flow rate versus the rate of valve degradation. This study showed that the SI and charging pump flow rate will increase in the range of 10 to 30 GPM during a two day period following a large break LOCA. This increase in pump flow due to the degrading balancing valves could allow the SI and charging pumps to exceed the allowable flow rates provided by TVA-91-309. If this occurs, these pumps may experience cavitation in the first and second impeller stages. However, Westinghouse believes that the SI pumps and centrifugal charging pumps will operate successfully for a period of two days based on the following arguments:

1. The cavitation should be relatively low in energy and its effect will be diminished by the high suction pressure created by the boost effect of the RHR pumps. Some pump degradation may occur as further discussed un TVA-91-309, but the pumps are expected to remain operable.

2. Recent testing by the pump vendor and several utilities has shown that the same pump models will operate acceptably at runout flows slightly higher than those identified in TVA-91-309.

Small Break LOCA

The pump flow rate following a small break LOCA is expected to remain within the existing approved limits based on lower initial pump flow due to higher system backpressure and the fact that the balancing valves will not degrade under this condition.

It can be concluded that the SI and centrifugal charging pumps will operate acceptably for a period of two days following a large break LOCA and indefinitely following a small break LOCA. After a large break LOCA, Westinghouse recommends that the SI pumps and centrifugal charging pumps be shut down after 2 days. The pumps may be run beyond 2 days if the flow rates do not exceed the limits identified in TVA-91-309.

LOCA

There are essentially four LBLOCA scenarios that must be considered to assess whether post-LOCA operation with a failure of the CCP/SI balancing valves will be acceptable; these scenarios can be summarized as: (1) large cold leg break, no MOV failure; (2) large cold leg break, MOV failure: (3) large hot leg break, no MOV failure; and, (4) large hot leg break, MOV failure. Each of these LBLOCA scenarios will be evaluated assuming that 1 RHR pump will be available to provide continuous flow in the long-term following a loss-of-coolant accident. In addition to the LBLOCA scenarios, Small Break LOCA must also be addressed. Finally, the potential for boron precipitation will also be considered. It should be noted that the evaluations presented below assume a conservatively early depressurization time of 12 hours.

1. Large Cold Leg Break, no MOV Failure

In this scenario, RHR flow would be realigned to the RCS hot legs 12 hours after accident initiation, consistent with the plant EOPs. With the break in the cold leg and LHSI to the hot legs, a flow path through the core would be established and maintained, thus ensuring core cooling and precluding boron buildup in the core. As a result, flow from one RHR pump would be sufficient for a cold leg break with no failure of the MOV.

2. Large Cold Leg Break, MOV Failure

In this scenario, RHR alignment to the RCS hot legs would be attempted at 12 hours after accident initiation. With a failure of the MOV, however, injection to the hot legs would not be possible; as a result, the only way to ensure any ECCS injection would be to realign the RHR flow back to the cold legs. In this alignment, cold leg injection would result in injected flow travelling around the downcomer and out the break, such that forced flow through the core could not be established. Boron concentration in the core would continue to increase, ultimately resulting in precipitation once the solubility limit was reached.

Ref. 3 describes a methodology where credit can be taken for flow through the gap between the core barrel and the reactor vessel at the hot leg nozzle locations. Essentially, flow through the gap will result in a forward flush path through the core, thus limiting boron buildup. As noted in Ref. 3, credit for flow through the hot leg nozzle gaps is currently being pursued as part of a Westinghouse Owners' Group (WOG) program. In any

event, taking credit for the potential flowpath through the hot leg nozzle gap will result in flow from one RHR pump being sufficient for a cold leg break with an assumed failure of the MOV.

3. Large Hot Leg Break, no MOV Failure

In this scenario, RHR flow would be realigned to the RCS hot legs 12 hours after accident initiation, consistent with the plant EOPs. Since the break is also in the hot leg, however, safety injection would flow across the core and out the break without mixing in the core. In such a scenario, it is necessary to demonstrate that the hot leg injection exceeds 3.3 times boiloff, based on decay heat levels at hot leg recirculation time.

With one RHR pump available, hot leg injection in this scenario was calculated to exceed 3.3 times boiloff. As a result, flow from one RHR pump will be sufficient for a hot leg break with no failure of the MOV.

4. Large Hot Leg Break, MOV Failure

In this scenario, RHR alignment to the RCS hot legs would be attempted at 12 hours after accident initiation, consistent with the plant EOPs. With a failure of the MOV, however, injection to the hot legs would not be possible; as a result, the only way to ensure any ECCS injection would be to realign the RHR flow back to the cold legs. With the break in the hot legs and ECCS injection to the cold legs, a flowpath through the core would be established and maintained, thus ensuring core cooling and precluding boron buildup in the core. As such, flow from one RHR pump would be sufficient for a hot leg break with a failure of the MOV.

5. **Small Break LOCA**

For a Small Break LOCA, shutting off the CCP/SI pumps in the long-term could potentially result in no ECCS injection if the system pressure remains above the RHR cut-in pressure when the high pressure flow is terminated. For Small Break LOCA, it was noted that the CCP/SI pumps can operate longer than for Large Break LOCA, as the pressure drop across the balancing valves is not sufficient to lead to a pump runout condition until the RCS has been depressurized. After the RCS has been depressurized, it is possible for the balancing valves to experience erosion, since the pressure drop across the valves will be increased. However, the CCP/SI termination criteria in the Emergency Operating Procedures will result in shut off of the CCP/SI pumps before the valves would erode to the point where pump runout would occur. As a result, it is assumed that CCP/SI termination following a Small Break LOCA will be performed in accordance with the plant Emergency Operating Procedures, which ensure that sufficient RHR flow will be available for core cooling and to preclude boron precipitation.

LOCA Conclusion

Based on the information presented above, it has been determined that a JCO exists for long-term failure of the CCP/SI balancing valves at Sequoyah Units 1 & 2, provided the operator can take actions to shut off the CCP/SI pumps prior to runout of the RHR pumps, such that at least one RHR pump is available for long-term core cooling.

It is noted that, for a Small Break LOCA, the pressure drop across the balancing valves is not sufficient to lead to a pump runout condition, until the RCS has been depressurized. Following depressurization, the CCP/SI termination criteria in the Emergency Operating Procedures will result in shut off of the CCP/SI pumps before the valves would erode to the point where pump runout would occur. As a result, it is assumed that CCP/SI termination will be performed in accordance with the plant Emergency Operating Procedures, which ensure that sufficient RHR flow will be available for core cooling and to preclude boron precipitation.

Note that the failure of an RHR-line MOV was considered, consistent with Ref. 2; in this scenario, it is necessary to realign RHR flow to the RCS cold legs following the attempt to switchover to hot leg recirculation. In addition, note that credit was taken for flow through the gap between the core barrel and reactor vessel at the hot leg nozzles to show acceptability for the cold leg break with an assumed MOV failure. While this methodology has not been approved, a WOG program is presently ongoing to take credit for this recirculation flowpath on a generic basis.

LOCA References

(1) TVA-94-118, "Tennessee Valley Authority, Sequoyah Nuclear Plant: CC/SI Balancing Valve Evaluation", 7/18/94.

(2) Nuclear Safety Advisory Letter NSAL-92-010, "Hot Leg Switchover Methodology", 1/9/93.

(3) Nuclear Safety Advisory Letter NSAL-94-016, "Core Recriticality During LOCA Hot Leg Recirculation", 7/25/94.

Long Term LOCA Mass & Energy Release Containment Integrity

The design basis Long Term LOCA mass and energy release analysis for the Sequoyah Units 1 and 2 is the double-ended pump suction minimum safeguards case. The design basis event for Sequoyah is documented in WCAP-12455, Supplement 1. For long term containment integrity consideration long residual heat removal (core-boiloff) is addressed during the recirculation phase of the transient. Delivery of water to the core is continued for long term cooling via sump recirculation. For the DBA the residual heat removal system (1 RHR pump; assumes diesel train failure) take suction from the containment sump delivering through a residual heat exchanger to the cold legs of the RCS (a total core flow of 1019 gpm) and the RHR containment spray headers.

Westinghouse
Energy
Systems
Business
Unit

NUCLEAR SAFETY ADVISORY LETTER

THIS IS A NOTIFICATION OF A RECENTLY IDENTIFIED POTENTIAL SAFETY ISSUE PERTAINING TO BASIC COMPONENTS SUPPLIED BY WESTINGHOUSE. THIS INFORMATION IS BEING PROVIDED TO YOU SO THAT A REVIEW OF THIS ISSUE CAN BE CONDUCTED BY YOU TO DETERMINE IF ANY ACTION IS REQUIRED.

P.O. Box 355, Pittsburgh, PA 15230-0355

Subject: Erosion of Globe Valves in ECCS Throttling Applications	**Number:** NSAL-96-001
Basic Component: Globe Valves	**Date:** 01-11-96
Plants: All Westinghouse PWRs	
Substantial Safety Hazard or Failure to Comply Pursuant to 10 CFR 21.21(a)	Yes ☐ No ☒
Transfer of Information Pursuant to 10 CFR 21.21(b)	Yes ☐
Advisory Information Pursuant to 10 CFR 21.21(c)(2)	Yes ☐

SUMMARY

A utility has notified Westinghouse about the potential cavitation and erosion of the Rockwell Edwards throttling valves which are plug guided globe valves with stellited plugs and seats and which are used in high head emergency core cooling system (ECCS) applications. The valve erosion may occur during a loss of coolant accident (LOCA) when the pressure drop across the valves is high. As a result of the erosion, the high head ECCS pumps (i.e. charging/safety injection (CHG/SI) and safety injection (SI) pumps) may exceed their runout point, and the operator may have to terminate the pumps during the LOCA. It is believed that you should be able to demonstrate that these pumps operate long enough to mitigate the LOCA before the operator would terminate the pump flow. Therefore, it is believed that the valve erosion should not represent a substantial safety hazard pursuant to 10 CFR 21. However, with the valve erosion, the pumps may not remain operable for their licensing basis time (e.g. - 100 days post LOCA). Thus, this issue may still present a licensing basis concern. You may wish to review your plant specific configuration to determine whether you have any valves affected by this issue and take corrective actions, as required.

Additional information, if required, may be obtained from the originator. Telephone 412-374-6460.

Originator(s): J W Fasnacht
J. W. Fasnacht
Regulatory & Licensing Initiatives

H A Sepp
H. A. Sepp, Manager
Regulatory & Licensing Initiatives

Attachment 2

Issue Description

Reference 1 identified a condition which may lead to the potential failure of the throttling function of the valves located in the safety injection lines for the centrifugal charging/safety injection pumps (CHG/SI) and the safety injection (SI) pumps. Reference 1 reported this condition for Rockwell-Edwards univalves (globe valves) applied for this throttling service. These valves are plug guided globe valves that are stainless steel with stellited plugs and seats.

The reported condition is that during a large break loss of coolant accident (LOCA), there is a high pressure drop across the throttling valves as the CHG/SI and SI pumps inject flow to the reactor coolant system (RCS). As a result of the high pressure drop, the valves may experience cavitation and internal erosion. Consequently, the valves may not restrict the CHG/SI and SI pump flows below the runout limits, and the pumps may not operate for the licensing basis required time (e.g. - 100 days post LOCA).

Technical Evaluation

The extent of the globe valve erosion is a function of several factors. Reference 1 reported the failure for the Rockwell Edwards valves which are stainless steel globe and have stellited plugs and seats. Westinghouse has only supplied these type valves for CHG/SI and SI pump throttling applications to a few plants. The valves for this application are often supplied by the utility and/or architect engineering company. Thus, there may be other valve designs used in this application.

The mechanism for the erosion is as follows. During a LOCA, the CHG/SI and SI pumps create a high pressure drop across the valves. The high pressure drop occurs when the back pressure from the RCS is small. It is difficult to predict the pressure drop at which the cavitation induced erosion occurs. However, it is believed that the cavitation induced erosion may begin when the pressure drop is about 30-40% of the valve inlet pressure. As the pressure drop increases, the cavitation becomes more pronounced and begins to erode the valve at a higher rate. The valve parts most susceptible to erosion are the plug and the seat. As the valve trim wears, the valve is less able to restrict the CHG/SI or SI pump flow to the point where these pumps may exceed their runout flows. Westinghouse has estimated for one four loop plant (with Rockwell Edwards globe valves) that with a valve inlet pressure of 880 psig, a valve pressure drop of 880 psid and a valve flow of 82 gpm, the potential valve erosion may increase the CHG/SI flow by as much as 7 gpm/hour. This erosion rate is an estimate and is provided for illustration purposes only. You should evaluate your valves to determine their resistance to cavitation induced erosion, as required.

Once it has been determined that the erosion may occur, it is necessary to evaluate the impact that the erosion has on the CHG/SI and SI pump performance. As previously mentioned, the erosion is most likely to occur when the pressure drop is high. This high pressure drop is expected to occur during a LOCA. The high pressure drop occurs at a different times depending on whether the LOCA is a Small Break (SB) LOCA or a Large Break (LB) LOCA.

Erosion During a LBLOCA

During a LBLOCA, the initial pressure drop across the valves is high and is a situation likely to induce the potential valve erosion. The injection phase of the LBLOCA will probably last between 20-45 minutes, depending upon the plant design. Afterwards, the CHG/SI and SI pumps would be aligned for cold leg recirculation. During the cold leg recirculation, the valves may continue to erode to the point where the operator may have to stop the CHG/SI and SI pumps. However, it is believed that you can demonstrate that the pumps should provide sufficient flow to mitigate the LBLOCA. before the operator has to stop the pumps and rely on the residual heat removal system (RHRS) to provide the required core

cooling. The reason is that the valves should erode at a slow enough rate to allow for continued CHG/SI and SI pump flow in a manner consistent with the flow requirements to mitigate the LBLOCA. However, this belief should ultimately be confirmed for your plant specific configuration, as required.

Erosion During a SBLOCA

During a SBLOCA, the pressure drop across the globe valves is initially low when compared to the pressure drop across the valves during a LBLOCA. As a result, the rate of erosion will be very small, if not negligible.[1] As the RCS is depressurized, the pressure drop across the valves will eventually increase which will increase the rate of erosion. It will take longer to erode the valves to the potential point where pump runout occurs for a SBLOCA than for a LBLOCA. However, the CHG/SI and SI pumps may need to be operable for a longer period of time during the SBLOCA than for the LBLOCA.

The SBLOCA Peak Clad Temperature (PCT) turnaround time would probably not be longer than 3 hours. Thus, the CHG/SI and SI pump should operate long enough to the point where PCT turnaround occurs. Furthermore, it is likely that the operator would depressurize the RCS and reach the emergency operating procedure (EOP) termination criteria for the CHG/SI and SI pumps before the pumps would reach their runout points. If the plant did not reach the EOP termination criteria and the CHG/SI and SI pumps approached their runout limits, the operator could (if necessary and permitted by EOPs) terminate the CHG/SI pumps and align the SI pumps for hot leg recirculation. This alignment would allow for continued SI pump flow to the core, until the EOP termination criteria were met. However, it should be noted that the hot leg recirculation lines may have the same type globe valves and may be subject to the same potential erosion over time. (See discussion below regarding hot leg recirculation).

Therefore, it is believed that the CHG/SI and SI pumps should remain operable long enough to perform their intended safety function during a SBLOCA and provide enough flow to the core before the RHRS could provide the required long term core cooling. This belief should ultimately be confirmed for your plant specific configuration, as required.

Impact on Hot Leg Recirculation

For many plant designs, the SI pumps are aligned for hot leg recirculation during a LOCA. If hot leg switchover is required, the operator may align both the RHR and SI pumps for hot leg recirculation. However, the globe valves in the SI pump hot leg recirculation lines may experience the same erosion as the globe valves located in the cold leg injection lines. As a result, SI pumps may not be able to provide the long term hot leg recirculation requirements. Consequently, the operator would rely on the RHR pumps to provide the required hot leg recirculation flow. The problem with relying on the RHR pumps for hot leg recirculation is that the RHRS hot leg recirculation path may not be single failure proof. If the single isolation valve for the RHRS hot leg recirculation line fails to open, then the RHRS could not be used for hot leg recirculation.

To address this issue, the plant can take credit for the bypass flow between the core barrel and reactor vessel at the hot leg locations. This credit would enable the plant to show that the hot leg recirculation is not required since the bypass flow is high enough to forward flush the core and prevent boron precipitation. (See Reference 2 for more information.) Reference 2 also indicates that the WOG has initiated a program to provide the necessary information to license this methodology with the NRC.

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[1] If the break is downstream of the globe valve, the pressure drop across the globe valve would be higher since the flow in the broken line would spill to containment pressure. Therefore, the erosion rate would be higher for the globe valve in the broken line.

Safety Significance

The technical evaluation indicates that even with the throttling valve erosion, the CHG/SI and SI pumps should remain operable long enough to mitigate the LOCA and to allow the operator to use the RHRS for long term core cooling. Thus, it is believed that the valve erosion should not represent a substantial safety hazard pursuant to 10 CFR 21.

However, the valves may eventually erode to the point where the CHG/SI and SI pumps exceed their runout points. Thus, the erosion may prevent the pumps from operating for their licensing basis operating time (e.g. - 100 days post LOCA). Thus, the erosion of the valves could still present a licensing basis issue.

NRC Awareness

At least one utility has filed a Licensee Event Report with the NRC regarding this issue. Therefore, the NRC is aware of this issue.

Recommendations

The following actions are recommended for this issue.

1. Determine whether you have throttling valves which may be affected by this issue. One utility identified the valve failure for Rockwell Edwards Univalves used in CHG/SI or SI pump throttling applications. The following are the primary valve characteristics which make the valves suspect to the potential for cavitation and erosion:

   - plug guided globe valves which are stainless steel with stellited plugs and seats, and
   - throttled to incur a high enough pressure drop that could induce cavitation and erosion.

   Westinghouse has only supplied these type valves to a few plants and typically does not supply the valves for this application. The valves for this application are often supplied by the utility and/or architect engineering company.

2. If you determine that you have globe valves which may be affected by this issue, you may wish to consider the following corrective actions:

   A. Replace the valves with another type valve which is more suitable for CHG/SI and SI pump throttling applications.

   B. Install orifice plates in series with each globe valve. The orifice plates would incur more of the required pressure drop and allow you to reset the globe valves to take less pressure drop. The orifice plates can be sized so that the globe valves can be set in a manner that would preclude globe valve erosion.

   C. Install variable orifice plates in each of the CHG/SI and SI pump discharge lines. The orifice plates would also incur more of the required pressure drop and allow you to reset the globe valves to take less pressure drop. The orifice plates can be sized so that the globe valves can be set in a manner that would preclude globe valve erosion.

D. Perform a test to better define the rate of wear of the valve trim. This test would provide for a more accurate characterization of the wear rate and help you determine whether any corrective actions are required.

References

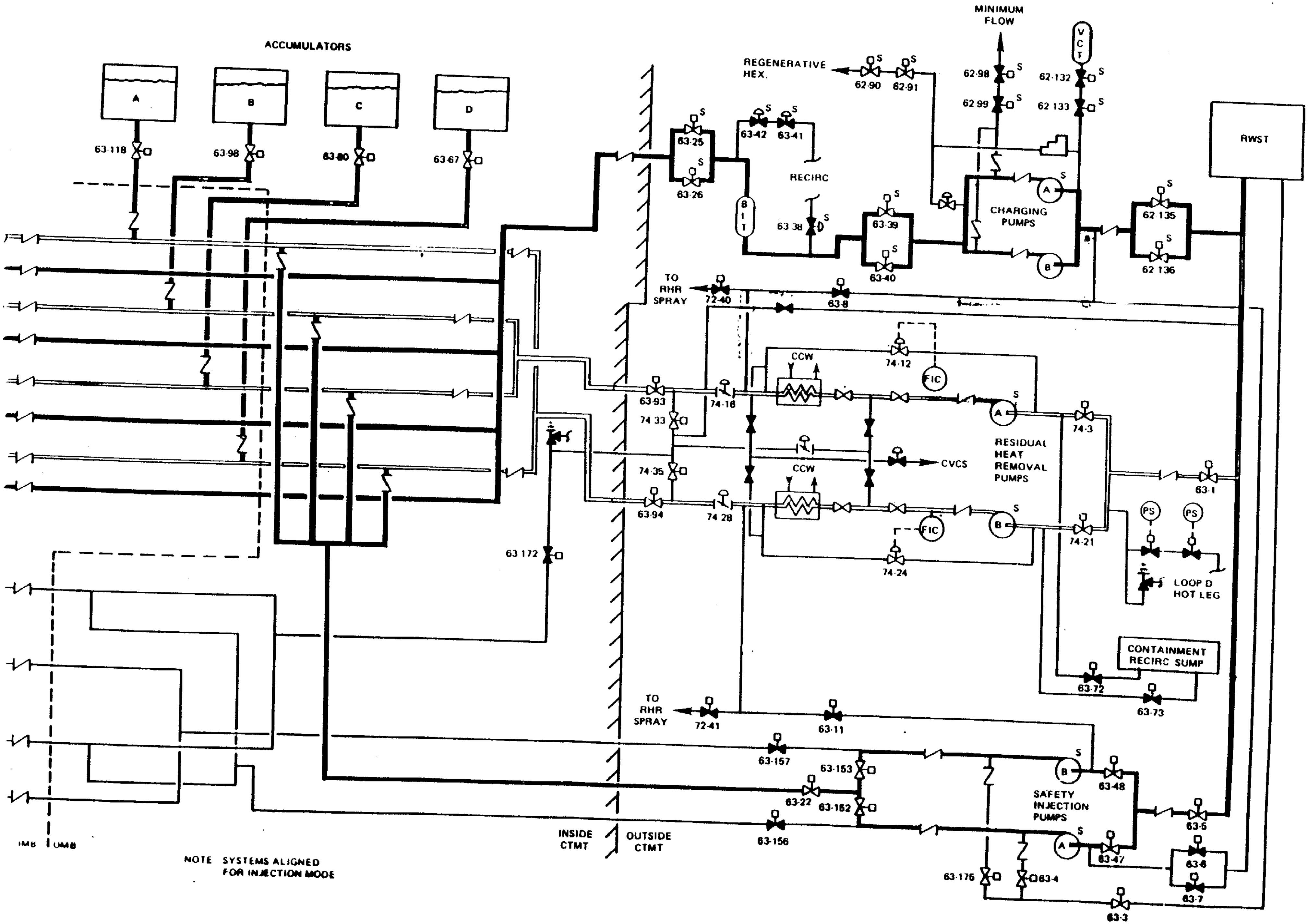
1. INPO Nuclear Network OE7127, "Sequoyah Nuclear Plant - Potential Post LBLOCA CCP/SIP Runout Damage Due to Throttle/Balance Valve Seat Erosion Caused by High DP"

2. Westinghouse Advisory Letter NSAL-94-016, dated 7/25/94

1998 NSAL INDEX

NSAL-98-001	01/11/98	EROSION OF GLOBE VALVES IN ECCS THROTTLING APPLICATIONS	ALL WESTINGHOUSE PWR'S

EMERGENCY CORE COOLING SYSTEMS



NOTE: SYSTEMS ALIGNED FOR INJECTION MODE

S64 951031 800

Tennessee Valley Authority, Post Office Box 2000, Soddy-Daisy, Tennessee 37379-2000

R.J. Adney
Site Vice President
Sequoyah Nuclear Plant

October 31, 1995

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

Gentlemen:

In the Matter of	)	Docket Nos. 50-327
Tennessee Valley Authority	)	50-328

SEQUOYAH NUCLEAR PLANT (SQN) - NRC INSPECTION REPORT NOS. 50-327, 328/95-18 - REPLY TO NOTICE OF VIOLATION (NOV) 50-327, 328/95-18-01

Enclosed is TVA's reply to Mark S. Lesser's letter to O. D. Kingsley, Jr., dated October 2, 1995, which transmitted the subject NOV. This NOV pertains to corrective action associated with emergency core cooling system throttle valves.

TVA denies that a violation occurred as stated in the NOV. The enclosure explains the reasons for TVA's denial.

In addition, the discussion in Inspection Report 95-18 concerning this issue indicates a difference of opinion between NRC and TVA in the application of Generic Letter 91-18 and 10 CFR 50.59 when discrepancies between the plant and its licensing basis are identified. TVA has requested a meeting with NRC to resolve this difference of opinion. We believe the resolution of this difference may affect the ultimate disposition of this violation.

If you have questions regarding this response, please telephone R. H. Shell at (423) 843-7170.

Sincerely,

R. J. Adney

Enclosure
cc (Enclosure):

Mr. D. E. LaBarge, Project Manager
Nuclear Regulatory Commission
One White Flint, North
11555 Rockville Pike
Rockville, Maryland 20852-2739

NRC Resident Inspector
Sequoyah Nuclear Plant
2600 Igou Ferry Road
Soddy-Daisy, Tennessee 37379-3624

Regional Administrator
U.S. Nuclear Regulatory Commission
Region II
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323-2711

ENCLOSURE
RESPONSE TO NRC INSPECTION REPORT
NOS. 50-327, 328/95-18
MARK S. LESSER'S LETTER TO OLIVER D. KINGSLEY, JR.
DATED OCTOBER 2, 1995

Violation 50-327, 328/95-18-01

"10 CFR 50, Appendix B, Criterion XVI requires, in part, that measures shall be established to ensure that conditions adverse to quality are promptly identified and corrected.

"Contrary to the above, prompt corrective action was not implemented for a condition adverse to quality. Specifically, on July 18, 1994, Westinghouse notified the licensee of an adverse condition involving accelerated degradation of Emergency Core Cooling System throttle valves during accident scenarios which could cause premature system pump failure, and actions to correct or compensate for the condition were not implemented until July 17, 1995.

"This is a severity level IV violation (Supplement 1)."

Background Information

The subject issue involves a condition where throttle valves in the high head and intermediate head safety injection portion of the emergency core cooling system may degrade as a result of a high pressure drop across the valves during a loss of coolant accident (LOCA). This pressure drop is postulated to result in cavitation-induced erosion of the throttle valve seats. Erosion of the valve seat could result in a loss of flow resistance which may allow the emergency core cooling system (ECCS) pumps to approach or exceed run-out flow within 48 hours post-LOCA

The issue was originally identified at Sequoyah Nuclear Plant (SQN) through the corrective action program generic review of a Watts Bar Nuclear Plant (WBN) problem. During a design review of the WBN emergency core cooling system, the WBN intermediate and high head injection flow balance valves were determined to be globe valves which are not designed for flow balancing operation. A subsequent review of the valves at SQN confirmed these valves to be significantly different from the WBN valves. The SQN valves are throttle valves which are specifically designed for flow balancing operation. To demonstrate the suitability of the SQN valves, TVA performed a calculation utilizing the methodology and acceptance criteria from Electric Power Research Institute (EPRI) Standard NP-6516, "Guide for the Application and Use of Valves in Power Plant Systems." This calculation concluded that the SQN valves could be required to operate under flow conditions which result in valve seat erosion.

At that time, Westinghouse was contacted to assist in the evaluation of the subject condition since the susceptible valves were within the scope of the equipment originally supplied by the nuclear steam supply systems (NSSS) equipment vendor and were originally specified by Westinghouse before the EPRI standard was published.

Westinghouse provided SQN with a justification for continued operation which indicated that the existing throttle valves will perform their function for a minimum of 48 hours following a LOCA. Westinghouse concluded that the intermediate and high-head injection pumps are not required to operate more than 48 hours following a LOCA since the reactor coolant system conditions are such that a single RHR pump is sufficient for long-term cooling. This evaluation assumed that the existing SQN emergency operating procedures would be in effect for accident mitigation and recovery.

As part of the closure process for the corrective action document, the Management Review Committee (MRC) reviewed the issue in February 1995. The MRC directed that this issue be captured in the Technical Support Center activation and operation procedure (Emergency Plan Implementing Procedure [EPIP] 6). The MRC believed that this procedure was the best place to remind plant personnel of the issue because the procedure would be in use if the potential degradation were to occur. The subject procedure was revised on July 17, 1995.

Basis for Denial of the Violation

TVA does not dispute that it did not promptly respond to the MRC direction to revise the Technical Support Center activation and operation procedure. However, TVA concluded in the Summer of 1994 and still concludes that based on the evaluation performed by Westinghouse, a revision to the procedure was not required for continued safe operation of the plant. Consequently, it was not a required corrective action in the context of 10 CFR 50, Appendix B, Criterion XVI. The subject procedure revision was merely an enhancement to remind plant personnel of a potential condition following a LOCA. This position is backed by a revised Westinghouse evaluation which clarified that the original justification of continued operation was based upon accident mitigation and recovery utilizing the existing emergency procedures and that no supplemental procedure changes were necessary. If the existing procedures were followed, RCS conditions would be such that operation of the intermediate and high-head ECCS pumps would be limited during the time when significant erosion of the flow-balancing valves is postulated to occur. The plant emergency procedures will ensure that sufficient RHR flow is available for long-term core cooling.

In summary, because the existing plant emergency procedures would limit throttle valve seat erosion and the fact that the subject pumps would not be required to operate if the valve seats became eroded several days following a LOCA, no further corrective actions were needed. The subsequent revision to EPIP-6 was an enhancement not a corrective action. As such, the timeliness of its implementation should not be the basis of a violation.

For these reasons, TVA denies this violation.

File

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
475 ALLENDALE ROAD
KING OF PRUSSIA, PENNSYLVANIA 19406

August 6, 1996

MEMORANDUM TO: Steven A. Varga, Director
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

FROM: Richard W. Cooper, II, Director
Division of Reactor Projects, Region I

SUBJECT: PROPOSED TASK INTERFACE AGREEMENT (TIA) REGARDING OYSTER CREEK DRY FUEL MOVEMENT WITH THE PLANT IN COLD SHUTDOWN

General Public Utility Nuclear (GPUN) has prepared a 50.59 evaluation for movement of irradiated dry fuel from the in plant spent fuel storage area to the on-site dry fuel storage vaults with the plant in the cold shutdown condition. GPUN has concluded that this can be accomplished in a safe manner in accordance with all license requirements and commitments. Your assistance is requested in addressing the following policy and technical questions relative to this GPUN 50.59:

POLICY

- The FSAR analysis for this fuel lift is for a heavy load of 30 tons, and with the cask the licensee plans to use the load is 100 tons. Since dropping this heavier load could possibly have increased consequences, does this constitute an unreviewed safety question?

- If NRR considers that this aspect of the above 50.59 does not involve an unreviewed safety question, are there any other reasons why NRR considers that use of a 50.59 analysis may not be an acceptable approach for the licensee to follow in performing a safety analysis of the above described movement of fuel to the on-site storage vaults? (see also technical questions below)

TECHNICAL

These questions stem from a site visit and brief look at the current subject 50.59 by Peter Eselgroth on July 23, 1996. The following questions are not intended to imply that Region I considers this movement of fuel with the plant shutdown to the unacceptable, but rather to indicate the need for NRR involvement in a determination of the acceptability of this approach:

- The load transfer path over the refueling floor area was revised to reduce the complexity of the crane manipulations involved. The revised path appears to include fewer reactor building structural members. Is this current path acceptable?

ITEM # 1

D/3

- Whether or not a 50.59 analysis of this cold shutdown fuel movement is acceptable, your assistance is requested in evaluating the following potential consequences of a dropped load:

  - The credibility of shock waves in the reactor building causing chatter and resultant change of state of relays leading to closure of MSIVs at a time when this may be the reactor vent path; opening of a vent path through the electromagnetic relief valves; the reactor no longer being in cold shutdown by Technical Specifications.

  - The consequences of severing reactor building SGTS ductwork in Area 1-6-B from the suction fan inlet and losing suction on other parts of the reactor building.

  - The consequences of damage to the torus (with some low levels of radioactive contamination) and the loss of shutdown cooling.

Since NRR has previously visited the site and reviewed areas discussed in this TIA, Region I desires to know if this is sufficient to address the questions raised in this TIA or if additional on site visits will be needed prior to GPUN exercising this option for moving fuel.

Oyster Creek enters their next refueling outage on September 7, 1996 and it is our understanding that they do not presently plan to move fuel to dry storage during this outage. However, this is not a certainty, particularly if the licensee were to encounter unforeseen delays associated with outage work. It is requested that NRR take the lead on resolution of the above policy and technical questions on a time scale commensurate with the need for NRC review of the cold shutdown fuel movement option and that, in any event, a response be provided to this TIA within 30 days of receipt. The Region I point of contact is Peter Eselgroth, Chief of Reactor Projects Branch 7 (610-337-5234).

cc:
J. Wiggins, DRS
P. Eselgroth, DRP
W. Travers, NMSS
R. Eaton, NRR
W. Reckley, NRR
L. Briggs, DRP
~~M. Oprendek, DRP~~
T. Frye, DRP
C. Anderson, DRP

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

M96352

November 21, 1996

MEMORANDUM TO: Richard W. Cooper, II, Director
Division of Reactor Projects, Region I

FROM: John F. Stolz, Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

John F. Stolz

SUBJECT: TASK INTERFACE AGREEMENT (TIA) REGARDING OYSTER CREEK MOVEMENT OF FUEL USING THE DRY FUEL STORAGE CASK TRANSFER SYSTEM WITH THE PLANT IN COLD SHUTDOWN

Your memorandum dated August 6, 1996, forwarded the subject TIA directed at answering several questions that your staff had regarding the 10 CFR 50.59 evaluation prepared by the licensee to allow movement of spent fuel from the spent fuel pool to the dry storage facility while the plant is in cold shutdown.

While some of your concerns raised in the TIA can be addressed based on the staff's review of the previous Oyster Creek 10 CFR 50.59 evaluation dated February 1996 for movement of the fully loaded dry storage cask during operations, other issues raised are best answered after the staff reviews the latest 10 CFR 50.59 evaluation.

The staff intends to inspect the licensee's completed 10 CFR 50.59 evaluation before they actually move the fuel. The movement of the fuel cannot take place until the licensee has completed a "dry run" without loading fuel in the cask <u>and</u> gives the NRC a 30-day notice of intent to move fuel. The technical staff will conduct the inspection when it is closer to the time the licensee performs the "dry run." The inspection which may involve observing the "dry run," will address the issues raised in your TIA. The staff will issue either a stand alone report or an inspection feeder at the conclusion of the inspection.

Docket No. 50-219

cc: E. W. Merschoff, RII
J. Caldwell, RIII
J. E. Dyer, RIV

ITE 2