NRC FORM 464 Part I
(6-1998)

U.S. NUCLEAR REGULATORY COMMISSION

UNITED STATES NUCLEAR REGULATORY COMMISSION

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

| FOIA/PA | RESPONSE NUMBER |
|---|---|
| 2000-0234 | 1 |

RESPONSE TYPE: FINAL [ ] PARTIAL [√]

REQUESTER: Ms. Theresa Sutter

DATE: JUN 20 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 A — 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 B — 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 B — 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 *
$

* 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)

SIGNATURE - FREEDOM OF INFORMATION ACT AND PRIVACY ACT OFFICER

for Carol Ann Reed — Nina L. Pugh

FOIA No. 2000-0234

## APPENDIX A
## RECORDS ALREADY AVAILABLE IN THE PDR

| NO. | DATE | ACCESSION NUMBER | DESCRIPTION/(PAGE COUNT) |
|---|---|---|---|
| 1. | 11/24/92 | 9212040323 | Memo to E. Merschoff, NRC/RII from G. Lainas, NRC/HQ Re: Close out of Task Interface Agreement (TIA) TIA 92-28, Turkey Point Unit 4 restart following hurricane Andrew (3 pages) |
| 2. | 09/28/92 | 9210060096 | Memo to G. Lainas, NRC/HQ from E. Merschoff, NRC/RII Re: Task Interface Agreement - Turkey Point Unit 4 restart following hurricane Andrew (4 pages) |
| 3. | 02/07/92 | 9202120116 | Memo to L. Reyes, NRC/RII from G. Lainas, NRC/HQ Re: Task Interface Agreement (TIA 92-03) concerning crack in Oconee DHR drop line (2 pages) |
| 4. | 11/21/96 | 9611250197 | Memo to R. Cooper, NRC/RI from J. Stolz, NRC/HQ Re: Task Interface Agreement (TIA) Regarding Oyster Creek Movement of fuel using the Dry Fuel Storage Cask Transfer System with the Plant in cold shutdown (2 pages) |
| 5. | 06/07/96 | 9606110328 | Memo to F. Hebdon, NRC/HQ from E. Merschoff, NRC/RII Re: TIA 96-001, Request for review assistance of Sequoyah JCO for potential degradation of ECCS throttle valves during a LOCA (1 page) |

FOIA No. 2000-0234

# APPENDIX
## RECORDS ALREADY AVAILABLE IN THE PDR

| NO. | DATE | ACCESSION NUMBER | DESCRIPTION/(PAGE COUNT) |
|---|---|---|---|
| 6. | 5/30/96 | 9606030170 | Memo to F. Hebdon, NRC from R. Jones, NRC Re: Sequoyah - TIA 96-001 concerning potential degradation of ECCS due to throttle valves erosion following a LOCA (3 pages) |
| 7. | 06/05/97 | 9708050065 | Memo to H. Berkow, NRC from J. Johnson, NRC Re: TIA 97-014 Catawba frequency requirements for quality assurance audits during routine inspection of Catawba facility (15 pages) |
| 8. | 06/12/98 | 98061802441 | Memo to J. Zwolinski, NRC/HQ from L. Plisco, NRC/RII Re: Task Interface Agreement (TIA 98-003) Crystal River Unit 3; Low Pressure injection emergency core cooling system valve configurations (3 pages) |
| 9. | 04/20/98 | 9804210396 | Letter to M. Roche, GPU Nuclear Corp. from R. Eaton, NRC/HQ Re: Completion of licensing action for NRC Bulletin 96-02, "Movement of heavy loads over spent fuel, over fuel in the reactor core, or over safety-related equipment," dated 4/11/96, for Oyster Creek Nuclear Generating Station (10 pages) |

FOIA No. 2000-0234

## APPENDIX
## RECORDS ALREADY AVAILABLE IN THE PDR

| NO. | DATE | ACCESSION NUMBER | DESCRIPTION/(PAGE COUNT) |
|---|---|---|---|
| 10. | 06/04/97 | 9812140151 | Memo to F. Hebdon, NRC/HQ from J. Johnson, NRC Re: Requests assistance in resolving issue of extent of Maintenance rule implementation for Browns Ferry Unit 1. (9 pages) |

FOIA No. 2000-0234

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

| NO. | DATE | DESCRIPTION/(PAGE COUNT) |
|---|---|---|
| 1. | 01/20/84 | Memo to R. Starostecki, NRC/RI from D. Eisenhut, NRC/HQ Re: Haddem Neck Fire Brigade Quarterly meetings Task Interface Agreement 84-01 (2 pages) |
| 2. | 11/26/85 | Memo to C. Norelius, NRC/RIII from J. Zwolinski, NRC/HQ Re: Technical Assistance to evaluate two fire protection issues at Quad Cities and Dresden (3 pages) |
| 3. | 11/18/92 | Memo to E. Merschoff, NRC/RII from G. Lainas, NRC/HQ Re: Turkey Point Plant, Unit 3 restart - (TIA 92-33) (3 pages) |
| 4. | 09/24/92 | Memo to E. Merschoff, NRC/RII from G. Lainas, NRC/HQ Re: Task Interface Agreement (TIA 92-03) concerning crack in Oconee decay heat removal (DHR) drop line (4 pages) |
| 5. | 01/11/93 | Memo to E. Merschoff, NRC/RII from G. Lainas, NRC/HQ Re: Close out of Task Interface Agreement (TIA) 92-33, Turkey point Unit 3 restart following Hurricane Andrew (2 pages) |
| 6. | 08/14/95 | Memo to J. Zwolinski, from E. Merschoff, re: Task Interface Agreement Technical Assistance Request, Service Water Pond Model Adequacy at Catawba Nuclear Power Station, (2 pages) |

FOIA No. 2000-0234

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

| NO. | DATE | DESCRIPTION/(PAGE COUNT) |
|---|---|---|
| 7. | 11/06/96 | Memo to E. Merschoff, NRC/RII from H. Berkow, NRC/HQ Re: Catawba Nuclear Station TIA 95-10, Standby Nuclear Service Water Pond analysis Model (1 page) |
| 8. | 10/28/97 | Memo to J. Johnson from F. Hebdon, re: NRR Resonse to TIA, Sequoyah Nuclear Plant - Offsite Power Technical Specifications (6 pages) |
| 9. | 12/11/98 | Memo to L. Plisco, NRC/RII from F. Hebdon, NRC/HQ Re: Task Interface Agreement (TIA 98-003) Crystal River Unit 3 low pressure injection system valve configuration (5 pages) |
| 10. | 03/11/98 | Memo to L. Plisco, NRC/RII from H. Berkow, NRC/HQ Re: Catawba Nuclear Station - response to TIA 97-14, frequency requirements for Quality Assurance Audits (3 pages) |
| 11. | 3/31/99 | Memo to L. Plisco, NRC/RII from C. Thomas, NRC/HQ Re: Response to Technical Assistance(TIA 97-015) Regarding the implementation of 10 CFR 50.65 - Browns Ferry Nuclear Plant Unit 1 (2 pages) **(This document is no longer proprietary)** |

50-213

January 20, 1984

LS05-84-01-028

MEMORANDUM FOR: Richard Starostecki, Director
Division of Project and Resident Programs
Region I

FROM: Darrell G. Eisenhut, Director
Division of Licensing

SUBJECT: HADDAM NECK FIRE BRIGADE QUARTERLY MEETINGS - TASK INTERFACE AGREEMENT 84-01

References:
(a) Memorandum of December 2, 1983, by T. T. Martin, DETP (Region I) to R. H. Vollmer, DE } not found *
(b) 10 CFR 50, Appendix R, Paragraph III.I.1.d.
(c) "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Control and Quality Assurance," June 14, 1977

In response to your request, we have reviewed Reference (a) which contained: a memorandum, dated December 2, 1983, to R. H. Vollmer, DE from T. T. Martin, DEPT (Region I); Inspection Report Number 50-213/83-22; and a Task Interface Agreement. We have also reviewed Reference (b): 10 CFR 50, Appendix R, Paragraph III.I.1.d and Reference (c): "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Control and Quality Assurance," June 6, 1977.

Based upon a review of these documents, we conclude that the licensee does not fully comply with the requirements of Appendix R, Paragraph III.I.1.d which states: "Regular planned meetings shall be held at least every three months for all brigade members to review changes in the fire protection program and other subjects as necessary."

To meet the above regulation, these meetings must not only be held quarterly, but each fire brigade member must attend the meetings. In the event that any fire brigade member is unable to attend the regularly scheduled meeting, one alternative is that a make-up meeting be held. A second alternative would be the showing of a videotape of the regularly scheduled meeting to any fire brigade member who missed this meeting.

The important point is that all fire brigade members receive proper training regarding any program changes or other subjects, whether this be at a regularly scheduled meeting or through an acceptable alternative.

* probably PDR doc# 8312200286

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B/1

Since this unresolved item is similar to an unresolved issue at the Three Mile Island Nuclear Plant, Unit 1 (TIA 83-108), we are in the process of preparing a generic letter to all licensees which addresses this issue as well as other related issues in the area of fire brigade training.

This completes NRR actions on TIA 84-01.

signed by
Darrell G. Eisenhut

Darrell G. Eisenhut, Director
Division of Licensing

Generic letter 86-10
issued 4/24/86

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

November 26, 1985

MEMORANDUM FOR: Charles E. Norelius, Director
Division of Reactor Projects
Region III

FROM: John A. Zwolinski, Director
BWR Project Directorate #1
Division of BWR Licensing, NRR

SUBJECT: TECHNICAL ASSISTANCE TO EVALUATE TWO FIRE PROTECTION ISSUES AT QUAD CITIES AND DRESDEN

REFERENCE: Memorandum from C. E. Norelius to D. G. Eisenhut, Subject: "Unresolved Fire Protection Issues at Quad Cities and Dresden Nuclear Power Stations. Not found

The referenced memorandum requests NRR's position concerning licensee actions relating to two fire protection issues, fire fighting strategies and fire drill frequency. The background information furnished by you and your regional positions are restated below and are followed by NRR's position.

1. Fire fighting strategies

Region III Background Information and Position

The Safety Evaluation supporting Quad Cities Amendments 52 and 49 for Units 1 and 2, respectively, apparently require the licensee to develop fire fighting strategies as described in Attachment No. 5 to the NRC guidance document, "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls and Quality Assurance." The licensee takes issue with this position indicating that fire protection reviewers verbally accepted their present program without specific fire fighting strategies. QCNPS has not established strategies for fighting fires in all safety-related areas and areas presenting a hazard to safety-related equipment. A similar program deficiency exists at DNPS; except, the DNPS license does not appear to require the establishment of strategies.

Establishing specific strategies and implementing them through fire brigade training and drills would greatly improve the fire brigade's ability to promptly extinguish fires which could degrade plant safety.

Region III recommends that the licensee be required to establish and implement specific strategies for fighting fires in all safety-related areas and areas presenting a hazard to safety-related equipment at DNPS and QCNPS. This is the Commission's position as presented in 10 CFR 50 Appendix R, paragraph III.K.12.



B/2

NRR Staff Position

The two plants do not have specific pre-planned strategies for each fire area. The licensee contends that the existing program was previously accepted, verbally, by the NRR reviewer. However, our SER states that the licensee will meet our guidelines. We have no basis for supporting the alleged verbal agreement rather than the SER.

2. Fire Drill Frequency

Region III Background Information and Position

License amendments No. 52 and No. 49 to licenses DPR-29 and DPR-30 and the QCNPS Fire Protection Safety Evaluation Report sections 6.0 and 6.2 apparently require the licensee to implement a fire brigade training program including fire drills as described in Attachment No. 2 to the NRC guidance document, "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls and Quality Assurance." The licensee takes issue with this position indicating that fire protection reviewers verbally accepted their present program with a requirement for having only eight fire drills per year. QCNPS has implemented procedural requirements to perform eight fire drills per year. In 1980, 38 percent of the fire brigade members (25 of 66) had attended no fire drills and fought no actual fires and an additional 38 percent (25 of 66) had only either attended one fire drill or fought one actual fire. A similar situation exists at DNPS; except, the DPNS license does not appear to require the more extensive training program. During 1979 and 1980, 84 percent (64 of 76) and 86 percent (66 of 77) of the fire brigade members at DNPS participated in fewer than two fire drills.

Region III does not believe that this drill frequency for a significant fraction of the plant fire fighters is sufficient to train the brigade members as an effective fire fighting team, assess the fire fighting tactics in the various plant areas and assure fire fighting preparedness at the plant.

Region III recommends that the licensee be required to implement a fire brigade training program at DNPS and QCNPS including fire drills at regular intervals not to exceed 3 months for each shift fire brigade. Each fire brigade member should participate in each drill, but must participate in at least two drills per year. At least one fire drill per year for each shift fire brigade must be unannounced to assess the plant fire fighting readiness. This is the Commission's position as presented in 10 CFR Part 50 Appendix R, paragraph III.I.3.b.

NRR Staff Position

The licensee does not perform fire drills with the frequency required by our guidelines. Again the licensee contends that the existing program was accepted verbally. However, the SER states that the licensee will meet our guidelines. We have no basis for supporting the alleged verbal agreement.

In conclusion, we agree with your recommendations for licensee actions and trust that our stated positions are responsive to your concerns.

Original signed by:

John A. Zwolinski, Director
BWR Project Directorate #1
Division of BWR Licensing, NRR

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

November 18, 1992

Docket No. 50-250

MEMORANDUM FOR: Ellis W. Merschoff, Director
Division of Reactor Projects, Region II

FROM: Gus C. Lainas, Assistant Director
for Region II Reactors
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

SUBJECT: TURKEY POINT PLANT, UNIT 3 RESTART - (TIA 92-33)

The purpose of this memo is to document our agreed-upon activities and responsibilities with respect to assuring the restart readiness of Turkey Point Unit 3 following its refueling outage and repair of the remaining damage from Hurricane Andrew. It supplements your memo to Stewart D. Ebneter on the same subject dated November 9, 1992.

NRC Region II has notified the Federal Emergency Management Agency (FEMA) Region IV of the planned restart of Unit 3 on November 4, 1992. NRR similarly notified FEMA Headquarters by letter from F. Congel to FEMA dated November 6, 1992. Both of these notifications solicited any FEMA concerns which need to be addressed prior to restart. Florida Power and Light Company (FP&L) has assured the staff that they are in contact with the cognizant Florida and Dade/Monroe County agencies regarding their restart plans, schedules, and ongoing activities.

The Region II and NRR staffs have reviewed the FP&L "Lessons Learned" listing which was presented to the NRC-Industry Lessons Learned Task Force and did not identify any significant additional corrective actions requiring attention prior to restart.

Contact: L. Raghavan
504-2019

9211250237/2F

B/3

The enclosure identifies the major Unit 3 restart verification activities to be accomplished by Region II and NRR. The principal NRR and Region II contacts for these activities are L. Raghavan and K. Landis, respectively.

(Original Signed By)

Gus C. Lainas, Assistant Director
for Region II Reactors
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosure:
Supplemented Review Plan

cc w/enclosure:
Charles W. Hehl, Region I
Edward G. Greenman, Region III
A. Bill Beach, Region IV
Ken Perkins, Region V

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verbal concurrence by M. Sinkule

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|---|---|---|---|---|---|
| NAME | ETana EH | LRaghavan:jkd | EMerschoff | HBerkow | GLainas |
| DATE | 11/17/92 | 11/17/92 | 11/18/92 | 11/18/92 | 11/18/92 |

ENCLOSURE

**REGION II - NRR TURKEY POINT UNIT 3 RESTART VERIFICATION ACTIVITIES**

1. Meet with licensee at site to review and confirm hurricane damage assessment and corrective actions and agree on restart items.

   **Responsibility:** NRR/Region II **Date:** November 16, 1992

   Document results in meeting summary.

   **Responsibility:** NRR **Date:** November 23, 1992

2. Confirmatory inspections by resident and regional inspectors of Unit 3 storm damage repair, to be documented in inspection reports.

   **Responsibility:** Region II **Date:** Inspection reports to be issued by January 4, 1993

3. Operational readiness inspections by resident inspectors to include operational readiness test program, selected surveillance tests, integrated safeguard test and selected system walkdowns.

   **Responsibility:** Region II **Date:** Inspection report to be issued by January 4, 1993

4. Teleconference with licensee and internal meeting, as required, to discuss restart readiness and verify that required restart activities have been completed.

   **Responsibility:** RII/NRR **Date:** November 24, 1992

5. Prepare memorandum to Region II Administrator with a copy to the NRR Director documenting the licensee's readiness to restart Unit 3.

   **Responsibility:** Region II **Date:** November 30, 1992

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON D.C. 20555

September 24, 1992

Docket No. 50-287

MEMORANDUM FOR: Ellis Merschoff, Director
Division of Reactor Projects, Region II

FROM: Gus Lainas, Assistant Director
for Region II Reactors
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

SUBJECT: TASK INTERFACE AGREEMENT (TIA 92-03) CONCERNING CRACK IN OCONEE DECAY HEAT REMOVAL (DHR) DROP LINE (TAC NO. M83247)

TIA 92-03 was issued to document the various NRC staff actions performed in relation to the crack which was identified by the licensee in the Oconee Unit 3 DHR drop line. The remaining open item was a review by the Materials and Engineering Branch (EMCB) of NRR of the failure analysis performed by B&W to determine if additional action was appropriate at Oconee or other facilities as a result of this failure.

EMCB has completed its review of the B&W failure analysis. As discussed in the enclosed memorandum, they agree with the conclusions reached by B&W in the failure analysis report. Since the analysis was limited to the cause of the specific event, there was insufficient information in the report to make a meaningful determination if additional action would be appropriate at Oconee or other facilities. However, Duke Power Company (DPC) completed a generic evaluation in their Problem Investigation Report (PIR). The DPC PIR indicated that the natural resonant frequency of the piping configuration was a dominant contributor to the failure. The piping configuration of the other Oconee units was sufficiently different to have natural resonant frequencies outside the range of concern. In addition, since the exact configuration is significant in determining the natural resonant frequency, no basis for a generic concern appears to exist.

This completes our efforts under TIA 92-03.

Gus Lainas, Assistant Director
for Region II Reactors
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosure
EMCB Evaluation of Failure
Analysis

B/4

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON D.C. 20555

JUL 0 7 1992

Docket No. 50-287

MEMORANDUM: Leonard A. Wiens, Project Manager
Project Directorate II-3
Division of Reactor Projects I/II

FROM: William Bateman, Acting Chief
Materials and Chemical Engineering Branch
Division of Engineering Technology

SUBJECT: EVALUATION OF B&W FINAL REPORT OF CRACKED LPI PIPE AT OCONEE-3 (TAC NO. M83247)

The staff has reviewed the Babcock & Wilcox's (B&W) final report," Cracked LPI Pipe at Oconee 3," dated January 1992. B&W performed for the Duke Power Company (DPC) a failure analysis of a cracked pipe section which was removed from the low pressure injection (LPI) system at Oconee-3. The subject piping (a twelve inch Schedule 10 pipe) in the LPI system was found to be leaking during a recent Oconee-3 start up. The throughwall crack was located at a half coupling weld joint, connecting a one inch Schedule 40 pipe to the twelve inch pipe. The one inch pipe consisted of a vertical run of seven inches to a relief valve (3LP-25) that weighed about 14 pounds. All piping was made of austenitic stainless steel. The length of the throughwall crack was about 2.5 inches on the outside diameter (OD) surface and about 1.5 inch on the inside diameter (ID) surface. Another shorter, partially throughwall crack was located adjacent to the throughwall crack. Various metallurgical examinations including liquid penetrant (PT), metallography, and scanning electron microscopy were performed on the pipe sections containing the throughwall crack. Based on the results of the failure analysis, B&W concluded that the root cause of the LPI pipe failure was due to mechanical fatigue. The loading on the joint is expected to be high cycle/low ... itude and the most likely source of such loading would be mechanical vibration of the LPI system piping. The staff agrees with B&W's conclusion because the reported characteristics of the failure mode as described below are typical of fatigue failure. (1) transgranular cracking, (2) no crack branching, (3) the presence of fatigue striations with micron size spacings on the fracture surface and (4) the initiation of cracks from the OD surface along the toe of the half coupling weld.

You requested the staff to determine if additional action would be appropriate for Oconee Unit 3 or other facilities. The staff cannot make a meaningful determination because there is not enough information in the failure analysis report, which only identified the failure mode and discussed the root cause of the failure. As a minimum, the licensee's submittal should provide a detailed discussion of the following issues pertaining to the referenced pipe failure event: (1) safety consequences of the failure event, (2) adequacy of the fix including plans for long term mitigation, and (3) generic nature of the failure event. Regarding the question of additional action at other facilities, it is apparent that inadequately supported geometries can lead to fatigue type failures.

This memorandum completes the work effort under TAC No. M83247.

William H Bateman

William Bateman, Acting Chief
Materials and Chemical Engineering Branch
Division of Engineering Technology

cc: B. D. Liaw
G. C. Lainas
D. B. Mathews

September 24, 1992

Docket No. 50-287

MEMORANDUM FOR: Ellis Merschoff, Director
Division of Reactor Projects, Region II

FROM: Gus Lainas, Assistant Director
for Region II Reactors
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

SUBJECT: TASK INTERFACE AGREEMENT (TIA 92-03) CONCERNING CRACK IN OCONEE DECAY HEAT REMOVAL (DHR) DROP LINE (TAC NO. M83247)

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This completes our efforts under TI 92-03.

/S/

Gus Lainas, Assistant Director
for Region II Reactors
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosure:
EMCB Evaluation of Failure
Analysis

DOCUMENT NAME: C:/TIA9203.OCO

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

January 11, 1993

MEMORANDUM FOR: Ellis W. Merschoff, Director
Division of Reactor Projects, Region II

FROM: Gus C. Lainas, Assistant Director for
Region II Reactors
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

SUBJECT: CLOSE OUT OF TASK INTERFACE AGREEMENT (TIA) 92-33, TURKEY POINT UNIT 3 RESTART FOLLOWING HURRICANE ANDREW (TAC NOs M84370/M84371)

All activities and responsibilities identified in TIA 92-33 with respect to assuring the restart readiness of Turkey Point Unit 3 following Hurricane Andrew have been completed as summarized below:

1. On November 16, 1992, NRR and Region II staffs held a meeting with the licensee at the Turkey Point site to review and confirm Hurricane Andrew damage assessment and corrective actions. The staff also toured the plant to review the progress of the restart activities. Based on the licensee's presentation, the staff agreed with the licensee's restart actions. The results of the meeting are documented in the meeting summary dated November 17, 1992. This completes item 1 in the TIA.

2. Region II confirmatory inspections relating to storm damage repairs and operational readiness inspections have been completed and are documented in Inspection Reports 50-250/251 92-28 and 92-30. This completes items 2 and 3 in the TIA.

3. On November 25, 1992, NRR and Region II staffs held a telephone conference with the licensee and reviewed its readiness to restart Unit 3. The teleconference is summarized in a Daily Highlight dated November 27, 1992 from L. Raghavan to T. Murley et al. This completes item 4 in the TIA.

4. The licensee's readiness to restart Unit 3 is documented in a memorandum dated November 27, 1992 from E. Merschoff to S. Ebneter. This completes item 5 in the TIA.



B/5

January 11, 1993



All the items assigned to NRR and Region II offices in the TIA are now complete and closed.

(Original Signed By)
Gus C. Lainas, Assistant Director for
Region II Reactors
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

cc: J. Partlow
S. Varga
G. Lainas
H. Berkow
M. Sinkule
K. Landis
L. Raghavan

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

August 14, 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 Projects

SUBJECT: TASK INTERFACE AGREEMENT (TIA 95-10) TECHNICAL ASSISTANCE REQUEST, SERVICE WATER POND MODEL ADEQUACY AT CATAWBA NUCLEAR POWER STATION

During a service water team inspection at Catawba in 1994, and during follow-up inspections in April and May 1995, concerns were raised as to the validity of the calculations and the model used by the licensee for predicting the safety performance of the standby nuclear service water pond (SNSWP) to mitigate the consequences of an accident. In response to some of these concerns, the licensee plans to submit a Technical Specification change raising the minimum pond level and thus increasing the minimum heat removal capacity of the pond. Although raising the minimum level may resolve a number of the concerns, some may not be addressed. Consequently, NRR needs to consider these concerns as part of their review of this Technical Specification change.

The concerns are as follows:

- The licensee does not incorporate or consider instrument inaccuracy (pond temperature or level) in their pond heat capacity evaluation. Without compensating for these errors in either the evaluation or surveillance procedure, the actual temperature and level of the pond can exceed the initial conditions established in the accident analysis.

- The NRC imposed a 2.4° F penalty to the licensee's pond model. This penalty was derived by taking the difference of the licensee's results and the NRC's results using a different model. Part of the reason for the disparity may lie in the mis-application of the licensee's pond model. Their model is based upon an MIT Department of Civil Engineering Report, "Analytical and Experimental Study of Transient Cooling Pond Behavior," dated January 1973. The report is predicated upon large deep draft cooling ponds. It is questionable whether the licensee's model is applicable given the shape and shallowness of the pond. However, no restrictions or prohibitions have been placed on how the licensee uses their model except to impose a 2.4° F penalty on the results. It is unclear whether this penalty is adequate.

B/4

- Pond performance in two cases may not have been fully evaluated. In one accident scenario, at four hours into the accident two service water pumps are used to mitigate the consequences of a LOCA on one unit with the other unit at power. In the other accident scenario, one service water pump is used to mitigate the consequences of a LOCA on one unit with the other unit in cold shutdown. In both accident scenarios flow to the pond would be significantly skewed to the short leg discharge path. The short leg is not far from the suction of the service water pumps. Therefore, a "short cycling" and uneven heat rejection to the pond would occur. Also, the team could not determine which accident would be the most bounding condition for pond performance. In the first accident, the heat loads would be higher but an equal thermal mixing was assumed for at least the first four hours. Whereas in the second accident, the flow would always be skewed, but the heat loads would be less.

- Recent service water flow testing in April and May 1995, indicate that each train of service water cannot achieve an equal flow split to the pond when operating in any accident configuration. Therefore, even during the first four hours of an accident, skewed flow would occur. The split is closer to 70/30 with 70 percent coming from the short leg. It is unclear how sensitive the NRC flow model used to establish the 2.4° F penalty at initial licensing is to unequal flow splits.

If you have any questions, your staff may wish to contact P. Kellogg at (404) 331-5594.

Docket Nos. 50-413, 50-414

cc: R. Cooper, RI
W. Axelson, RIII
J. Dyer, RIV
K. Perkins, WCFO
H. Berkow, NRR
S. Vias, RII
J. Barnes, RII
Docket/Central Files

ACC. NO. 9611130249/CF

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

November 6, 1996

MEMORANDUM TO: Ellis W. Merschoff, Director
Division of Reactor Projects
Region II

FROM: Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

SUBJECT: CATAWBA NUCLEAR STATION - TIA 95-10, STANDBY NUCLEAR SERVICE WATER POND ANALYSIS MODEL (TAC M95256 AND M95257)

By memorandum dated August 14, 1995, Region II requested NRR technical assistance (Task Interface Agreement, TIA 95-10) to address concerns regarding the validity of the calculations and model used by the licensee for predicting the safety performance of the Standby Nuclear Service Water Pond (SNSWP) to mitigate the consequences of an accident.

As a result, we requested assistance from the Office of Nuclear Material Safety and Safeguards (NMSS) for assistance in the hydrology area. The attached safety evaluation, performed by Richard Codell of NMSS, sets forth details of this review. We conclude that (1) raising the pond water level (approved by Amendments 152 and 144 on September 20, 1996) would have no detrimental effect on the performance of the pond; (2) peak pond temperature for a loss-of-coolant accident in one unit and normal shutdown in the other, under conditions of worst-case meteorology would be less than 100 °F; and (3) the pond could supply water below 92 °F for up to 12.5 hours. These analyses took into consideration the best estimates of heat load, meteorology and thermal hydraulic behavior of the pond.

Our review was based on information submitted by the licensee in a letter dated September 10, 1996. Accordingly, unless Region II objects, we plan to issue the attached safety evaluation to the licensee 2 weeks after the date of this memorandum.

If you have any questions, please contact the Project Manager, Peter Tam (301-415-1451). We consider our efforts complete on TIA 95-10.

Docket Nos. 50-413 and 50-414

Attachment: Safety evaluation

cc: R. W. Cooper, RI
E. Greenman, RIII
A. B. Beach, RIV
K. Perkins, RIV, WCFO

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001
October 28, 1997

MEMORANDUM TO: Jon R. Johnson, Director
Division of Reactor Projects
Region II

FROM: Frederick J. Hebdon, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Frederick J. Hebdon

SUBJECT: NRR RESPONSE TO TIA 94-021, SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2 - OFFSITE POWER TECHNICAL SPECIFICATIONS (TAC NOS. M93319 & M93320)

This is the Office of Nuclear Reactor Regulation (NRR) response to the request for technical assistance regarding the Sequoyah Nuclear Plant offsite power distribution system. The request was made via memorandum from Ellis W. Merschoff to John A. Zwolinski dated August 11, 1995. This response closes out Task Interoffice Agreement (TIA) No. 94-021.

During an inspection at Sequoyah in 1993 (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. The TIA requested that NRR to review various grid load studies and design calculations provided with the TIA and to reach a conclusion regarding whether a special Technical Specification (TS) should be requested by TVA to cover the contingency of the intertie transformer not being available. Specifically, The TIA aked the following questions:

1. Based on the new Transmission System Study and the new Common Station Service Transformers, does the plant have an acceptable immediate preferred offsite power source if the 500 kV to 161 kV intertie transformer is not operable? Does the 161 kV analysis demonstrate that the plant can achieve safe shutdown without the intertie transformer?

2. Should the plant's technical specifications be amended to require that [Limiting Condition for Operation] LCO 3.8.1.1, Action C, be entered following a loss of the intertie transformer?

NRR has completed its evaluation as summarized in the attachment. The staff concluded that the Sequoyah plant will not have an acceptable immediate preferred offsite power source when the 500 kV to 161 kV intertie transformer is out of service because the offsite source will not meet commitments (design description) specified in Update 12 of the Sequoyah Updated Final Safety Analysis Report (UFSAR). The 161 kV analysis, however, demonstrates that the plant will have sufficient capacity so that plant shutdown can be achieved when the intertie transformer is out of service. The staff also concluded that a requirement to amend the



B/

TS is not necessary to assure that LCO 3.8.1.1, Action C, will be entered following a loss of the intertie transformer. The basis for these conclusions is described in the attachment.

NRR understands that TVA is making changes (Update 13) to the UFSAR that may be related to the subject matter. TVA has opted to not submit Update 13 to the NRC until such time as 10 CFR 50.59 guidelines, including changes to the UFSAR, have been finalized and published by the NRC. This will probably not occur until the end of calendar 1997, at the earliest.

In the meantime, NRR's staff responses to your questions remain valid until such a time that new information is submitted for staff review. Any further review of this subject, if necessary, will be done under a separate TAC. We consider our response to TIA 94-021 complete and the corresponding TACs are closed.

If you have any questions regarding this response, please contact Ronald W. Hernan, the Sequoyah Project Manager.

Attachment: As stated

Docket Nos. 50-327 and 50-328

cc: C. Hehl, Region I
G. Grant, Region III
T. Gwynn, Region IV

TS is not necessary to assure that LCO 3.8.1.1, Action C, will be entered following a loss of the intertie transformer. The basis for these conclusions is described in the attachment.

NRR understands that TVA is making changes (Update 13) to the UFSAR that may be related to the subject matter. TVA has opted to not submit Update 13 to the NRC until such time as 10 CFR 50.59 guidelines, including changes to the UFSAR, have been finalized and published by the NRC. This will probably not occur until the end of calendar 1997, at the earliest.

In the meantime, NRR's staff responses to your questions remain valid until such a time that new information is submitted for staff review. Any further review of this subject, if necessary, will be done under a separate TAC. We consider our response to TIA 94-021 complete and the corresponding TACs are closed.

If you have any questions regarding this response, please contact Ronald W. Hernan, the Sequoyah Project Manager.

Attachment: As stated

original signed by F.Hebdon

Docket Nos. 50-327 and 50-328

cc: C. Hehl, Region I
G. Grant, Region III
T. Gwynn, Region IV

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## OFFICE OF NUCLEAR REACTOR REGULATION RESPONSE TO TIA 94-021 RELATING TO OFFSITE POWER FOR SEQUOYAH NUCLEAR PLANT UNITS 1 AND 2 DOCKET NOS. 50-327 AND 50-328

### BACKGROUND

At the Sequoyah Nuclear Plant, the Tennessee Valley Authority (TVA) replaced their preferred offsite power system's Common Station Service Transformers (CSSTs) with new transformers equipped with automatic load tap changers. In connection with this replacement, TVA revised their analysis that demonstrated that the 161 kV transmission network remains stable and thus available as a reliable offsite power supply to ensure safe shutdown of the Sequoyah units in the event of: (1) anticipated operational occurrences and accidents at the nuclear facility; or (2) anticipated contingencies on the transmission network such as the loss of the transformer that interties the 161 kV and 500 kV switchyards located near the Sequoyah plant.

By memorandum dated August 11, 1995, Region II requested the Office of Nuclear Reactor Regulation's (NRR's) review of the licensee's revised transmission network analysis. Region II specifically requested that this review include answers to the following questions:

1. Based on the new Transmission System Study and the new Common Station Service Transformers, does the plant have an acceptable immediate preferred offsite power source if the 500 kV to 161 kV intertie transformer is not operable? Does the 161 kV analysis demonstrate that the plant can achieve safe shutdown without the intertie transformer?

2. Should the plant's Technical Specifications be amended to require that [Limiting Condition of Operation] LCO 3.8.1.1, Action C, be entered following a loss of the intertie transformer?

To accomplish this review, NRR issued a task order (Task No. 102 under Contract No. NRC-03-95-026) to obtain the technical expertise of Scientech, Inc. Scientech reviewed TVA's revised transmission analysis and concluded, in part, that the immediate availability of an offsite power source could not be substantiated. The analysis indicated that the transmission network will be capable of providing adequate post event steady state voltage and frequency. Scientech considered the licensee's analysis to be acceptable; however, the analysis did not specifically address transient voltages which will occur during the transition from pre to post-event conditions. Transient voltages can exceed protective relay set points causing disconnection of offsite circuits. If offsite circuits are disconnected due to these transient voltages, the immediate availability of offsite power to safety loads will be lost when needed following an event. Thus, based on analysis which addressed only steady state conditions after an event, Scientech was not able to substantiate the immediate availability of offsite circuits.

Subsequently, by letters dated July 17, 1996, and June 2, 1997, TVA provided results of transient stability analysis for an undefined (normally anticipated) transmission network configuration. The results indicated that voltage recovery times are within the time limits required to ensure that protective relaying will not cause disconnection of offsite circuits for the following postulated transmission disturbances:

1. a 3-phase fault and a stuck breaker on either the 500 kV bus 1 or 2 or the 161 kV bus 1 or 2; or

2. a phase-phase-ground fault on the 161 kV side of the 500/161 kV intertie transformer bank and a stuck breaker in the 161 kV switchyard.

During a July 10, 1997, telephone conference call and subsequently by letter dated August 5, 1997, TVA restated that these transients are considered worst case conditions and thus encompass transient conditions that would be caused by simultaneous trip of both Sequoyah units plus simultaneous connection of required loads.

**Response to the first question -- Part 1:**

The answer is no. Based on analysis results, the Sequoyah plant will not have an acceptable immediate preferred offsite power source when the intertie transformer is out of service. When the transmission network is operating with the intertie transformer out of service, the Sequoyah plant will not meet the following design commitment and will, thus, not have an acceptable immediate preferred offsite power source.

> The eight 161-kV transmission lines connected to the 161-kV switchyard, the 500-161-kV intertie transformer bank, two 84 [megavolt-ampere reactive] MVAR capacitor banks for the 161-kV switchyard, and the five 500-kV transmission lines have sufficient capacity to supply the total required power to the plant's electrical auxiliary power system under normal, shutdown, and loss of coolant accident (LOCA) conditions for any single transmission contingency...
>
> (Ref: last paragraph on page 8.2-20 of the Updated Final Safety Analysis Report (UFSAR) amendment 12)

Current analysis results for operation with the intertie transformer out of service, documented on page 8.2-21 of UFSAR Amendment 12, indicates that if the capacitor bank becomes unavailable (as a single transmission system event) it will require 10 minutes for the system dispatcher to adjust the transmission network so that the remaining eight 161 kV transmission lines will be fully capable of providing adequate voltage and power. Thus, because analysis results indicate that the design commitment (defined above) will not be met, the Sequoyah plant will not have an acceptable immediate preferred offsite power source when the intertie transformer is out of service.

The answer to the above question has been predicated on our interpretation of the Sequoyah plant's design as described in the UFSAR (defined above) and accepted by the staff's Safety Evaluation Report (SER), NUREG-0011, for issuing the Sequoyah operating licenses. With respect to capacity and capability following a LOCA (i.e., the GDC 17 immediate access circuit), we believe that our SER acceptance was based on a steady state transmission network analysis which demonstrated network stability following any single transmission contingency. In addition, because of a safety system design criteria which permits extensive sharing of systems between units and UFSAR commitments relating to capacity of the offsite system following LOCA, we believe our SER acceptance was based on network stability assuming simultaneous tripping of both units and loading of safety buses for both units as a result of a LOCA in one unit.

**Response to the first question** -- **Part 2:**

The answer is yes. The 161 kV analysis demonstrates that the plant can achieve safe shutdown without the intertie transformer. Based on Scientech's review of load flow studies (defined below) for an out of service intertie transformer and based on transient analysis results subsequently provided by the licensee, we agree that the licensee's analysis results demonstrate that the plant can achieve safe shutdown without the intertie transformer.

> Load flow studies have been performed for the normal power flow around the Sequoyah 500- and 161-kV buses. Studies have been performed for power flow assuming a design basis event on one unit, and orderly shutdown of the other unit and one of the following: (1) a normal transmission network, (2) the loss of the 500-kV intertie transformer bank, (3) the loss of the 161-kV bus 1, (4) the loss of the 161-kV bus 2, (5) the loss of the 500-kV bus 1, (6) the loss of the 500-kV bus 2, (7) the loss of the largest generating unit, or (8) the loss of the most critical 500-kV transmission line.
>
> (Ref: UFSAR page 8.2-21)

**Response to the second question**

The answer is no. It is not necessary to amend the plant's Technical Specifications (TS) to require that LCO 3.8.1.1, Action C, be entered following a loss of the intertie transformer. The TS require two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system that are each operable. Operability is predicated on compliance of these circuits with the commitments that have been accepted by the staff in its SER and have been described/analyzed in the plant's UFSAR as meeting the requirements of GDC 17 (i.e , the TS basis). For Sequoyah, the commitment, in part, as defined in the first paragraph of section 8.2.2 of the UFSAR and accepted by the staff in its SER, specifies that these circuits have sufficient capacity to supply the total required power to the plant's electrical auxiliary power system under normal, shutdown, and LOCA conditions for any single transmission contingency. When the 500 kV to 161 kV intertie transformer is out of service (or when any other component of the offsite system is out of service), we believe that it is the licensee's responsibility to assure continued system operability. If analysis does not support sufficient capacity following a LOCA for any transmission system contingency (or analysis is not available), we believe the licensee is obligated in accordance with their TS basis to enter the appropriate TS LCO. Based on documented information provided by the licensee, it appears that an out of service intertie transformer, for example, may create an operating configuration for which appropriate analysis is not available; thus, anytime the intertie transformer is out of service, we believe the licensee is obligated in accordance with their TS basis to enter the appropriate TS LCO. An out of service intertie transformer can, thus, be considered inherently included in the TS. It is not considered practicable to include in the TS one of many components that may or may not cause the loss of operability based on continuously changing system operating conditions and analysis results or based on the unavailability of appropriate analysis.

MEMORANDUM TO: Loren R. Plisco, Director
Division of Reactor Projects, RII

December 11, 1998

9312160197 ...

FROM: Frederick J. Hebdon, Director (Original signed by J. Zwolinski for)
Project Directorate II-3
Division of Reactor Projects - I/II

SUBJECT: TASK INTERFACE AGREEMENT (TIA 98-003) CRYSTAL RIVER UNIT 3 LOW PRESSURE INJECTION SYSTEM VALVE CONFIGURATION (TAC NO. MA2125)

By memorandum dated June 12, 1998, the Division of Reactor Projects, Region II, requested the assistance of NRR in evaluating certain aspects of the Crystal River Unit 3 (CR-3) low pressure injection (LPI) system design. Specifically, NRR was asked to:

1. Evaluate the licensee's conclusion's as it relates to the normal standby position of the LPI discharge valves, ascertain the appropriate normal position for these valves, and take the appropriate licensing action if that position is normally open.

2. Ascertain whether the present design of the LPI system, specifically the location of the flow sensing device providing feedback to valves DHV-110 or DHV-111, renders the LPI system inoperable in that operator action, instead of the flow controllers, is necessary to preclude possible LPI pump run out conditions in the "piggy back" mode of operation.

The NRR Reactor Systems Branch (SRXB) reviewed a number of documents that relate to this issue. The Attachment lists these documents and provides our responses to the two questions. SRXB concluded that there was no technical concern with the normally closed position of the LPI discharge valves nor was there a basis for taking licensing action to change the valve position. In addition, our review of the issue indicated that the normally closed position for these valves was consistent with the CR-3 licensing basis. With regard to the second question, SRXB concluded that reliance on manual action during the "piggy back" mode of operation to be acceptable.

If you have questions concerning the positions in the attachment, please contact Len Wiens at (301) 415-1495.

Docket No. 50-302

Attachment: SRXB Evaluation

cc: C. W. Hehl, Region I
G. E. Grant, Region III
T. P. Gwynn, Region IV

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

December 11, 1998

MEMORANDUM TO: Loren R. Plisco, Director
Division of Reactor Projects, RII

FROM: Frederick J. Hebdon, Director
Project Directorate II-3
Division of Reactor Projects - I/II

SUBJECT: TASK INTERFACE AGREEMENT (TIA 98-003) CRYSTAL RIVER UNIT 3 LOW PRESSURE INJECTION SYSTEM VALVE CONFIGURATION (TAC NO. MA2125)

By memorandum dated June 12, 1998, the Division of Reactor Projects, Region II, requested the assistance of the NRR in evaluating certain aspects of the Crystal River Unit 3 (CR-3) low pressure injection (LPI) system design. Specifically, NRR was asked to:

1. Evaluate the licensee's conclusion's as it relates to the normal standby position of the LPI discharge valves, ascertain the appropriate normal position for these valves, and take the appropriate licensing action if that position is normally open.

2. Ascertain whether the present design of the LPI system, specifically the location of the flow sensing device providing feedback to valves DHV-110 or DHV-111, renders the LPI system inoperable in that operator action, instead of the flow controllers, is necessary to preclude possible LPI pump run out conditions in the "piggy back" mode of operation.

The NRR Reactor Systems Branch (SRXB) reviewed a number of documents that relate to this issue. The Attachment lists these documents and provides our responses to the two questions. SRXB concluded that there was no technical concern with the normally closed position of the LPI discharge valves nor was there a basis for taking licensing action to change the valve position. In addition, our review of the issue indicated that the normally closed position for these valves was consistent with the CR-3 licensing basis. With regard to the second question, SRXB concluded that reliance on manual action during the "piggy back" mode of operation to be acceptable.

If you have questions concerning the positions in the attachment, please contact Len Wiens at (301) 415-1495.

Docket No. 50-302

Attachment: SRXB Evaluation

cc: C. W. Hehl, Region I
G. E. Grant, Region III
T. P. Gwynn, Region IV

14B21

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

December 1, 1998

MEMORANDUM TO: Leonard A. Wiens, Project Manager
Project Directorate II-3
Division of Reactor Projects

FROM: Eric W. Weiss, Chief
Pressurized Water Reactor Section
Reactor Systems Branch
Division of Systems Safety and Analysis

SUBJECT: TASK INTERFACE AGREEMENT (TIA) 98-003 CRYSTAL RIVER 3, LOW PRESSURE INJECTION EMERGENCY CORE COOLING SYSTEM VALVE CONFIGURATIONS (TAC No. MA2125)

In TIA 98-003, dated June 12, 1998, Region II requested NRR's position on the adequacy of aspects of the Crystal River Unit 3 (CR-3) low pressure injection (LPI) system. The TIA specifically asks two questions. The first question relates to the normal position of the LPI discharge valves. The second question relates to the need to rely on manual operator action to throttle LPI flow while in the LPI to high pressure injection (HPI) or "piggy back" flow path configuration.

Issue 1 of TIA 98-003 requests that NRR, "ascertain the appropriate normal position of the LPI discharge valves and take appropriate licensing action if that position is normally open." The Reactor Systems Branch (SRXB), through interactions with the licensee and a review of some of the documented information on the subject, concludes that the normally closed position of the LPI discharge valves is not inappropriate and that no licensing action is warranted. The staff bases this conclusion on the following information. Although some of the correspondence during the original licensing process, referenced in the TIA, indicates that the licensee stated that valve would be normally open, the normal valve position was never changed to be open. A letter from the licensee, dated October 22, 1998 with the subject, "Low Pressure Injection Engineering Study," stated that maintaining the valves closed is consistent with licensing basis. The staff attempted to verify that by reviewing some of the licensing documentation. In 1980, the staff had a generic safety concern with regard to the likelihood of an intersystem loss-of-coolant accident (LOCA). The staff issued a Generic Letter (GL) dated February 23, 1980 and requested licensees evaluate specific vulnerable configurations associated with the likelihood of an intersystem LOCA. The correspondence associated with the licensee response to that GL and the subsequent NRC order confirm that at that time the valve position was normally closed. The staff concluded, with the normally closed valves, the high pressure/low pressure isolation with additional leakage testing specifications was adequate. The normally closed discharge valve reduces the likelihood of an intersystem LOCA. Additionally, this plant configuration, with the LPI discharge valves closed is an acceptable configuration described in section 6.3 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear

Contact: Christopher Jackson, DSSA/SRXB
415-2947

Power Plants," since an engineered safeguards actuation system signal opens these closed discharge valves and assures the required emergency core cooling system function following an accident while meeting the single failure criteria. As a result, the staff has not identified a technical concern with the normally closed position of the LPI discharge valves and does not believe there is a basis for taking licensing action to change the valve position.

Issue 2 of TIA 98-003 requests that NRR "ascertain whether the present design of the ECCS, specifically the location of the flow sensing device providing feedback to valves DHV-110 or DHV-111, renders the LPI system inoperable in that operator action instead of the flow controllers is necessary to preclude possible LPI pump run out conditions in the 'piggy back' mode of ECCS operation." The determination of LPI system operability is the responsibility of the licensee rather than the staff, however, the staff has reviewed the information in the TIA and some additional supporting information. Although the LPI flow controllers were intended to prevent the operators from needing to manually throttle the LPI flow, the NRC has accepted manual operator action to initiate sump recirculation. The CR3 design already requires manual swapover to the sump recirculation flow path. As a result, relying on manual operator action under these circumstances is acceptable. The staff has reviewed Inspection Report (IR) No. 50-302/98-02 where Region II concluded that, "the licensee had adequate technical justification for operating in the piggyback mode." Although SRXB has not evaluated the operators ability to perform these specific tasks, the staff finds the Region II conclusion reasonable, based, in part, on the operators ability to establish the necessary LPI flow by manually throttling the necessary valves during a simulator scenario (also described in the IR). Additionally, the licensee has indicated that they intend to add additional valves to the LPI system that will preclude the operators need to continually manually throttle the LPI flow. They would only have to reset the flow at which the flow controller regulates flow. Although this modification has not been reviewed by the staff, it should enhance the system and reduce the reliance on the operators to complete the safety function. In conclusion, although the Standard Review Plan and staff practice emphasize the minimization of required operator actions, it is recognized and accepted that establishment and maintenance of ECCS sump recirculation requires manual operator action.

This completes SRXB action on TAC No. MA2125.

References:

1. Memorandum, Plisco, Loren, "Task Interface Agreement (TIA 98-003) Crystal River Unit 3: Low Pressure Injection Emergency Core Cooling System Valve Configurations," dated June 12, 1998.

2. Letter, Holden, J. J., "Low Pressure Injection Engineering Study," dated October 22, 1998.

3. Generic Letter to All LWR Licensees, Eisenhut, D. G. , "LWR Primary Coolant System Pressure Isolation Valves," dated February 23, 1980.

4. Letter, Bright, Ronald, M., "Crystal River Unit No. 3, Docket No. 302, Operating License No. DPR-72, Letter to All LWR Licensees from D. G. Eisenhut dated 2-23-80 - LWR Primary Coolant System Pressure Isolation Valves," dated March 14, 1980.

5. Letter, Stolz, John, F., "Order for Modification of License Concerning Primary Coolant System Pressure Isolation Valves," dated April 20, 1981.

6. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," dated June 1987.

7. Inspection Report No. 50-302/98-02, dated March 16, 1998.

ACC. No. 9803160236/CF

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

March 11, 1998

MEMORANDUM TO: Loren Plisco, Director
Division of Reactor Projects
Region II

FROM: Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

H. N. Berkow

SUBJECT: CATAWBA NUCLEAR STATION - RESPONSE TO TIA 97-14, FREQUENCY REQUIREMENTS FOR QUALITY ASSURANCE AUDITS (TAC NOS. M98929 AND M98930)

Fm PDR

By memorandum dated June 5, 1997, Region II requested that NRR perform a technical review of Catawba's Quality Assurance (QA) Topical Report, which was prepared for Duke Power facilities in general. The purpose of the review was "to determine if, for Catawba specifically, but all Duke facilities in general, the changes to the report of only placing an audit frequency on Category 1 [defined by the licensee as safety-related components and services] functions met the intent stated in the licensee's justification for removing all the audit frequencies from the TS [Technical Specifications] Section 6.5.2.9, by Amendment Nos. 96 and 90." Those amendments dealt with the relocation of audit frequency provisions from the Technical Specifications into the QA program. Currently, QA audits are only required for the licensee-defined QA Category 1 functions.

The safety evaluation attached to this memorandum provides details prepared by the NRR Quality Assurance, Vendor Inspection , and Maintenance Branch. We have concluded that the licensee modified its QA program in accordance with the licensing submittals provided as part of License Amendments 96 and 90.

We would like to note that for certain nonsafety-related audits, the Nuclear Safety Review Board remains responsible for the conduct of the associated audits in accordance with Technical Specification provisions.

This completes our efforts on the subject TIA. If you have any questions, please contact the Catawba project manager, Peter Tam (301-415-1451).

Docket Nos. 50-413 and 50-414

Attachment: Safety Evaluation

cc w/att: C. W. Hehl, RI
G. E. Grant, RIII
T. P. Gwynn, RIV

B/10

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

CATAWBA NUCLEAR STATION

DOCKET NOS. 50-413 AND 50-414

FREQUENCY REQUIREMENTS FOR QUALITY ASSURANCE AUDITS

(REGION II TIA 97-014)

## I.0 INTRODUCTION

By memorandum, J. R. Johnson to H. N. Berkow, dated June 5, 1997, Region II requested that NRR perform a technical review of Catawba's Quality Assurance (QA) Topical Report, which was prepared for Duke Energy Corporation (previously Duke Power Company) facilities in general. The purpose of the review was "to determine if, for Catawba specifically, but all Duke facilities in general, the changes to the report of only placing an audit frequency on Category 1 [defined by the licensee as safety-related components and services] functions met the intent stated in the licensee's justification for removing all the audit frequencies from the TS [Technical Specifications] Section 6.5.2.9, by Amendment Nos. 96 and 90." Those amendments dealt with the relocation of audit frequency provisions from the TS into the QA program. Currently, QA audits are only required for the licensee-defined QA Category 1 functions.

The NRR Quality Assurance, Vendor Inspection, and Maintenance Branch (HQMB) has performed a review of the Catawba QA Topical Report.

## 2.0 BACKGROUND

Technical Specification (TS) Section 6.5.2.9 had originally stated that audits performed under the cognizance of the Nuclear Safety Review Board (NSRB) would be performed at specific frequencies. By letter dated December 18, 1991, the licensee proposed to delete the prescriptive audit frequencies in the TS; however, the QA program commitments for the conduct of audits was to be modified. Specifically, the licensee's letter stated:

> Audit frequencies are being deleted here but in the revised QA Topical we are preparing the following statement, using SRP [Standard Review Plan] 17.3 guidance on planned and periodic assessment scheduling and resource allocation: "Audits of selected aspects of operational phase activities are performed with a frequency commensurate with safety significance and in such a manner as to assure that an audit of all safety-related functions is completed within a period of two (2) years. The audit system is reviewed periodically and revised as necessary to assure coverage commensurate with the current and planned activities."

The staff's associated Safety Evaluation (SE) stated that "audit frequency requirements are now addressed in the Duke Quality Assurance Topical and are performance based on the safety significance and extent of activities."

The Duke QA program 17.3.3.2.2 states that:

> Audits of selected aspects of operational phase activities are performed with a frequency commensurate with safety significance and in such a manner as to ensure that an audit of all QA Condition 1 functions is completed within a period of two (2) years. The audit system is reviewed periodically and revised as necessary to assure coverage commensurate with current and planned activities.

Further, the Introduction of the Duke QA program defined QA Condition 1 as:

> QA Condition 1 covers those systems and their attendant components, items, and services which have been determined to be nuclear safety related. These systems are detailed in the Safety Analysis Report applicable to each nuclear station. The Topical report applies in its entirety to systems, components, items, and services identified as QA Condition 1.

We conclude that the licensee modified its QA program in accordance with the licensing submittals provided to support License Amendment Nos. 96 and 90, as accepted by the staff's SE. There was no licensee commitment to relocate explicit audit frequency provisions for other than safety-related audits, nor was a relocation of all audit frequencies a condition of the staff's SE. The Duke QA program provides for a graded application of quality controls based on safety significance (QA Conditions 1 through 4).

For the categories of nonsafety-related audits (such as for QA Condition 3 fire protection area) the licensee is still encumbered with implementing the TS provisions. The NSRB remains responsible for (1) review of Quality Verification Department audits relating to station operations and actions taken in response to those audits (Section 6.5.2.8.i), and (2) audits of fire protection (Sections 6.5.2.9.g and .h). The NSRB would need to be able to justify the adequacy of the audit periodicity for nonsafety-related fire protection audits that are under their cognizance.

## 3.0 CONCLUSION

The Duke QA program was modified in accordance with the licensee's submittals associated with License Amendments 96 and 90 that resulted in a relocation of audit frequency provisions.

Principal Contributors: Robert Gramm
Edward J. Ford

Date: March 11, 1998

JUNE 5, 1997

MEMORANDUM TO: Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects I/II

FROM: Jon R. Johnson, Director
Division of Reactor Projects

ORIGINAL SIGNED BY
R. CRLENJAK FOR:

SUBJECT: TASK INTERFACE AGREEMENT (TIA 97-014) CATAWBA FREQUENCY REQUIREMENTS FOR QUALITY ASSURANCE AUDITS

During a routine inspection of the Catawba facility, a Region II inspector noted that Section 6.5.2.9 of the Technical Specifications (TS) had been revised in 1992 to eliminate the frequency requirements for QA audits.

The attachment (pages 2-7) to the licensee's letter, dated December 18, 1991, stated that audit frequencies were being deleted from the TS but the Duke QA Topical Report was to be revised to specify that audits of selected aspects of the operational phase activities were to be performed with a frequency commensurate with the safety significance and in such a manner that audits of all safety related functions would be completed within a period of two years.

NRC's letter dated May 7, 1992, issued Amendment Nos. 96 and 90 for Catawba Units 1 and 2, respectively, and revised the TS to eliminate the audit frequencies previously specified by TS, Section 6.5.2.9, of the specifications since these frequencies were to be addressed by the Duke QA Topical Report.

Subsequently, the QA Topical Report was revised to specify an audit frequency of two years only for QA Condition 1 (safety related components and services). Audit frequencies were not specified for QA Conditions 2, 3, and 4 functions.

TS, Section 6.5.2.9, identifies 13 activities requiring audits. Most of these activities are not Category 1 functions and do not meet the two year audit frequency requirements of the QA Topical Report. The audit frequencies for the Security and Emergency Programs are specified by other licensing documents. However, it appears that audits of the remaining functions will only be performed at the discretion of the licensee. We believe that the QA Topical Report change did not meet the intent of the justification provided in TS Amendment Nos. 96 and 90.

A review of the TSs for the other Duke facilities indicated that the audit frequency requirements at these two sites had also been removed from the TS.

Attached are portions of the licensee's December 18, 1991, submittal, NRC letter of May 7, 1992, McGuire TS, Section 6.5.2.9, Oconee TS, Section 6.1.3.4, and Duke QA Topical Report, Sections 17.0 and 17.3.3.2.3.

Herbert N Berkow 2

We request that NRR perform a technical review of the QA Topical Report for the Duke facilities to determine if, for Catawba specifically, but all Duke facilities in general, the changes to the report of only placing an audit frequency on Category 1 functions met the intent stated in the licensee's justification for removing all the audit frequencies from TS Section 6.5 2.9 by Amendment Nos. 96 and 90

This issue has been discussed with Peter Tam, NRR/DRPE/PD II-2, and Edward Ford, NRR/DRCH/HQMB. If you have any questions please contact Paul Fredrickson at (404) 562-4667 or Bill Miller at (404) 562-4673

Attachments: 1. Duke's letter dated December 18, 1991, Pages 6-10, 6-11, 2-6 and 2-7
2. NRC's letter dated May 7, 1992, Pages 6-10, 6-11 and 3
3. McGuire TS Section, Pages 6-10 and 6-11
4. Oconee TS Section, Page 6.1-5
5. Duke QA Topical Report, Pages 17-1, 17-42 and 17-43

cc w/attachments.
G. Edison, NRR PM
C. Hehl, RI
G. Grant, RIII
T. Gwynn, RIV
J. Lieberman, OE
J. Barnes, RII

(*) SEE PREVIOUS PAGE FOR CONCURRENCES

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OFFICIAL RECORD COPY [illegible]

Enclosed are portions of the licensee's December 18, 1991 submittal, NRC letter of May 7, 1992, McGuire TS Section 6 5 2 9, Oconee TS Section 6 1 3 4 and Duke QA Topical Report Sections 17 0 and 17 3 3 2 3

We request that NRR perform a technical review of the QA Topical Report for the Duke facilities to determine if, for Catawba specifically, but all Duke facilities in general, the changes to the report of only placing an audit frequency on Category 1 functions met the intent stated in the licensee's justification for removing all the audit frequencies from TS Section 6 5 2 9 by Amendment Nos 98 and 90

This issue has been discussed with Peter Tam, NRR/DRPE/PD II-2, and Edward Ford, NRR/DRCH/HOMB. If you have any questions please contact Paul Fredrickson at (404) 562-4667 or Bill Miller at (404) 562-4673

Attachments

1 Duke's letter dated December 18, 1991, Pages 6-10, 6-11, 2-6 and 2-7
2 NRC's letter dated May 7, 1992, Pages 6-10, 6-11 and 3
3 McGuire TS Section Pages 6-10 and 6-11
4 Oconee TS Section Page 6 1-5
5 Duke QA Topical Report, Pages 17-1, 17-42 and 17-43

cc w/attachments
O Edison, NRR PM
R Cooper, RI
W Axelson, RII
J Dyer, RIV
J Barnes, RII

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We find these changes acceptable as they reflect the revised organization and the reassignment of responsibilities. The change in approval authority meets the appropriate acceptance criteria of Section 13.5.1 of NUREG 0800, the Standard Review Plan.

h. Section 6.5.2 - Nuclear Safety Review Board (NSRB) - The titles Vice President, Nuclear Production, Nuclear Production Department, Quality Assurance Department, and Executive Vice President, Power Group have been replaced to reflect the revised organization. In Subsection 6.5.2.2 the qualification requirements for NSRB members has been revised to allow, in special cases, an individual with ten years experience in a specific technical area. In Subsection 6.5.2.9, Audits, the licensee has relocated the audit frequency from the audits required by this subsection. Audit frequency requirements are now addressed in the Duke Quality Assurance Topical and are performance based on the safety significance and extent of the activities except those for the Emergency and Security Plans as discussed below.

We find these changes acceptable as they reflect the revised organization and reassignment of responsibilities, the appropriate acceptance criteria of Section 13.4 of NUREG 0800, the Standard Review Plan, and the commitment to performance based audits in their revised Quality Assurance Topical Report, a document controlled in accordance with 50.54(a).

i. Section 6.6 - Reportable Event Action - The review of reportable events has been revised to reflect the new titles in the revised organization.

We find these changes acceptable as they reflect the revised organization.

j. Section 6.7 - Safety Limit Violation - The titles in this section have been revised to reflect the revised organization.

We find these changes acceptable as they reflect the revised organization.

k. Section 6.8 - Procedures and Programs - The approval authority for procedures and temporary changes to procedures has been revised by deleting specific titles and specifying a predesignated level of management.

We find this change acceptable as it meets the appropriate acceptance criteria of Section 13.5.1 of NUREG 0800, the Standard Review Plan.

We find the above changes to the Administrative Controls Section of the Technical Specifications, as described in the DPC letter dated December 18, 1991, and as revised by letter dated February 17, 1992, acceptable.

Changes reviewed
against amendments 96/90
5-27-92

DUKE POWER

FOR INFORMATION ONLY

December 10, 1991

U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

ATTENTION: Document Control Desk

SUBJECT: Catawba Nuclear Station
Docket Nos. 50-413 and 50-414
Proposed Technical Specification Changes

Gentlemen:

Pursuant to 10 CFR 50.4 and 50.90, attached are proposed license amendments to Appendix A, Technical Specifications, of Facility Operating Licenses NPF-35 and NPF-52 for Catawba Nuclear Station Units 1 and 2, respectively.

Effective November 1, 1991 Duke Power Company implemented a reorganization that essentially decentralized the corporate management of nuclear activities to each of the nuclear sites. In meetings with your staff on October 30, 1991, we discussed the new organizational structure and its resultant benefits on overall plant safety and efficiency.

At that time we also discussed our plans to revise the documents that assigned specific corporate responsibilities and functions. We are in the process of revising the Duke Quality Assurance (QA) Topical Report, using the guidance provided in Standard Review Plan (SRP) Section 17.3. We will soon be issuing as a special update to each FSAR, a revised Chapter 13, "Conduct of Operations".

As soon as we have the revised FSAR, the revised QA Topical Report and the Technical Specification revisions in place, we can proceed with the revision of other administrative documentation. Your timely consideration of this request will certainly help us hasten the completion of this transition.

Attachment 1 contains the proposed Technical Specification changes. Attachment 2 contains the summary and justification for the proposed changes. Pursuant to 10CFR 50.91, Attachment 3 provides the analysis performed in accordance with the standards contained in 10CFR 50.92 which concludes that the proposed amendments do not involve a Significant Hazards Consideration. Duke Power is forwarding a copy of this amendment request application and No Significant Hazards Consideration analysis to the appropriate South Carolina state officials. The proposed amendments have been reviewed and have

## 17.3.3 SELF ASSESSMENT

### 17.3.3.1 Methodology

The Self-Assessment process encompasses internal and corporate audits, independent review committee activities, in-plant reviews, and other independent assessments. This process is to confirm to management that activities affecting quality comply with the quality assurance program and that the quality assurance program has been implemented effectively. These functions are directed by the Manager, Nuclear Assessment & Issues Division and the Managers of Safety Assurance. The assessment activities are performed in accordance with instructions and procedures by organizations independent of the areas being assessed. Organizations performing self-assessment activities are technically and performance oriented, with the primary focus on the quality of the end product and a secondary focus on procedures and processes.

### 17.3.3.2 Assessment

#### 17.3.3.2.1 Nuclear Safety Review Board

The Senior Vice President, Power Generation Group, appoints a Nuclear Safety Review Board (NSRB) to serve as a nuclear safety review and audit backup to the normal operating organization. The Nuclear Safety Review Board reviews proposed tests and experiments, proposed station modifications, and proposed changes to procedures, when such involve an unreviewed safety question. Also, the Board reviews reportable occurrences and violations of a station's technical specifications and makes recommendations to prevent recurrence. Functions, operations and responsibilities of the NSRB are detailed in Chapter 6 of the technical specifications for each station.

#### 17.3.3.2.2 Plant Operations Review Committee

The Site Vice President appoints a Plant Operations Review Committee (PORC) to review selected nuclear safety related issues. The PORC is composed of specified senior members of the site management team most responsible for the safe and reliable operation of the station. The PORC also reviews the effectiveness of corrective actions taken for specified reportable events.

#### 17.3.3.2.3 Internal Audits

Duke's Quality Assurance Program requires a comprehensive system of planned and periodic internal audits for all phases of station operations and supporting activities.

All organizational units conducting quality assurance activities are evaluated with a system of audits. These audits are performed to determine the effective implementation of all applicable criteria of 10CFR 50, Appendix B. Periodic audits of activities or records of processes (e.g., welding, maintenance, development of design, record management, or system testing), to verify compliance and effectiveness of the implementation of the Quality Assurance Program are performed. Internal audits are initiated under the direction of the Manager, Regulatory Audits. The Manager, Nuclear Assessment and Issues Division may initiate special audits or expand upon the scope of an existing audit. The scope of each audit is determined by the responsible Lead Auditor, under the direction of the Manager, Regulatory Audits Section. Additionally, the scope of audits performed under the cognizance of the Nuclear Safety Review Board (NSRB) are reviewed for compliance with NSRB requirements by the NSRB staff. The lead auditor directs the audit team in developing checklists, instructions, plans and in the performance of the audit. The audit shall be conducted in accordance with checklists; the scope may be expanded upon by the audit team during the audit, if needed. One or more persons comprise an audit team, one of whom shall be qualified lead auditor.

Audits of selected aspects of operational phase activities are performed with a frequency commensurate with safety significance and in such a manner as to assure that an audit of all QA Condition 1 functions is completed within a period of two (2) years. The audit system is reviewed periodically and revised as necessary to assure coverage commensurate with current and planned activities.

The audit team concludes with a post-audit conference between the audit team and responsible management. The conference includes a brief discussion of audit results, including any deficiencies and recommendations. The audit results are documented in a report.

Within thirty (30) days of the post-audit conference, a report is issued to the responsible management with copies sent to the Vice President of the audited Site or department and other management as appropriate.

Within thirty days after receipt of the audit report, responsible management replies in writing to the Manager, Regulatory Audits Section, describing corrective action and an implementation schedule. The established electronic corrective action process may be used to convey this information. When necessary, after receipt of the management reply, a re-evaluation is made to verify implementation of corrective action. This re-evaluation is documented. The audit is closed with a letter to the responsible management. All pertinent correspondence, checklists, and reports related to the audit are filed.

Audit data are analyzed and the resulting reports on the effectiveness of the QA program, including any quality problems, are reported to management through the Integrated Safety Assessments, for review and assessment. This data is also used to modify the audit schedule as necessary to assess potential weaknesses.

**17.3.3.2.4 Safety Assurance**

Safety Assurance, through the Safety Review Group, and Regulatory Compliance, monitors the day to day and overall performance of each nuclear station.

The Safety Review Group investigates significant occurrences and problems to determine the root cause(s) and to identify actions necessary to prevent recurrence. The Safety Review Group also performs in-plant reviews including checking documents, records, and work in progress to determine that quality assurance requirements are being properly implemented. Work in progress includes such activities as welding, maintenance, system testing, station operation, station modifications, refueling, and record management. These investigations and reviews are documented in reports and submitted to Management, NRC, and other authorities as appropriate. The Safety Review Group also coordinates the development of corrective actions for significant occurrences and problems.

The Regulatory Compliance Group is responsible for the preparation, issue, and maintenance of all site licensing documents; providing site personnel with interpretations on the licensing documents, the preparation and submittal of violation responses, and coordination of NRC inspection activities on site.

# 17. QUALITY ASSURANCE

## INTRODUCTION

Duke Power Company maintains full responsibility for assuring that its nuclear power plants are designed, constructed, tested and operated in conformance with good engineering practices, applicable regulatory requirements and specified design bases and in a manner to protect the public health and safety. To this end Duke has established and implemented a quality assurance program which conforms to the criteria established in Appendix B to 10CFR, Part 50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants" published June 27, 1970 (35 F. R. 10499) and amended September 17, 1971 (36 F. R. 18301) and amended January 20, 1975 (40 F. R. 3210D).

This topical report is written in the format of a Safety Analysis Report (SAR) Chapter 17, "Quality Assurance", in accordance with Revision 2 of the NRC's Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants - LWR Edition" and subsequent NRC guidelines. The quality assurance program described herein is applicable to all Duke nuclear power plants as referenced by Chapter 17 of the plants' SAR's.

This Topical Report describes the Quality Assurance Program for those systems, components, items, and services which have been determined to be nuclear safety related (QA Condition 1). In addition, Duke's Quality Assurance Program provides a method of applying a graded Quality Assurance Program to certain non-safety related systems, components, items, and services. These are classified as QA Conditions 2, 3, or 4. This method involves defining a Quality Assurance "Condition" for each level of quality assurance required. These will be designated as "QA Condition ____". The quality of systems, components, items, and services within the scope of QA Conditions 1, 2, 3, and 4 is assured commensurate with the system's, component's, item's, or service's importance to safety. The following conditions have been defined.

QA Condition 1 covers those systems and their attendant components, items, and services which have been determined to be nuclear safety related. These systems are detailed in the Safety Analysis Report applicable to each nuclear station. The Topical Report applies in its entirety to systems, components, items, and services identified as QA Condition 1.

QA Condition 2 covers those systems and their attendant components, items, and structures important to the management and containment of liquid, gaseous, and solid radioactive waste.

QA Condition 3 covers those systems, components, items, and services which are important to fire protection as defined in the Hazards Analysis for each station. The Hazards Analysis is in response to Appendix A of NRC Branch Technical Position APCSB 9.5-1.

QA Condition 4 covers those seismically designed/restrained systems, components, and structures whose continued functions are not required during and after the seismic event. The general scope of these systems, components, and structures, identified as Seismic Category II (SCII) are defined in Regulatory Guide 1.29, Seismic Design Classification.

ATTACHMENT 5

6.1.3.4 Audits

Audits of Site activities shall be performed under the cognizance of the NSRB. These audits shall encompass:

a. The conformance of station operation to provisions contained within the Technical Specifications and applicable facility operating license conditions.

b. The performance, training and qualifications of the station staff.

c. The results of actions taken to correct deficiencies occurring in equipment, structures, systems or methods of operation that affect nuclear safety.

d. The performance of activities required by the quality assurance program to meet the criteria of Appendix B to 10 CFR 50.

e. The station emergency plan and implementing procedures.

f. The station security plan and implementing procedures.

g. Any other area of station operation considered appropriate by the NSRB or the Senior Vice President, Nuclear Generation.

h. The station fire protection program and implementing procedures.

i. The Offsite Dose Calculation Manual and implementing procedures.

j. The Radiological Environmental Monitoring Program and the results thereof.

k. The Process Control Program and implementing procedures for solidification of radioactive wastes.

l. The performance of activities required by the Quality Assurance Program to meet the criteria of Regulatory Guide 1.21 Revision 1, June 1974 and Regulatory Guide 4.1 Revision 1, April 1975.

ATTACHMENT 4

REVIEW

6.5.2.8 The NSRB shall review:

a. The safety evaluations for: (1) changes to procedures, equipment, or systems, and (2) tests or experiments completed under the provision of Section 50.59, 10 CFR to verify that such actions did not constitute an unreviewed safety question;

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR;

c. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR;

d. Violations of Codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance;

e. Significant operating abnormalities or deviations from normal and expected performance of unit equipment that affect nuclear safety;

f. All REPORTABLE EVENTS;

g. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems or components that could affect nuclear safety;

h. Quality Assurance Program audits relating to station operations and actions taken in response to these audits; and

i. Reports of activities performed under the provisions of Specifications 6.5.1.1 through 6.5.1.10.

AUDITS

6.5.2.9 Audits of site activities shall be performed under the cognizance of the NSRB. These audits shall encompass:

a. The conformance of unit operation to provisions contained within the Technical Specifications and applicable license conditions;

b. The performance, training, and qualifications of the entire station staff;

ATTACHMENT 3

ADMINISTRATIVE CONTROLS

AUDITS (Continued)

c. The results of actions taken to correct deficiencies occurring in unit equipment, structures, systems, or method of operation that affect nuclear safety;

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix B, 10 CFR Part 50;

e. The Emergency Plan and implementing procedures;

f. The Security Plan and implementing procedures;

g. The Facility Fire Protection programmatic controls including the implementing procedures;

h. The fire protection equipment and program implementation utilizing either a qualified offsite licensee fire protection engineer or an outside independent fire protection consultant. An outside independent fire protection consultant shall be used at least every third year;

i. The Radiological Environmental Monitoring Program and the results thereof;

j. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures;

k. The PROCESS CONTROL PROGRAM and implementing procedures for SOLIDIFICATION of radioactive wastes;

l. The performance of activities required by the Quality Assurance Program for effluent and environmental monitoring and;

m. Any other area of site operation considered appropriate by the NSRB or the Senior Vice President, Nuclear Generation.

AUTHORITY

6.5.2.10 The NSRB shall report to and advise the Senior Vice President, Nuclear Generation on those areas of responsibility specified in Specifications 6.5.2.8 and 6.5.2.9.

AUDITS (Continued)

c. The results of actions taken to correct deficiencies occurring in unit equipment, structures, systems, or method of operation that affect nuclear safety;

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix B, 10 CFR Part 50;

e. The Emergency Plan and implementing procedures;

f. The Security Plan and implementing procedures;

g. The Facility Fire Protection programmatic controls including the implementing procedures;

h. The fire protection equipment and program implementation utilizing either a qualified offsite licensee fire protection engineer or an outside independent fire protection consultant. An outside independent fire protection consultant shall be used at least every third year;

i. The Radiological Environmental Monitoring Program and the results thereof;

j. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures;

k. The PROCESS CONTROL PROGRAM and implementing procedures for processing and packaging of radioactive wastes;

l. The performance of activities required by the Quality Assurance Program for effluent and environmental monitoring; and

m. Any other area of site operation considered appropriate by the NSRB or the Executive Vice President Power Generation.

RECORDS

6.5.2.10 Records of NSRB activities shall be prepared, approved, and distributed as indicated below:

a. Minutes of each NSRB meeting shall be prepared, approved, and forwarded to the Senior Vice President, Nuclear Generation and to the Executive Vice President, Power Generation within 14 days following each meeting;

REVIEW

6.5.2.8 The NSRB shall be responsible for the review of:

a. The safety evaluation for: (1) changes to procedures, equipment, or systems, and (2) tests or experiments completed under the provision of Section 50.59, 10 CFR to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment, or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR;

c. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR;

d. Proposed changes in Technical Specifications or this Operating License;

e. Violations of Codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance;

f. Significant operating abnormalities or deviations from normal and expected performance of unit equipment that affect nuclear safety;

g. All REPORTABLE EVENTS;

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety;

i. Quality Verification Department audits relating to station operations and actions taken in response to these audits; and

j. Reports of activities performed under the provisions of Specifications 6.5.1.1 through 6.5.1.12.

AUDITS

6.5.2.9 Audits of site activities shall be performed under the cognizance of the NSRB. These audits shall encompass:

a. The conformance of unit operation to provisions contained within the Technical Specifications and applicable license conditions;

b. The performance, training, and qualifications of the entire station staff;

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

May 7, 1992

FOR INFORMATION ONLY

Docket Nos. 50-413
and 50-414

Mr. M. S. Tuckman
Vice President, Catawba Site
Duke Power Company
4800 Concord Road
York, South Carolina 29745

Dear Mr. Tuckman:

SUBJECT: ISSUANCE OF AMENDMENTS - CATAWBA NUCLEAR STATION, UNITS 1 AND 2 (TACS M82006/M82007)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 96 to Facility Operating License NPF-35 and Amendment No. 90 to Facility Operating License NPF-52 for the Catawba Nuclear Station, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated December 18, 1991, as supplemented on February 17, 1992.

The amendments revise the TS to reflect a reorganization of the Duke Power Company (DPC). The reorganization essentially decentralizes the corporate management of nuclear activities to each of DPC's three nuclear site facilities, including the Catawba Site.

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Robert E. Martin

Robert E. Martin, Senior Project Manager
Project Directorate II-3
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 96 to NPF-35
2. Amendment No. 90 to NPF-52
3. Safety Evaluation

cc w/enclosures:
See next page

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ATTACHMENT 2

REVIEW

6.5.2.8 The NSRB shall be responsible for the review of:

a. The safety evaluation for: (1) changes to procedures, equipment, or systems, and (2) tests or experiments completed under the provision of Section 50.59, 10 CFR to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment, or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR,

c. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR;

d. Proposed changes in Technical Specifications or this Operating License;

e. Violations of Codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance;

f. Significant operating abnormalities or deviations from normal and expected performance of unit equipment that affect nuclear safety,

g. All REPORTABLE EVENTS;

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety;

Verification

i. Quality ~~Assurance~~ Department audits relating to station operations and actions taken in response to these audits; and

j. Reports of activities performed under the provisions of Specifications 6.5.1.1 through 6.5.1.~~11~~ 12.

AUDITS

SCC

6.5.2.9 Audits of ~~unit~~ activities shall be performed under the cognizance of the NSRB. These audits shall encompass:

a. The conformance of unit operation to provisions contained within the Technical Specifications and applicable license conditions ~~at least once per 12 months~~,

Station

b. The performance, training, and qualifications of the entire ~~unit~~ staff ~~at least once per 12 months~~;

NSRB organization and therefore should be considered acceptable.

Audit frequencies are being deleted here but in the revised QA Topical we are preparing the following statement, using SRP 17.3 guidance on planned and periodic assessments scheduling and resource allocation:

"Audits of selected aspects of operational phase activities are performed with a frequency commensurate with safety significance and in such a manner as to assure that an aud t of all safety related functions is completed within a period of two (2) years., The audit system is reviewed periodically and revised as necessary to assure coverage commensurate with current and planned activities."

The naming of Executive Vice President, Power Generation reflects the realignment of reporting authority fc. the NSRB as discussed in draft revised CNS FSAR Chapter 13 and in the QA Topical Section 17.3. This change is administrative, since the functions of the NSRB are unaffected but we also believe this change is philosophically correct and acceptable.

These changes do not alter the function nor diminish the quality of the Audit Program. Therefore, they should be considered acceptable.

6.5.2.10 These changes reflect realignment of authority or responsibility as discussed above and described in proposed TS 6.2.1. These reporting requirements to Senior Management are also discussed in the revised QA Topical Report in Section 17.3.3 "Self Assessment". These changes are administrative only and therefore are acceptable.

6.6.1 These changes reflect realignment of authority or responsibility as discussed above and described in proposed TS 6.2.1, and draft revised CNS FSAR, Chapter 13.

The key supervisory titles have been revised to reflect the reorganization and their re-naming. The changes are purely administrative and should be acceptable.

6.7.1 The changes in (a) and (c) reflect realignment of authority or responsibility to the site as discussed above and described in TS 6.2.1, and draft revised CNS the site. The number 12 is a typographical correction. These changes are clearly editorial or administrative and therefore should be considered acceptable.

6.5.2.2 The naming of Executive Vice President, Power Generation reflects the realignment of reporting authority for the NSRB as discussed in draft revised CNS FSAR Chapter 13 and in the QA Topical Section 17.3. This change is administrative, since the functions of the NSRB are unaffected but we also believe this change is philosophically correct and acceptable.

The other proposed chance in this section would increase the pool of qualified individuals from which candidates could be appointed to independently review operational phase activities. The requirements for those who operate the plant are at least this flexible. The appointment would be subject to the approval of the Executive Vice President, Power Generation.

This proposed change is not directly related to the Duke reorganization or the requisite revisions to our licen'ing documents. If its consideration might slow down the review of all o'her changes in this chapter, we would prefer to consider it separately in another submittal.

6.5.2.7 These changes reflect realignment of authority or responsibility as discussed above and described in proposed TS 6.2.1, draft revised CNS FSAR, Chapter 13, and the revised QA Topical Report.

The use of the term "site" assures the continued independent nature of the NSRB.

6.5.2.8 This change reflects the renaming of the Department, the function of which is unchanged. This administrative change should be considered acceptable.

The "12" corrects a typographical error and is purely editorial.

6.5.2.9 We are applying here the broader term "site" to reflect all those activities associated with the station, not those that are only specific to the operation of the unit.

The term "station" in (b) is consistent with its use in Specification 6.2.2(f) and implies those people reporting to the Station Manager, responsible for operation and maintenance of the unit. These changes are administrative and do not alter the function of the

AUDITS (Continued)

c. The results of actions taken to correct deficiencies occurring in unit equipment, structures, systems, or method of operation that affect nuclear safety ~~at least once per 6 months~~;

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix B, 10 CFR Part 50, ~~at least once per 24 months~~;

e. The Emergency Plan and implementing procedures ~~at least once per 12 months~~.

f. The Security Plan and implementing procedures ~~at least once per 12 months~~.

g. The Facility Fire Protection programmatic controls including the implementing procedures ~~at least once per 24 months by qualified licensee QA personnel~~;

h. The fire protection equipment and program implementation ~~at least once per 12 months~~ utilizing either a qualified offsite licensee fire protection engineer or an outside independent fire protection consultant. An outside independent fire protection consultant shall be used at least every third year.

i. The Radiological Environmental Monitoring Program and the results thereof ~~at least once per 12 months~~.

j. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures ~~at least once per 24 months~~;

k. The PROCESS CONTROL PROGRAM and implementing procedures for processing and packaging of radioactive wastes ~~at least once per 24 months~~.

l. The performance of activities required by the Quality Assurance Program for effluent and environmental monitoring ~~at least once per 12 months~~; and

SITE

m. Any other area of ~~unit~~ operation considered appropriate by the NSRB or the ~~Vice President, Nuclear Production~~.

Executive Vice President, Power Generation

RECORDS

6.5.2.10 Records of NSRB activities shall be prepared, approved, and distributed as indicated below.

Senior Vice President, Nuclear Generation

a. Minutes of each NSRB meeting shall be prepared, approved, and forwarded to the ~~Vice President, Nuclear Production~~ and to the Executive Vice President, ~~Power Group~~ within 14 days following each meeting.

Power Generation

Of

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

March 31, 1999

MEMORANDUM TO: Loren R. Plisco, Director
Division of Reactor Projects
Region II

FROM: Cecil O. Thomas, Director
Project Directorate II-3
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

SUBJECT: RESPONSE TO TECHNICAL ASSISTANCE (TIA 97-015) REGARDING THE IMPLEMENTATION OF 10 CFR 50.65 – BROWNS FERRY NUCLEAR PLANT UNIT 1 (TAC NO. M98931)

By memorandum dated June 4, 1997, the Division of Reactor Projects, Region II requested the assistance of the Office of Nuclear Reactor Regulation (NRR) in determining the acceptability of the Tennessee Valley Authority's (TVA) actions with respect to the implementation of 10 CFR 50.65 at Browns Ferry Unit 1. The issue arose as a result of an inspection conducted at the Browns Ferry Nuclear Plant on April 4-18, 1997. The results of that inspection are documented in NRC combined Inspection Reports 50-259/97-04, 50-260/97-04, and 50-297/97-04 (IR 97-04) issued on May 21, 1997.

IR 97-04 documents that the NRC inspectors determined that TVA's actions to implement the rule at Unit 1 are technically adequate, however the report raised the question as to whether or not the approach taken is, in fact, legal under the maintenance rule.

The inspection team determined that TVA's implementation of 10 CFR 50.65 for Unit 1 is for a facility that is shutdown and defueled. As such, the Unit 1 program does not encompass all systems and components that would be covered for an operating unit. The team did not have any specific safety concerns with the program, and concluded that TVA's actions to implement the regulation were adequate technically. However, a question arose, as to whether or not the program meets the requirements of the regulation. At issue is whether TVA's approach to scoping Unit 1 structures, systems, and components by considering the defueled and indefinite shutdown condition of Unit 1 satisfies the requirements of the rule.

Region II requested NRR's assistance (with OGC participation) to develop a list of actions necessary for TVA to comply legally with §50.65 for Unit 1. The Region requested that the list of actions should be provided directly to TVA.

By letter dated July 30, 1997, NRC informed TVA that the limitation of applicability stated in §50.65(a)(1) does not apply to Browns Ferry Unit 1 in the absence of certification per §50.82(a)(1). The letter identified three alternatives available to TVA to resolve the issue. The three options identified were:



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R/1

1. Revise the scope of the monitoring program for Unit 1 to include structures, systems, and components as specified in §50.65(b), or

2. Submit written certification per §50.82(a)(1) that operations have ceased permanently, or

3. Request an exemption from the requirements of §50.65 that are not now being met.

TVA responded by letter on September 29, 1997, asserting that it believes that the Browns Ferry Unit 1 program is in compliance with §50.65 and presenting its rationale for this assertion. The NRC did not respond to TVA's September 29th letter and, instead met with TVA representatives on January 26, 1998, to discuss the issue further. The meeting was documented in a summary issued February 6, 1998.

TVA proposed an alternative program to resolve the issue by letter dated April 3, 1998. Subsequent to this submittal, the staff had several discussions by telephone with TVA representatives to obtain clarification and to explore possible program modifications. The staff did not issue an evaluation of TVA's proposed alternate, but instead referred back to the staff's position identified in the July 30, 1997, letter.

On February 4, 1999, TVA submitted a request for a temporary partial exemption from the requirements of §50.65. The staff review of the proposed exemption will be tracked under TAC No. MA5033.

Action on TIA 97-005 is complete.

Docket No. 50-296

cc: A. R. Blough, RI
G. E. Grant, RIII
K. E. Brockman, RIV

CONTACT: A. De Agazio, NRR
(301) 415-1443