NSRS REPORT NO. R-85-07-NPS

Item No.	Item Title/Status	Date Closed
01	Manager of Power and Engineering Appointment of a Records Manager	

NSRS REPORT NO. ~~R-85-08-OE/NUC PR~~

Item No.	Item Title/Status	Date Closed
01 (OE)	Inadequate OE Environmental Qualification Procedure for Equipment Qualification by Similarity	
02	Inadequate OE and NUC PR Procedures for Initiating and Processing NCR - FE/ERs	
03 (OE)	Lack of Priority on SQN NCR Initiation	
04 (OE)	Lack of OE Engineer Awareness of the NCR - FE/ER Process	
05 (OE)	Lack of Timeliness in Initiating and Processing NCR - FE/ERs	
06	Failure of Management to Correct Problems with Timeliness and Responsiveness Involving the NCR - FE/ER Process	

NSRS REPORT NO.

Item No.	Item Title/Status	Date Closed
01	Lack of Procedures for Handling Employee Concerns/Allegations	

ENCLOSURE 4

NSRS REPORTS

79-10-01

TVA 64 (OS-9-65)

UNITED STATES GOVERNMENT

Memorandum

TENNESSEE VALLEY AUTHORITY

TO : G. H. Kimmons, Manager of Engineering Design and Construction, W12A9 C-K
H. G. Parris, Manager of Power, 500C CST2-C

FROM : Charles Bonine, Jr., Manager of Management Services, W12A1 C-K

DATE :

SUBJECT: BROWNS FERRY NUCLEAR PLANT - CONFIDENTIAL EMPLOYEE CONCERN OVER OPERATING PRACTICES WHERE PROTECTIVE SYSTEM SIGNALS ARE BYPASSED

Attached is the Nuclear Safety Review Staff's report on confidential employee concern 79-10-01. The report indicates that the plant instruction, Standard Practice BF 8.2, for control of temporary alterations fails to properly implement the requirements of the Division of Nuclear Power's Division Procedures Manual for control of temporary alterations. It further indicates that the instructions contained in Standard Practice BF 8.2 were not followed during the installation of an electrical jumper on September 26, 1979. The unauthorized and uncontrolled installation of the jumper caused both units' recirculation pumps to trip. Based on these findings, it is recommended that the Division of Nuclear Power perform an evaluation to determine why appropriate procedures were not available at Browns Ferry Nuclear Plant for control of temporary alterations and why the available procedure was not followed. As part of the evaluation, it should also be determined if disciplinary action is appropriate to assure the prevention of recurrence of this or similar occurrences.

The remainder of the recommendations which will close the employee concern when implemented are contained in the report. If you do not concur with the recommendations in the report and in this memorandum, please let me know by November 19, 1979. Otherwise, we will consider the employee concern closed. The Nuclear Safety Review Staff will monitor the implementation of the recommendations.

Charles Bonine, Jr.

KWW:KRW
Attachment
cc: E. A. Belvin, ROB-M (Attachment)

bc: E. Gray Beasley, 309 GB-K (Attachment)

TENNESSEE VALLEY AUTHORITY

NUCLEAR SAFETY REVIEW STAFF

CONFIDENTIAL EMPLOYEE CONCERN - CASE NO. 79-10-01

Subject: TVA
Browns Ferry Nuclear Plant

Employee Concern Over Operating Practices Where Protective System Signals are Bypassed

Period of Investigation: October 1-16, 1979

Investigator: H. L. Jones
H. L. Jones — 10/17/79 Date

Reviewed By: K. W. Whitt
K. W. Whitt — 10/17/79 Date

H. E. McConnell
H. E. McConnell — 10/18/79 Date

Approved By: E. Gray Beasley
E. Gray Beasley, Acting Chief, Nuclear Safety Review Staff — 10/18/79 Date

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I. Case No. 79-10-01 Confidential Concern

On Monday, October 1, 1979, the Nuclear Safety Review Staff (NSRS) received the following confidential employee concern from Individual A.

> At Browns Ferry Nuclear Plant operating practices are such that protective system isolation functions are frequently bypassed.
>
> A specific example offered was an event that occured on September 26, 1979, which involved the negating of valid automatic steam line isolation signals during a troubleshooting and problem evaluation period just prior to unit 1 manual shutdown due to high steam vault temperatures. NSRS was requested to investigate the matter on a confidential basis.

II. Conclusions

A. The concern expressed by Individual A is valid, evidenced by the fact that temporary alteration procedures are not being adhered to and are inadequate.

B. The Browns Ferry Nuclear Plant Standard Practice Procedure entitled "Temporary Alterations" (BF 8.2) does not provide for adequate control of temporary conditions established when installing temporary alterations. Specifically, Standard Practice Procedure BF 8.2 does not adequately implement the requirements of Divisions Procedure Manual (DPM) No. N73011 (entitled "Control of Temporary Alterations and Use of the Temporary Alteration Order").

C. Though apparently acting within the tacit guidelines previously established by plant management, Plant Operations personnel installed an unauthorized and uncontrolled electrical jumper in violation of approved procedures during the course of events on September 26.

D. Throughout the course of events on September 26, the bypassing of protective signals did not negate the ability to automatically close the main steam isolation valves upon receipt of steam tunnel high temperature. One train was available and would have tripped.

III. Recommendations

A. DPM No. N73011 should be revised by the Division of Nuclear Power to clearly affirm that temporary alterations (specifically electrical alterations) not covered by written procedures constitute modifications and must be reviewed to determine whether an unreviewed safety question is involved as specified by 10CFR50.59 prior to implementation.

B. Standard Practice BF 8.2 should be revised by the Division of Nuclear Power to implement the requirements of DPM No. N73011.

C. DPM No. 73011 defines the use of the temporary alteration control form (TVA 6266). This form should be revised by the Division of Nuclear Power to provide for Plant Operations Review Committee (PORC) approval of all temporary alterations prior to implementation.

D. Modifications should be made to the ventilation system such that a reasonable margin between normal steam tunnel temperature and isolation setpoint is maintained. The conditions that make it necessary to leave the door between the steam tunnel and reactor building open and to utilize a portable fan for ventilation should be eliminated. The Division of Engineering Design should evaluate and recommend alteratives to resolve this situation. Consideration should also be given to 1) installing more precision temperature sensors and 2) types of insulating material utilized.

This item should be given a very high priority and temporary instructions or measures should be given by EN DES to provide interim relief.

E. Plant personnel should review the identification of panels, boards, terminal strips, etc., to ensure that equipment is correctly marked in order to eliminate errors when making temporary alterations. See item V. for details.

F. From a generic standpoint, the Division of Nuclear Power should review applicable plant instructions at all nuclear plants to ensure that the requirements of DPM No. N73011 are being implemented.

G. The Division of Nuclear Power should submit a revision to the BFN technical specifications to clarify the interpretation and intent of inoperable temperature switches in technical specification table 3.2.A.

H. The Division of Nuclear Power should assure that the policy of following procedure is carried out at all nuclear plants as required by 10CFR50, Appendix B, Criterion V.

I. The Division of Nuclear Power should define an "emergency condition" as used in DPM No. N73011 and standard practice BF 8.2 and provide a set of criteria for determining such a condition exists.

J. The Division of Nuclear Power should establish a requirement that each declared emergency condition be recorded in the shift engineer's log and submitted to the Plant Operations Review Committee for review during the committee's next scheduled meeting.

K. EN DES should resolve why both recirculation pumps tripped when a jumper was placed in panel 9-17 terminal strip CC between terminals Nos. 3 and 4. See item V. for details.

IV. Scope of Investigation

The scope of the investigation included the following:

A. Interviewed seven Division of Nuclear Power employees.

B. Reviewed the procedures prepared by the Division of Nuclear Power for the control of temporary alterations.

C. Reviewed drawings that are exemplary of the circuits involved during the investigated events on September 26.

D. Reviewed Licensee Reportable Event Determination and Daily Journal Logs relating to the events occurring on September 26.

E. Observed the as-installed hardware on panel 9-17.

V. Details of Investigation

A. Meeting with Division of Nuclear Power Employees

The investigation into Case No. 79-10-01, employee concern over operating practices where protective system signals are bypassed, began with individual interviews on October 4-5, 1979, at Browns Ferry Nuclear Plant (BFN) with the following individuals:

H. L. Abercrombie, Plant Superintendent
J. L. Harness, Assistant Plant Superintendent
J. Pittman, Instrumentation Supervisor
J. D. Thomason, Instrumentation Specialist
R. T. Smith, QA Supervisor
J. Marbutt, Assistant Shift Engineer
[redacted]

1. The sequence of events on September 26 relating to the employee concern that operating practices resulted in negating valid automatic isolation signals prior to unit 1 manual shutdown follows:

a. Prior to 12 noon, unit 1 operating at approximately 1089 MWe.

b. 12 noon, received main steam half isolation signal due to steam line high temperature. Control room instrumentation and local inspection indicated that the main steam tunnel temperature was not high. Later investigation revealed that an open cable caused the half isolation signal.

Note: The half isolation signal was from trip logic B_2. At this time, a half isolation signal form trip logic A_1 or A_2 would have automatically closed the main steam isolation valves (see figures 1 and 2).

c. 1222 hours, while troubleshooting for the cause of the half isolation, the reactor recirculation pumps were tripped.
Note: The troubleshooting process involved the placement of a jumper on terminal strip BB, between terminals Nos. 3 and 4 in panel 9-17. Terminal strip CC was incorrectly labeled BB. The placement of the jumper on terminal strip CC between terminals Nos. 3 and 4 resulted in the recirculation pump trip by actuation of the logic for the recirculation pump trip breakers.
d. 1250 hours, restarted 1A recirculation pump.
e. 1305 hours, restarted 1B recirculation pump.
f. 1430 hours, PORC reviewed and recommended approval of a jumper on panel 9-17 terminal strip BB, between terminals Nos. 3 and 4.
g. 1525 hours, jumper approved by PORC was installed.
h. Personnel entered the steam tunnel to inspect the condition of the temperature switches. During the inspection, one of the individuals bumped a temperature switch causing a half isolation signal on trip logic A_1 and the associated relay was subsequently blocked such that an isolation signal through trip logic A_1 could not occur.
Note: In this condition, main steam line isolation would only be initiated by coincident actuation of the unmodified trip logic A_2 and B_1.
i. 1859 hours, unit 1 manually scramed and depressurized to close main steam isolation valves.
j. 0114 hours, on September 27, main steam isolation valves closed.
k. The fuse was replaced on the trip logic A_1 circuit; an investigation of the trip logic B_2 circuit revealed that a cable was open, and the open cable was replaced. During the investigation another cable was damaged and was repaired.
l. Unit 1 was restarted on September 27.

2. Inadequate control of Temporary Alternations

The jumper that was placed at terminal strip CC, between terminals Nos. 3 and 4 which tripped the recirculation pumps was placed in violation of Standard Practice BF 8.2. This standard practice requires that the installation of temporary alternation be covered by instructions or TVA Form 6266. No instructions or TVA Form 6266 were used. The standard practice also provides for modification of these normal requirements under emergency conditions. The employee that installed the jumper apparently did so under the emergency conditions provision. However, the standard practice provides for the shift engineer only to modify the instructions under emergency conditions. The employee that installed the jumper was not the shift engineer.

Browns Ferry plant personnel have implemented a jumper control system which requires that jumpers be obtained from the shift engineer. However, the installed jumper was not obtained from the shift engineer and its installation was not approved by the shift engineer. Further the unit operator was not informed that the jumper was being installed.

It is not clear who determines when an emergency exists and when it is over. Since the shift engineer is responsible for taking the action under emergency conditions, it is assumed that he also has responsibility for making such determinations. No criteria have been developed for determining emergency conditions. It appears unfair to the shift engineer and not in the best interest of TVA to place the entire burden of determining if an emergency condition exists on the shift engineer without previously approved guidelines. The fact that the shift engineer is probably in the best position and possesses the most complete set of data for making such a determination is fully appreciated and is the proper level for that determination. It also seems reasonable to assume that this task could be safer and assure more consistent results if a good definition of an "emergency condition" and a set of criteria for determination were made available to him.

DPM N73011 also provides for the installation of temporary alterations associated with troubleshooting without instructions or TVA Form 6266 if the shift engineer concurs. Under this provision, the alteration must be accomplished with jumpers or test probes which do not leave the craftsman's hand; the trouble shooting may involve only the normal use of components whose design function is to modify circuits such as PK blocks, cutout switches, test switches, etc.; when these components are used, a second party must be present to witness the temporary alteration and to verify the return to normal. At least one individual interviewed indicated that the jumper in question was installed while troubleshooting the steam tunnel temperature detector problem. In this event, DPM N73011 was not implemented as follows: the shift engineer did not concur, the jumper remained installed without the aid of a craftsman, the bypassed circuit was not an example of those allowed by DPM N73011, a second party was not present to witness the temporary alteration and to verify the return to normal.

Standard Practice BF 8.2, which contains the instructions to be used by plant personnel states that hand held devices that cannot be attached such as holding a relay in is not considered a temporary alteration. This is in conflict with the requirements of DPM N73011 and deviates in the nonconsertative direction. It represents an example of failure by the plant to implement requirements established by Division management. It is also an example of failure by Division management to assure proper review of plant instructions and could represent a weakness in the quality assurance program. From discussions with plant management it appeared that installation of electrical jumpers by personnel other than the shift supervisor under emergency and troubleshooting conditions was within the practice that had tacit acceptance of Browns Ferry plant management. It seemed apparent that the situation would have been acceptable to plant management if the jumper had been properly installed.

3. Additional pertinent items were discussed as follows:

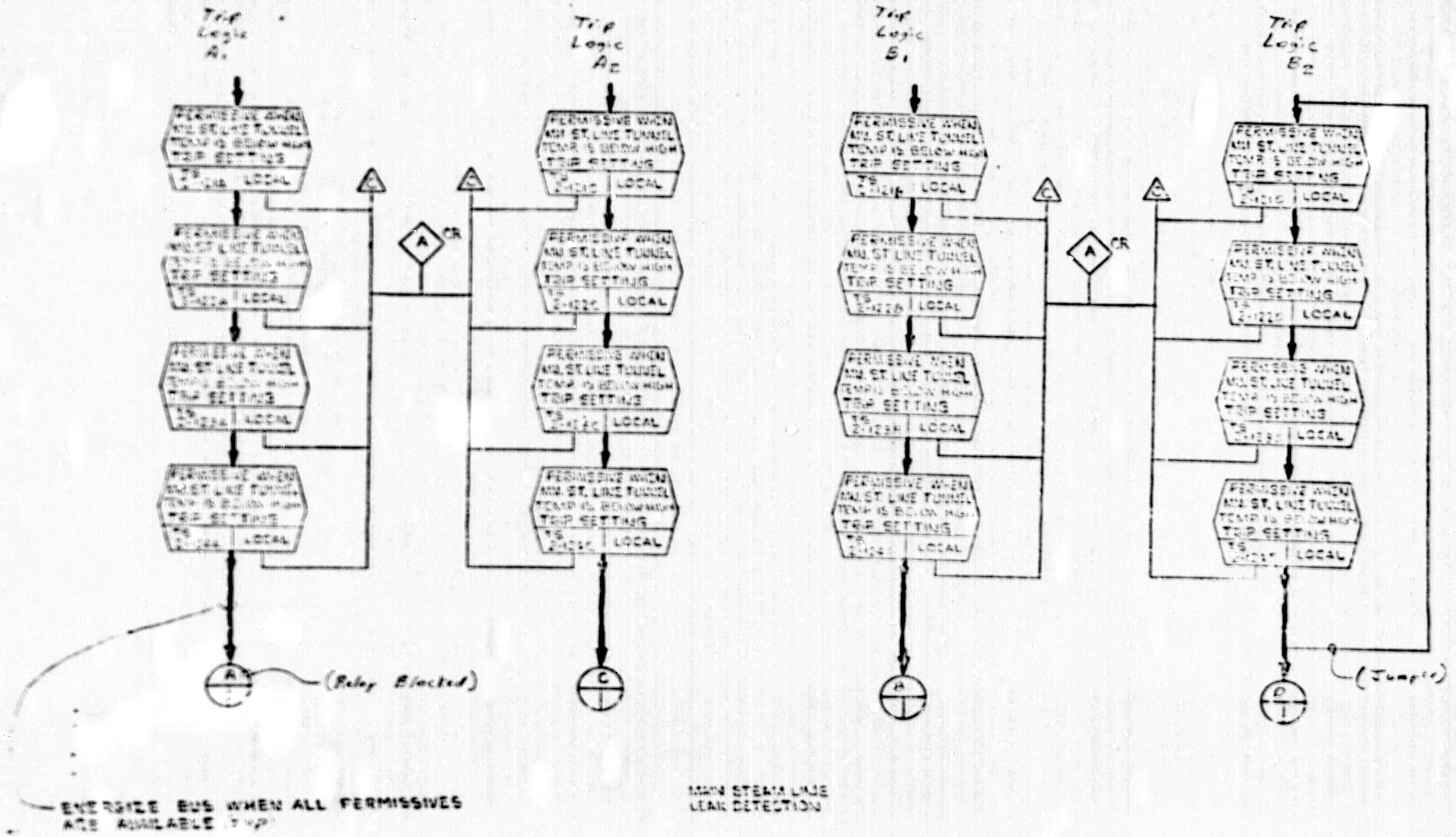
   A. Due to variation in interpretation of technical specifications, the main steam lines were not isolated within eight hours as required by technical specification table 3.2.A. Following consultation with offsite personnel, the new interpretation is that upon loss of operability of any two temperature switches, the main steam isolation valves must be closed within eight hours if the condition is not corrected. With the new interpreation it was determined that the main steam lines should have been isolated by 2325 hours on September 26. Reportable Occurrence Report BFRO-50-259/7925 was written to notify the NRC.

   B. EN DES issued an Engineering Change Notice (ECN L1991) to provide for improved flow distribution and to balance the ventilation system in the steam tunnel. The ECN has been completed in unit 2 and according to site personnel has degraded the ability to maintain lower temperatures in the steam tunnel. This necessitated changing the setpoint of the temperature switches from 185° to 195° F (units 1 and 3 setpoint remain at 185° F).

4. Followup Actions
   The Nuclear Safety Review Staff has reviewed the recirculation pump trip electrical circuits and has found that a jumper placed in panel 9-17 on terminal strip CC, between terminals Nos. 3 and 4 should result in a Division II A loop recirculation pump trip and should not result in the tripping of both recirculation pumps as experienced on September 26.

VI. References

1. Minutes of Plant Operations Review Committee meeting No. 3446.
2. Reportable Occurrence Report BFRO-50-259/7925.
3. Standard Practice BF 8.2, "Temporary Alterations."
4. DPM No. N73011, "Control or Temporary Alterations and Use of the Temporary Alteration Order."
5. Daily Journal Logs for September 26, 1979.



MAIN STEAM LINE LEAK DETECTION

Figure 1

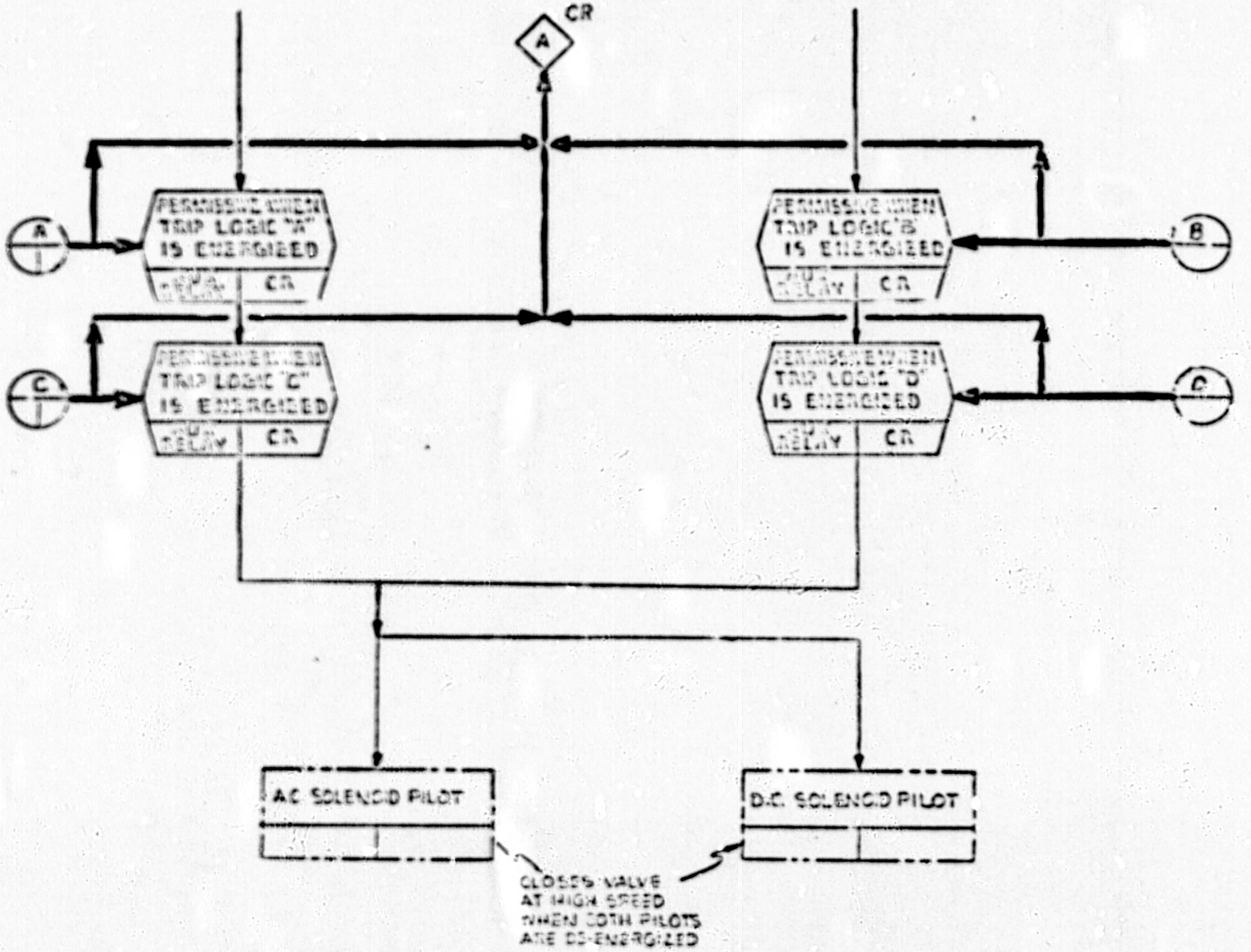

CR
A
PERMISSIVE WHEN TRIP LOGIC "A" IS ENERGIZED
AUX RELAY
CR
PERMISSIVE WHEN TRIP LOGIC "B" IS ENERGIZED
AUX RELAY
CR
PERMISSIVE WHEN TRIP LOGIC "C" IS ENERGIZED
AUX RELAY
CR
PERMISSIVE WHEN TRIP LOGIC "D" IS ENERGIZED
AUX RELAY
CR
A.C. SOLENOID PILOT
D.C. SOLENOID PILOT
CLOSES VALVE AT HIGH SPEED WHEN BOTH PILOTS ARE DE-ENERGIZED

Figure 2

79-10-02

TVA 64 (OS-9-69)

UNITED STATES GOVERNMENT

Memorandum

TENNESSEE VALLEY AUTHORITY

TO : G. H. Kimmons, Manager of Engineering Design and Construction, W12A9 C-K

FROM : E. A. Belvin, Director, Office of Health and Safety, ROB-M

DATE : Sent to EAB 10-26-79

SUBJECT: SEQUOYAH NUCLEAR PLANT - EMPLOYEE CONCERN OVER NONREPORTABLE DETERMINATION ON NONCONFORMANCE REPORT 1866

Attached is the Nuclear Safety Review Staff's report on employee concern 79-10-02. The report concludes that no valid significant nuclear safety concern was shown to exist during the course of the investigation and that TVA's present mechanism for determination of reportability satisfies all of the requirements of 10CFR21 and 10CFR50.55(e). These findings do not substantiate the claims made in the employee concern, and NSRS considers this employee concern closed.

E. A. Belvin

TGT:WC

Attachment

cc (Attachment):

W. F. Willis, E12A1 C-K

Charles Bonine, Jr., W12A1 C-K

RF

TENNESSEE VALLEY AUTHORITY

NUCLEAR SAFETY REVIEW STAFF

EMPLOYEE CONCERN - CASE NO. 79-10-02

Subject: Tennessee Valley Authority
Sequoyah Nuclear Plant

Employee Concern Over Nonreportable Determination on Sequoyah Nuclear Plant Nonconformance Report (NCR) 1866

Period of Investigation: October 10-17, 1979

Investigator: T. G. Tyler
T. G. Tyler — Date: 10/23/79

Reviewed by: H. E. McConnell
H. E. McConnell — Date: 10/24/79

K. W. Whitt
K. W. Whitt — Date: 10/23/79

Approved by: E. Gray Beasley
E. Gray Beasley, Acting Chief
Nuclear Safety Review Staff — Date: 10/24/79

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I. Case 79-10-02 Concern

On October 10, 1979, the Nuclear Safety Review Staff (NSRS) received the following employee concern from W. D. DeFord of the Quality Engineering Branch (QEB) in the Division of Engineering Design (EN DES).

> The determination of "Non-Reportable" by the Nuclear Licensing Section (NLS) on Sequoyah Nuclear Plant Nonconformance Report (NCR) 1866 seems in error for the following reasons:
>
> 1. The investigation notes indicate that the craft individual did not know how to properly terminate the cable. Conversations with NLS personnel indicate he has been terminating for at least five years. This makes all his terminations of that type suspect. No check of this individual's past terminations has been made. He is simply going to be retrained. It is a little late in the game for retraining.
>
> 2. Sections B and E of the 10CFR50.55(e) reportability worksheets used by NLS are in error. Sections B(1) and B(3) should be checked "affirmative" which would make the item reportable.
>
> 3. The investigation notes indicate a similar occurrence at Sequoyah Nuclear Plant (NCR 1194 dated September 21, 1978) involving a different cable vendor.
>
> Note: It is just possible that poor installation practice as well as marginal conductor material could be the problem. In any event, if the cable breaks that easily while being stripped or crimped, there is a good possibility of inservice failures occurring which was mentioned in section A of the 10CFR50.55(e) worksheet.
>
> Note: Since Watts Bar Nuclear Plant (WBN) personnel, using WBN procedures, also broke a wire on the first try at Sequoyah and then changed to 1/16-inch stripping, there is a strong possibility of similar problems at WBN which were not reported.

II. Conclusions

A. No valid significant nuclear safety concern was shown to exist during the course of this investigation.

B. The responsibility for reporting significant defects and/or failures to comply was assigned to NLS of the Nuclear Engineering Branch (NEB) on January 1, 1978. (See reference 9.) On the other hand, specification and/or approval of corrective action for NCR's has been and continues to be the responsibility of the applicable thermal power engineering branch or the design project. TVA's NCR reporting mechanism has been the subject of this investigation. Based on the findings of the investigation, we have concluded that TVA's present mechanism for determination of reportability satisfies all of the requirements of 10CFR21 and 10CFR50.55(e).

C. The procedures, including criteria for determining reportability, that were developed and that are being utilized by NLS are adequate.

D. While the importance of reporting occurrences to the NRC during the design and construction phases at a nuclear facility is recognized, the more significant aspect for nuclear safety is the assurance that the conditions that led to the occurrence are identified and that the initiating conditions and the results of the occurrence are appropriately corrected.

E. Technical evaluation by personnel in the organization that has the needed technical expertise is the appropriate basis for the determination of reportability. Technical evaluation was appropriately utilized in the reportability determination for SQN NCR 1866.

F. In the resolution of deficiencies or nonconforming conditions and the determination of corrective actions, full consideration must continually be given to the implementation of the quality assurance program.

G. With regard to the specific NCR's considered during this investigation (SQN NCR 1866 and SQN NCR 1194), we concur with the original NLS determination that the NCR's are nonreportable.

H. The wire strippability/crimping problem documented in SQN NCR's 1194 and 1866 constitutes a production difficulty wherein the standard wire stripping practices resulted in the coax conductors breaking during the stripping process or when the crimp was made between the amphenol connector and the coax cable. This was confirmed when the WBN stripping practice was utilized successfully on the suspect cable at SQN.

I. Neither the qualifications of the sole electrician who is performing all of the stripping/crimping operations on the radiation monitoring cable at SQN nor his previous terminations are questionable. This conclusion is based on the fact that all terminations made on the amphenol connector end of the radiation monitoring cables in the main control room at SQN unit 1 have been made using standard stripping/crimping practices and Continental cable which has not demonstrated a strippability/crimping problem to date. Also, none of these cable terminations have failed post-installation tests that were performed by TVA Construction (CONST) personnel.

J. There were no indications that there was any unsafe attitude on the behalf of any individual interviewed in person or contacted by phone during the course of this investigation.

III. Recommendations

A. EN DES should reconcile the differences of conservatism in reporting philosophies between QEB and NLS.

B. EN DES in the disposition of NCR 1866 should recommend that CONST forces at SQN adopt the WBN stripping practice for coax cable and utilize a controlled set of stripping and crimping tools that are calibrated frequently and kept in a "fine tuned" state of repair. (NSRS understands that CONST has already done this.)

IV. Scope of Investigation

The scope of the investigation included the following:

A. Reviewed EN DES Engineering Procedures 1.26, 2.02, and 2.12; NLS PWR Procedure No. 19; and Title 10 of the Code of Federal Regulations Part 21 and Part 50.55(e) regarding reportability determinations on nonconformance reports.

B. Interviewed two NLS, three QEB, one EN DES Sequoyah/Watts Bar Design Project (SWP), and four Electrical Engineering Branch (EEB) individuals in person.

C. Interviewed one SWP, one CONST Quality Assurance, and two SQN CONST individuals by phone.

V. Details of Investigation

A. Reportability Aspect

The investigation into the reportability aspect of Case 79-10-02, employee concern over nonreportable determination on SQN NCR 1866, consisted of the following activities:

1. On October 11, 1979, a meeting was held among John Cox (NLS), Bill Kelley (NLS), Kermit Whitt (NSRS), and Terry Tyler (NSRS) to discuss NLS's reportability determination practices in general and those used for NCR 1866.

2. On October 12, 1979, a meeting was held among Jim Colley (QEB), Dan DeFord (QEB), Bill Trout (QEB), Kermit Whitt (NSRS), and Terry Tyler (NSRS) to discuss all the aspects of [redacted] concern.

3. From October 10, 1979, to October 16, 1979, EN DES Engineering Procedures 1.26, 2.02, and 2.12: NLS-PWR Procedure 19; and Title 10 of the Code of Federal Regulation Parts 21 and 50.55(e) were reviewed for nonconformance reporting requirements.

   The key aspects of this part of the investigation are as follows:

   This case was developed because at least one employee of QEB believed that a problem experienced at SQN by an electrician in the stripping and crimping of coax conductors in a radiation monitoring cable represented a reportable condition under 10CFR-50.55(e). The stripping and crimping problem was recorded and brought to the attention of EN DES through the use of a nonconformance report, NCR 1866. The NCR was marked "not significant" by CONST personnel, but when the NCR was reviewed by the Office of Engineering Design and Construction (OEDC) Quality Assurance Staff, the NCR was upgraded to significant and sent to NLS for a reportability determination. NLS evaluated the NCR and determined that the condition described in the NCR was not reportable under 10CFR50.55(e) or 10CFR21.

   A subsequent discussion with the employee that had the concern revealed the following:

   a. His concern was relative to the reportability determination process, wherein he was of the opinion that the reportability system had broken down in general and that a conservative approach was not being practiced. Therefore, in his opinion TVA was failing to report many occurrences that the NRC and the nuclear industry should be aware of.

   b. The termination problem was reportable because there was a significant breakdown of a portion of the QA program for the plant, and the condition represented significant damage to a component which would require extensive evaluation and repair to establish the adequacy of the component to perform its intended safety function. He also thought it should be reported on the basis of generic implications to other facilities in the nuclear industry.

   During the course of the investigation, the EN DES procedures for handling NCR's and for determining reportability were reviewed. Throughout these procedures, the responsibility for determining reportability is assigned to NLS. The internal NLS procedure, NLS-PWR Procedure 19, used for implementing the requirements of the EN DES procedures was also reviewed. This procedure establishes worksheets and criteria to be used in determining reportability. The NLS procedure specifies in the Policy Section that when the reportability of an item is in doubt, it is to be considered reportable. The instructions contained in the procedure were determined to be adequate.

The NCR that served as the catalyst in the initiation of this employee concern was NCR 1866. This NCR reported that one wire of an eight-conductor cable used in the radiation monitoring system tended to break during the stripping and crimping process. During the evaluation process, NLS personnel concluded that the problem was caused by a less-than-adequate insulation stripping technique employed by the electrician making the terminations. It was also learned that the same electrician had been making terminations with this type of cable for several years and had experienced similar problems in the past.

The NSRS investigation did not uncover evidence to support the precept that the wire stripping problem resulted from a significant breakdown in the QA program. There may have been a deficiency in the implementation of the QA program which contributed to or caused a delay in the identification of the wire stripping problem. Such a deficiency does not, however, appear to constitute a significant breakdown in the QA program.

As part of the evaluation of NCR 1866, EEB determined that the cable in question was acceptable for use.

The electrician responsible for the terminations at SQN has been instructed in an acceptable method for stripping the insulation from the problem wire and has demonstrated that he can make satisfactory terminations when he uses the correct technique and tools. There is no evidence that the terminations made by the electrician at SQN prior to the identification of this occurrence are not acceptable. The indications are that difficulties were experienced in making some of the terminations because of the stripping and crimping process. This difficulty required repeat termination work, but there is no indication that the eventual terminations were faulty. All the terminations are tested by CONST prior to being functionally tested during the preoperational testing program. Based on the foregoing discussion, NLS concluded for the purpose of reportability that no extensive evaluation or component repair was required. NSRS concurs with this assessment.

Generic considerations associated with potential component defects are not required to be reported per 10CFR50.55(e). Since the cable was determined to be acceptable for use, there were no generic defects to report.

The OEDC Manager has determined that reportability determinations should be made on the basis of technical evaluations. The Director of EN DES has concurred with that determination. NSRS also concurs. This determination does not restrict the responsibility of the QA staffs for QA and safety aspects of NCR evaluations and resolutions including assessments of corrective actions.

B. Electrician and Cable Qualifications Aspect

As a continuation of the investigation into Case 79-10-02, the following individuals were contacted with regard to the qualifications of the SQN electrician and radiation monitoring cable that are the subject of SQN NCR 1866:

	Individuals	Organization	Date Contacted	Contact Method
1.	Joe Bradley	EEB	10/12/79	Meeting
	David Dayton	EEB	10/12/79	Meeting
	Bill Mita	EEB	10/12/79	Meeting
	Tony Pagano	EEB	10/12/79	Meeting
2.	John Flemings	CONST-QA	10/15/79	Phone
3.	Ron Yost	CONST-Training Office at SQN	10/15/79	Phone
4.	Jim Holt	SWP	10/15/79	In Person
5.	Jack Prince	SWP	10/15/79	Phone
6.	Tom Miller	CONST-Coordination Unit at SQN	10/16/79	Phone

The key points derived from the contacts with these persons are as follows:

SQN NCR 1866 was prepared by Tom Miller of the SQN Unit 1 CONST Coordination Unit on SQN Unit 1 when problems were experienced with the termination of the control room end of the two radiation monitoring cables associated with RE-90-133 and RE-90-140 that were repulled in conduit to eliminate a noise problem on the circuits. Teledyne cable was used in the first repull attempt when the termination problems were experienced. All of the other radiation monitoring system terminations in the Unit 1 main control room had been made without difficulty using cable supplied by Continental exclusively. (Note: As SQN NCR 1194 points out, a prior attempt had been made to use radiation monitoring cable manufactured by Times. However, due to strippability problems encountered with this cable, Continental cable was used exclusively for the radiation monitoring system in SQN Unit 1.)

When the problem with the Teledyne cable was discovered, the two Teledyne cables that had been pulled in conduit were removed and replaced with Continental cable. The Continental cable was terminated without difficulty. (Note: EN DES had not officially dispositioned NCR 1866 at the time of this investigation.) All of the terminations to date have successfully passed continuity, high potential, and functional tests performed by CONST prior to the radiation monitoring system being transferred to POWER for preoperational testing. Some Teledyne cable has been pulled in SQN Unit 2; however, they are awaiting resolution of SQN NCR 1884 prior to terminating the cable.

The journeyman electrician who was having problems stripping and terminating the cable discussed in SQN NCR 1864 possessed the qualifications required by TVA for terminating the radiation monitoring system cable. The qualifications included: possession of a journeyman electrician card certifying that the person has the skills and the proficiency necessary to perform all aspects of the tasks assigned to the electrical craftspersons at a nuclear plant, and documentation that the electrician had attended an in-house training course conducted at SQN wherein a film depicting all the tools, fittings, stripping methods, termination methods, and acceptance criteria necessary to accomplish successful termination of the radiation monitoring cable was shown. (Note: The stripping and termination methods depicted in this film were being utilized when the stripping and termination problems were encountered with the cable supplied by Times and Teledyne.)

The radiation monitoring cable in question is supplied to TVA from three manufacturers, Continental, Times, and Teledyne, for both SQN and WBN. No special instructions with regard to stripping and/or terminating the cable were supplied from any of the manufacturers with the cable. The cable was manufactured and tested in accordance with applicable MIL Standards as defined in the contract prior to shipment to TVA.

The seven-strand coax signal conductors in the special eight-conductor radiation monitoring system cable are the only conductors in this cable that have exhibited strippability and termination problems. Tests have shown that it only requires application of 7 to 9 pounds of force in tension to break the seven conductors and application of approximately 100 pounds of force in tension to break the insulation and the seven conductors. Also, normal variations in the manufacturing process for this type of cable will result in a variety of surface tensions between the seven-strand conductor and the coax insulation material. Subsequently, the need to have calibrated stripping and termination tools that are kept in a good state of repair is an essential ingredient in being able to strip and terminate the coax conductors.

When the termination of the radiation monitoring cable started at WBN, similar difficulties were experienced in being able to strip and terminate the seven-strand coax conductors. However, after trial and error including use of a heated stripping tool as recommended by Teledyne, the electricians at WBN discovered that if the insulation is stripped in 1/16-inch bites using a controlled set of stripping and termination tools maintained in a "fine tuned" and calibrated state, the seven strand coax signal conductors can be repeatedly stripped and terminated successfully. The WBN method was successfully demonstrated on all of the suspect cables at SQN by WBN electricians.

Based on the above, neither the cable nor the electrician's qualifications have been determined to be suspect by this investigation. Subsequently, no nuclear safety problem has been determined to exist.

VI. References

A. NLS 10CFR50.55(e) and 10CFR21 reportability worksheets for SQN NCR 1866 (NEB 791002 158).

B. SQN NCR 1884 contained in reference 1 (NEB 791002 158).

C. NLS 10CFR50.55(e) and 10CFR21 reportability worksheets for SQN NCR 1194 (MEB 781001 359).

D. EN DES Engineering Procedure 1.26, Revision 3, All Nuclear Projects, "Nonconformances Reporting and Handling by EN DES" (ESS 781102 202).

E. EN DES Engineering Procedure 2.02, Revision 2, All Nuclear Projects (except STRIDE), "Handling of Conditions Potentially Reportable Under Title 10 of the Code of Federal Regulations, Parts 21 and 50.55(e)" (ESS 790518 205).

F. EN DES Engineering Procedure 2.12, Revision 1, All Nuclear Projects, "Implementation of Title 10 of the Code of Federal Regulations, Part 21 (10CFR21)" (ESS 780811 206).

G. NLS-PWR Procedure 19, "Nonconformances-Determination of Reportability and NRC Notification"

H. Memorandum from G. H. Kimmons to Roy H. Dunham, Joseph P. Knight, and Horace H. Mull dated December 1, 1977, subject, "All Nuclear Plants - Reporting to NRC Per 10CFR21, 10CFR50.55(e), and Responses to NRC Inspection Reports - OEDC Responsibilities" (EDC 771202 001).

I. Memorandum from Roy H. Dunham to J. L. Parris and D. R. Patterson dated December 2, 1977, subject, "All Nuclear Plants - Reporting to NRC Per 10CFR21, 10CFR50.55(e), and Responses to NRC Inspection Reports - OEDC Responsibilities" (QAS 771202 001).

TVA 64 (OS-9-65)
UNITED STATES GOVERNMENT

Memorandum

TENNESSEE VALLEY AUTHORITY

TO : G. H. Kimmons, Manager of Engineering Design and Construction, W12A9 C-K

FROM : H. N. Culver, Chief, Nuclear Safety Review Staff, 249A HBB-K

DATE : April 23, 1980

SUBJECT: EMPLOYEE CONCERN - CASE NO. 79-12-01 - SAFETY CONCERN ON SEQUOYAH NUCLEAR PLANT ERCW PUMPING STATION

Attached is the NSRS report of the findings resulting from our investigation of the subject employee concern. The employee concern that the SQN FSAR treatment of the ERCW pumping station was inadequate was found to be valid in the areas of hazards analyses and foundation exploration. Recommendations are made in the report that the SQN FSAR be amended to address the potential collision of a barge(s) traveling in the upstream direction with the intake structure and to discuss the ERCW pumping station exploration and improvement efforts. The schedule for amending the FSAR should support the scheduled licensing date for unit 2.

The report indicates that the level and degree of review provided for the design and construction of the ERCW pumping station were adequate. No information was identified that would indicate the presence of a physical deficiency in the pumping station due to material or workmanship. NSRS also concluded that the processing of an NCR is not an appropriate method for initiating an FSAR change.

You are requested to inform NSRS of your plans for implementing the recommendations presented in the report by May 21, 1980. If you have any questions, we will be pleased to discuss them.

H N Culver

H. N. Culver

KWW:LML
Attachment
cc: H. G. Parris, 500A CST2-C

Sent 4-24-80

NSRS FILE

TENNESSEE VALLEY AUTHORITY

NUCLEAR SAFETY REVIEW STAFF

EMPLOYEE CONCERN - CASE NO. 79-12-01

Subject: Tennessee Valley Authority
Sequoyah Nuclear Plant

Required Material not in Sequoyah FSAR - Safety Concern on ERCW Pumping Station

Period of Investigation: January 7, 1980 - February 8, 1980

Investigators: Janice O. Vantrease — 4-21-80
Janice O. Vantrease — Date

Kermit W. Whitt — 4/21/80
Kermit W. Whitt — Date

Approved by: H. N. Culver — 4/24/80
H. N. Culver, Chief — Date
Nuclear Safety Review Staff

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I. EMPLOYEE CONCERN - CASE NO. 79-12-01

On December 1, 1979, the Nuclear Safety Review Staff (NSRS) received an employee concern from [redacted] A complete statement of [redacted]'s concern is provided in Appendix A to this report. In summary, [redacted] concerns were:

A. There is no description in the Sequoyah Final Safety Analysis Report (FSAR) of the geologic foundation exploration and improvement for the Essential Raw Cooling Water (ERCW) pumping station.

B. The FSAR description of the hazards to the ERCW pumping station due to river traffic is misleading and incomplete. In particular, the following problem areas exist:

1. FSAR section 2.2.3 states that the ERCW pumping station is protected by its inland location and the skimmer wall as shown in figure 2.2-2. However, this must be discussing the Condenser Circulation Water (CCW) pumping station. Moreover, the ERCW pumping station is not enclosed by the skimmer wall.

2. Figure 2.1-4 of the FSAR does not show the dike or skimmer wall adjacent to the ERCW pumping station.

3. There is no discussion in the FSAR of the capability of the skimmer wall to withstand a collision.

4. Neither section 2.2.3 nor section 9.2.5.2 of the FSAR discusses collision hazards during nonflood conditions.

C. [redacted] questions whether proper procedure was followed on the design review of the station and whether sufficient TVA and NRC review was obtained.

D. [redacted] was told by several people that his concerns were not the basis for a Nonconformance Report (NCR); however, he feels that it should be investigated.

II. BACKGROUND

[redacted] has been dealing with the above concern for an extended period. He has discussed a portion of his concern, namely the collision hazard of a barge traveling upstream, with his supervisor and proposed a NCR which was dispositioned in accordance with an EN DES procedure and determined to be inappropriate. In addition, [redacted] by form TVA 45D, requested the Quality Engineering Branch (QEB) to investigate two concerns: (1) the lack of FSAR description of the geologic studies for the ERCW pumping station and (2) the lack of FSAR description of the ability of the ERCW pumping station to withstand an impact of a barge traveling upstream.

The QEB conducted an investigation as requested and recorded the findings in a memorandum to QEB Files from W. E. Troutt and W. D. DeFord dated November 23, 1979. The results of the investigation indicated that: (1) The work performed by TVA in the area of geologic investigations relating to the SQN ERCW pumping station had been adequately addressed and subsequently accepted by NRC. No further investigations were felt to be necessary with respect to the adequacy of geological considerations for the ERCW pumping station. (2) The ability of the ERCW pumping station and skimmer wall to withstand barge impact from a string of barges traveling upstream was not addressed in the FSAR because it was not included in the design basis for the pumping station. The discussion as to whether this is a credible event that should be considered was not considered within the scope of QEB.

Shortly after receiving the results of the QEB investigation, [redacted] requested NSRS to investigate his concern.

III. CONCLUSIONS

A. The foundation exploration and improvement effort for the ERCW pumping station was not sufficiently addressed in the FSAR to satisfy the requirement of 10CFR50.34. (see section IV.A.)

B. The FSAR treatment of the analysis of hazards to the ERCW pumping station is inadequate. (See section VI.B.)

C. The level and degree of review afforded the ERCW pumping station was in accordance with standard practice and appears to have been adequate. (See section VI.C.)

D. The processing of an NCR is not an appropriate method for initiating FSAR amendments. (See section VI.D.)

IV. RECOMMENDATIONS

A. The FSAR should be amended to address the foundation exploration and improvement for the ERCW pumping station. A level of detail equivalent to that incorporated in the FSAR for other Category I structures should be provided. (See section VI.A.)

B. The FSAR should be amended to address the potential hazards to the ERCW pumping station. The amendment should be worked on a schedule to support unit 2 fuel loading and should include the following as a minimum:

1. A clear distinction between the description of the ERCW and CCW pumping stations.

2. Updated figures to properly correspond to the FSAR text. Specifically, figures 2.1-4 and 2.2-2 do not appear to be in complete agreement with the text.

3. A description of the methodology utilized in addressing the potential hazards resulting from collisions during nonflood conditions, including the possible collision of a barge travelling in the upstream direction.

V. SCOPE OF INVESTIGATION

The scope of the investigation included interviews with personnel and review of documentation to determine whether the SQN ERCW pumping station foundation exploration and improvement activities have been sufficiently described in the SQN FSAR; whether the FSAR treatment of the analysis of hazards to the ERCW pumping station is adequate; whether the technical review of the ERCW design effort was adequate; and whether the processing of an NCR is an appropriate means to initiate FSAR amendments. The SQN ERCW pumping station is required to be in service prior to the licensing of unit 2. It is not required for unit 1 operation. Therefore, the conclusions and recommendations of this report apply to activities associated with licensing of unit 2 and the operation of both units after the licensing of unit 2. The findings of this investigation do not affect the testing or power operation of unit 1.

VI. DETAILS OF INVESTIGATION

The SQN ERCW system is a seismically qualified, safety-related system that is required to operate during normal plant operation and emergency conditions. It is designed to provide cooling water to equipment in both the primary and secondary portions of the plant. The ERCW system is essential for plant cooldown and mitigation of consequences during and subsequent to postulated accidents. For this reason, the ERCW system was designed to prevent a single system failure from limiting the ability of the engineered safety features to perform their functions in the event of natural disasters or plant accidents. The system consists of eight essential raw cooling water pumps, four traveling water screens, four screen wash pumps, four strainers, and associated piping and valves. All eight pumps, the traveling water screens, four screen wash pumps, and four strainers are located with the ERCW pumping station and are housed and/or supported by the intake structure. Based on this information, the importance of maintaining the integrity of the ERCW pumping station is easily recognized.

A. ERCW PUMPING STATION FOUNDATION EXPLORATION

10CFR50.34(a) states that, "Each application for a construction permit shall include a preliminary safety analysis report" (PSAR). 10CFR50.34(a)(4) specifies that the PSAR contain a preliminary analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of (1) the margins of safety during normal operations and transient conditions anticipated during the life of the facility and (2) the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents.

10CFR50.34(b) states that, "Each application for a license to operate a facility shall include a final safety analysis report" (FSAR). 10CFR50.34(b)(4) specifies that the FSAR shall include a final analysis and evaluation of the design and performance of structues, systems, and components with the objectives stated in paragraph (a)(4) of this section and taking into account any pertinent information developed since the submittal of the PSAR.

10CFR50.34(a)(1) indicates that special attention should be directed to the site evaluation factors identified in Part 100 of this chapter. Paragraph V(d)(4) of Appendix A of 10CFR100 states that, "Those structures which are not located in the immediate vicinity of the site but which are safety related shall be designed to withstand the effect of the Safe Shutdown Earthquake and the design basis for surface faulting determined on a comparable basis to that of the nuclear power plant, taking into account the material underlying the structures and the different location with respect to that of the site."

The above discussion appears to be sufficient basis on which to conclude that a description of the ERCW pumping station must be included in the FSAR. It is also clear that the geologic foundation exploration should be discussed. The ERCW pumping station is discussed to some extent in section 2.2.3 of the FSAR. However, the discussion is confusing because the CCW pumping station is discussed in the same section, and it is difficult to determine exactly which pumping station is being described in various paragraphs.

There are no clear-cut criteria for determining those structures that are not in the immediate vicinity of the site, but NSRS is of the opinion that the ERCW pumping station should fall within that category. The FSAR does not indicate that core samples were taken of the material directly under the pumping station. However, figures 2.5-101 and -102 show bore locations and soil investigations for access dike cells G, H, J, K, L, M, and N. In addition, a TVA report prepared by the Geologic Services Branch of the Division of Water Management entitled "Sequoyah Nuclear Plant - ERCW Pumping Station Rock and Concrete Investigations" and dated January 1978 indicates that eight boreholes in the area of the ERCW pumping station were geophysically surveyed according to standards established by the Geologic Services Branch. A memorandum from G. L. Buchanan to R. G. Domer dated August 28, 1979, transmitted a document which indicated that 32 coreholes and 41 probes had been made in the area of the ERCW pumping station and roadway cells to determine the top of rock. This document also contains a statement that "Experience from the main plant area and examination of cores for the pumping station area show that solution cavities are not a problem in the limestone facies of the pumping station foundation."

In paragraph 5.a of the details section of NRC IE Inspection Report Nos. 50-327/79-12 and 50-328/79-7, an IE inspector indicated that field observations and documentation reviews had been made regarding the ERCW pumping station. The matter of the exploration problem was closed out in the above referenced report. The inspector also indicated that the FSAR would be revised to incorporate the description of the pumping station. Findings of this investigation indicate that the ERCW pumping station exploration activites were adequately performed and documented, but they should be better described in the SQN FSAR.

B. ERCW PUMPING STATION HAZARDS

In the third paragraph of section 2.2 of the Sequoyah SER, the statement is made that, "The new essential raw cooling water intake structure will be protected against barge collision by a dike which will be constructed on the upstream side of the intake structure, and by the skimmer wall on the downstream side." This statement indicates that NRC may have been led to believe that the skimmer wall will protect the ERCW pumping station intake structure against barge collision. Since the skimmer wall is located on the downstream side of the intake structure, the only type of collision that it could protect the intake structure from would be the collision of a barge traveling upstream. This implies that the NRC could have concluded that TVA has considered and analyzed for hazards associated with the movement of barges in the upstream direction.

In section 2.2.3 of the SQN FSAR, a statement is made that the skimmer wall is designed to resist barge impact. This statement could have contributed to the NRC conclusion that the intake structure would be protected by the skimmer wall on the downstream side. Calculations made by EN DES show that the skimmer wall cells were designed to withstand a force of 50 kips and not to resist barge impact. In addition, the skimmer wall is not properly located to provide protection against the collision of a barge traveling upstream. The intake structure is unprotected on the east side. In the evaluation of a possible collision of a barge traveling downstream, credit was taken for current characteristics which would aid in carrying the barge away from the intake structure. It is not apparent that the current would be a factor in protecting the intake structure from a barge traveling upstream, since the current would be opposing the motivational force moving the barge. It appears that the FSAR does not address the possiblity of an upstream collision with the intake structure. NSRS believes that a potential upstream collision should be addressed in the FSAR. The evaluation could take the form of an analysis to show that the intake structure will withstand a barge collision or to show that the probability of such a collision is sufficiently low to be considered insignificant.

C. DESIGN REVIEW

As part of the employee concern, the question of sufficient review by TVA and NRC was raised.

1. TVA Review

The design review for the ERCW pumping station appears to have been done in accordance with EN DES-EP 3.01, Design Criteria Documents - Preparation, Review, and Approval; and EN DES-EP 4.04, Handling of Squadchecks. According to EP 3.01, design criteria are engineering requirements which provide the basis for conceptual and detail design. Design criteria are the basis for making design decisions, establishing design inputs, accomplishing design verification measures, and evaluating design changes. Design criteria are prepared for all safety-related structures, systems, and components. The EP also specifies that the thermal power engineering (TPE) branches are responsible for the development and approval of all general design criteria documents. The responsible TPE branch may prepare and approve detail design documents, or delegate the responsibility to the design implementing organization. The preliminary drafts of safety-related design criteria documents are usually reviewed through the squadcheck procedure in accordance with EP 4.04. Extensive rewrites are submitted in final draft form to the same TVA organization that conducted the initial review.

The design criteria document used for the SQN ERCW pumping station included:

a. General Design Criteria for Design of Reinforced Concrete Structures (SQN-DC-V-1.1)

b. General Design Criteria for Flood Protection Provisions (SQN-DC-V-12.1)

c. Design Criteria for ERCW Pumping Station (Steel) (SQN-DC-V-1.4.9)

d. Detailed Design Criteria for Essential Raw Cooling Water Supply Structures (SQN-DC-V-1.4.5)

SQN-DC-V-1.1 was reviewed by four TPE branches, the SQN Civil Design Project Group, and four engineering and design branches. SQN-DC-V-12.1 was reviewed by the TPE branches, three thermal power design project groups, and four engineering and design branches. SQN-DC-V-1.4.9 was reviewed by one TPE branch, four thermal power design project groups, and three engineering and design branches. SQN-DC-V-1.4.5 was reviewed by one TPE branch and three engineering and design branches. Thus, it would seem that the design review afforded the SQN ERCW pumping station by TVA was well within the established review practice.

The appropriateness of designing and constructing the ERCW pumping station before it is described in the FSAR was questioned by the concerned employee. TVA is responsible for designing and constructing the ERCW pumping station such that it will perform its intended function. The NRC is responsible for reviewing and evaluating the design and construction to assure that these processes are carried out such that the risk to the public from operation of the plant is kept as low as possible. With the exception of the potential collision of a barge(s) traveling upstream, the NRC appears to have received the necessary information from TVA to allow them to perform their evaluation. The FSAR is required to be up-to-date and show actual plant conditions at the time the plant is licensed. The ERCW pumping station is not required until unit 2 is licensed. We believe that the FSAR should be kept as current as possible. However, we do not consider the deliquency of the FSAR update in this case to be a safety problem since the information regarding the ERCW pumping station has been provided in support of the application for license. The SQN FSAR should be updated to properly describe the ERCW pumping station, including foundation exploration, as far in advance of unit 2 fuel loading as practical.

2. <u>NRC Review</u>

It is not within the scope and authority of NSRS activities to critically assess the adequacy of the NRC review of the ERCW pumping station. However, our efforts in this area indicate that NRC did evaluate the TVA documentation and physical efforts relating to the ERCW pumping station.

The NRC Office of Inspection and Inforcement (IE) evaluated the deficient foundation preparation for concrete cells for the access roadway which was identified in NCR 37. The evaluation was performed through field observations and review of documentation and was closed out in IE inspection report Nos. 50-327/79-12 and 50-328/79-7 dated March 26, 1979. The NRC Office of Nuclear Reactor Regulation (NRR) evaluated the design of the ERCW pumping station. The results of the evaluation were reported in the Safety Evaluation Report (SER) which was issued in March 1979. The SER indicated that the ERCW design was basically acceptable although NRR still had a few questions primarily relating to the subsurface condition for ERCW pipes and seismic Category I electrical conduits between the main plant area and the ERCW pumping station. Supplement 1 to the SER was issued in February 1980. This supplement indicates that the NRC review of the ERCW pumping station had been completed and the station had been determined to be acceptable. Section 2.6.1.3 of the supplement states that the "subsurface conditions for the ERCW pipes and seismic Category I electrical conduits that extend approximately 2400 feet between the main plant area and the ERCW pumping station in Chickamauga Reservoir were shown to consist of residual soils, described as dense silty gravels, hard clays, and soft medium silts. Alluvial clay soils averaging 13 feet in thickness existed on the reservoir bank and reached thickness of 30 feet beneath the ERCW pumping station. Beneath the alluvial clay soils, the weathered shale zone was shown by exploration to average 10 feet in thickness." Section 2.6.2.3 of the SER supplement states that "Seismic Category I electrical conduits and ERCW piping leaving the main plant area are founded on natural soils and travel approximately 2100 feet ... to the concrete supporting slab that is founded on H-piles driven to rock. The supporting slab, founded on piles then carries the piping, the electrical conduits, and the access road to six interlocking sheet pile cells that approach the ERCW pumping station in Chickamauga Reservoir. The pumping station is also founded on interlocking sheet pile cells. All sheet pile cells were constructed by driving the sheet piling to bedrock and then excavating to bedrock within the cell, prior to backfilling with tremie concrete." Section 2.6.3 of the SER supplement provides the NRC conclusions regarding foundation evaluations for the main power building complex, the diesel generator building area, and the ERCW pumping station area. The final conclusion regarding the foundation evaluation is stated as follows: "In summary, based on our review of the information provided as discussed above, we conclude that the site and plant foundations are acceptable for safe operation of units 1 and 2."

D. APPROPRIATENESS OF NCR

The method for handling NCR's within EN DES is presented in EN DES-EP 1.26, Nonconformances - Reporting and Handling by EN DES. The statement of purpose of EP 1.26 strongly suggests that NCR's are to be utilized for controlling characteristics of hardware and documentation relating to hardware. An NCR is defined as a deficiency in characteristics, documentation, or procedure which renders the quality of an item unacceptable or indeterminate. An item is defined as any level of unit assembly, including structure, system, subsystem, subassembly, component, part, or material. A characteristic is defined as any property or attribute of an item, process, or service that is distinct, describable, and measureable, as conforming or nonconforming to specified quality requirements. It could conceivably be argued that inadequacies or suspected inadequacies in the FSAR represent deficiencies in documentation of analyses, evaluation, or description of activities relating to items. However, this would be stretching a point to an extreme. NSRS believes that the processing of an NCR to effect a change to the FSAR is not the appropriate approach.

The methodology for amending the FSAR by members of EN DES is desribed in EN DES-EP 2.04, Amendments to Safety Analysis Reports - Preparation, Review, and Approval, and EN DES-EP 2.05, Maintenance of Final Safety Analysis Reports in a Current Status. This methodology should be utilized to the fullest extent practical. If this approach fails to accomplish the appropriate results, the specific concerns should be resolved through office management.

VII. REFERENCES

A. Report entitled "Sequoyah Nuclear Plant - ERCW Pumping Station Rock and Concrete Investigations," TVA, Division of Water Management, Geologic Services Branch, January 1978.

B. Design Criteria SQN-DC-V-7.4, "Design Criteria for the Essential Raw Cooling Water System," September 16, 1970

C. Memorandum from W. E. Troutt and W. D. DeFord to CEB Files dated November 23, 1979, on "Sequoyah Nuclear Plant - Description of ERCW Pumping Station - Report on a Request for QA Evaluation" (QAS 791123 001)

D. Memorandum from R. G. Domer to D. R. Patterson dated January 11, 1978, on "Sequoyah Nuclear Plant Units 1 and 2 - NRC Reportable Condition - Deficient Concrete Cells in the ERCW Pumping Station Access Roadway - Fifth Interim Report" (CEB 780111 004)

E. Memorandum from G. L. Buchanan to R. G. Domer dated August 28, 1978, on "Sequoyah Nuclear Plant - ERCW Pumping Station and Roadway Cells - Geologic Summary for Submittal to NRC" (CDB 780828 012)

F. Memorandum from D. R. Patterson to L. M. Mills dated November 21, 1978, on "Sequoyah Nuclear Plant Units 1 and 2 - Deficient Foundation Preparation for Concrete Cells for the ERCW Pumping Station Access Roadway - NCR-37 - Revised Final Report" (MEB 781121 378)

G. Memorandum from R. G. Domer to D. R. Patterson dated July 28, 1978, on "Sequoyah Nuclear Plant Units 1 and 2 - Deficient ERCW Access Roadway Cells E and F - (NCR-37) Final Report" (CEB 780731 013)

H. Letter from C. E. Murphy to H. G. Parris dated March 26, 1979 (MEB 790329 380)

I. Memorandum from D. R. Patterson and R. G. Domer to Roy H. Dunham dated February 25, 1974, on "Sequoyah Nuclear Plant Units 1 and 2 - ERCW System Relocation of ERCW Pumping Station"

J. Memorandum from J. H. Coulson to G. L. Buchanan dated January 26, 1978, on "Sequoyah Nuclear Plant - ERCW Pumping Station, Cells A Through F - Approval of Rock Foundation after Placement of Concrete" (CDB 780126 003)

K. U.S. NRC Report NUREG-0011 entitled "Safety Evaluation Report Related to the Operation of Sequoyah Nuclear Plant Units 1 and 2," March 1979 (Sections 2.2, 2.5, 9.2.2, and 9.2.3)

L. Final Safety Analysis Report for Sequoyah Nuclear Plant Units 1 and 2

M. Letter from J. E. Gilleland to J. P. O'Reilly dated November 29, 1978, on "Sequoyah Nuclear Plant Units 1 and 2 - Deficient Foundation Preparation for Concrete Cells for the ERCW Pumping Station Access Roadway - NCR-37 - Revised Final Report" (DES 781201 017)

N. Transcription of taped interviews with the following persons:

Name and Title	Date
Robert O. Barnett, Chief, Civil Engineering Branch	January 9, 1980
Ronald D. Guthrie, Senior Civil Engineer	January 9, 1980
[redacted]	January 23, 1980
John F. Cox, Senior Nuclear Engineer	February 19, 1980

O. Regulatory Guide 1.70 (Revision 10) entitled "Standard Format and Content of Safety Analysis REports for Nuclear Power Plants," October 1972

P. Regulatory Guide 1.91 entitled "Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants," January 1975

Q. Code of Federal Regulations, 10CFR50.34

R. Code of Federal REgulations, 10CFR100, Appendix A, Paragraph V(d)(4)

S. EN DES Engineering Procedure 1.26, Revision 2, "Nonconformances - Reporting and Handling by EN DES," November 8, 1978

T. Design Criteria SQN-DC-V-1.1, "Design Criteria for the Design of Reinforced Concrete Structures, November 8, 1978

U. Design Criteria SQN-DC-V-12.1, "Design Criteria for Flood Protection Provisions," April 28, 1978

V. Design Criteria SQN-DC-V-1.45, "Design Criteria for Essential Raw Cooling Water Supply Structures," December 7, 1978

W. Design Criteria SQN-DC-V-1.49, "Design Criteria for ERCW Pumping Station (Steel)," May 4, 1976

X. EN DES Engineering Procedure 3.08, "Soil and Rock Investigations," August 24, 1979

Y. TVA Drawing 31W501

Z. TVA Drawing 31N307

AA. TVA Drawing 31W211

BB. TVA Drawing 10N200-01

CC. EN DES Engineering Procedure 3.01, Revision 3, "Design Criteria Documents - Preoperation, Review, and Approval," December 27, 1978

DD. EN DES Engineering Procedure 4.04, Revision 5, "Handling of Squadchecks," November 1, 1979

EE. En DES Engineering Procedure 2.04, Revision 2, "Amendments to Safety Analysis Reports - Preparation, Review, and Approval," July 6, 1978

FF. EN DES Engineering Procedure 2.05, Revision 0, "Maintenance of Final Safety Analysis Reports in a Current Status," September 20, 1976

APPENDIX A

EMPLOYEE CONCERN NO. 79-12-01 - REQUIRED MATERIAL NOT IN SEQUOYAH FSAR

SAFETY CONCERN ON ERCW PUMPING STATION

There is no description in the Sequoyah FSAR of the geologic foundation exploration and improvement for the ERCW pumping station. This information is required by the SAR Format Guide (Regulatory Guide 1.70) in paragraphs 2.5.4.3 and 2.5.4.12. See also 10CFR50.34 and 10CFR100, particularly 10CFR100, Appendix A, paragraph V(d)(4).

The evaluation in subsection 2.2.3 of the Sequoyah FSAR, of hazards to the ERCW pumping station due to transportation facilities, is misleading and in some cases actually false. The fourth paragraph under 2.2.3 states that the intake structure is protected by location, refers to figure 2.2-3, and discusses the location inland and inside the skimmer wall. The figure shows the features discussed, which obviously must refer to the CCW pumping station, and also shows the ERCW pumping station definitely enclosed within the upstream dike and the skimmer wall. This latter feature is false; the intake side of the station is actually directly exposed to river traffic and to any liquid releases, fires, or explosions which might occur in the main river channel.

The fifth paragraph under subsection 2.2.3 states that protection of the ERCW structures from river traffic is described in paragraph 9.2.5.2 and that collision probabilities are described in the response to Question 2.47. Paragraph 9.2.5.2 states that for nonflood conditions the ERCW structure is protected by its location between the dike and the skimmer wall, as shown in figure 2.1-4. That figure does not show the skimmer wall adjacent to the pumping station, but does show the structure clearly exposed to river traffic. There is no discussion of the capability of the skimmer wall to withstand collision. The response to Question 2.47 states that for nonflood conditions the station is protected from collision by location and refers to paragraph 9.2.5.2, which does not present a collision analysis for normal lake levels. The remainder of the response deals with flood conditions. The discussion of collisions under nonflood conditions is thus evaded in both places, as are discussions of fire, chemical, and explosive effects.

I have discussed these problems with various EN DES personnel. I was told that the foundation investigation and improvement work was done and has been discussed with NRC, and that it will be included in an FSAR amendment. I question whether this is a proper procedure; whether sufficient internal TVA review and agreement on the design was provided; and whether sufficient review within NRC was obtained. The PSAR design (the single CCW-ERCW pumping station) received wide review by many governmental agencies, but the new ERCW design is not even in the FSAR (i.e., the features and facets discussed above).

No one knew of the collision investigation having been made, but I was told that the ERCW pumps in the CCW pumping station would provide a backup. The FSAR states clearly on page 1.2-62(2) that those pumps will be decommissioned. Also, the new ERCW station was provided because the CCW location for the ERCW pumps was unsatisfactory. I was also told that the skimmer wall is resistant to some collision impact, but this capability is unclear. Also, the outboard face of the pumping station is not protected by the skimmer wall.

I was told, finally, that these concerns are not, in several people's opinions, a basis for an NCR being written. I believe that this possibility should be investigated.

[redacted]
December 7, 1979

79-12-02

GNS 8005070105

W. F. Willis, General Manager, E12B16 C-K

E. A. Belvin, Director, Office of Health and Safety, ROB-M

November 5, 1979

ACCIDENT INVESTIGATION - CHLORINE ACCIDENT - BROWNS FERRY NUCLEAR PLANT (BFN) - JUNE 3, 1979

Attached for your review is the final report of investigation for the subject accident which has been transmitted to the Director of Nuclear Power. The report has been revised from that transmitted by my memorandum of June 22, 1979, to the General Manager on the above subject. This revision is based on further investigation and comments made by the Office of Power during the June 27, 1979, meeting with you.

During the meeting on June 27, 1979, the Office of Power proposed to prevent future accidents of this type at BFN by removing all chlorine and substituting a temporary sodium hypochlorite (NaOCl) system. This system will ultimately be replaced by a permanent NaOCl generation facility. A review of the properties of NaOCl revealed that it presents many of the same hazards as chlorine. In fact, NaOCl will release elemental chlorine under certain conditions. — Mix w/ acid like H_2SO_4?

Volatile?
How likely/How much

We believe that the immediate cause of the accident and significant findings of the investigation remain unchanged from my June 22, 1979, memorandum. However, additional explanation may be useful regarding our finding pertaining to the compromise of the unit 3 control room.

This finding resulted from discussions with several employees involved in the accident who reported that chlorine entered the unit 3 control room. The control room atmosphere was degraded to the point that the unit operator was exposed to airborne chlorine concentrations ranging from 15 to 30 parts per million. These concentrations, which are based on available medical and toxicological data, apparently exceed guidelines issued by the Nuclear Regulatory Commission (NRC).

NRC guidelines do not clearly define control room habitability in terms of operator exposure to hazardous chemicals. By copy of this memorandum, I am asking the Nuclear Safety Review Staff to evaluate this accident with respect to control room habitability and equipment reliability as a result of exposure to hazardous chemicals or gases.

Effect on unit &
Safety Limits
Masks

Based upon the measures proposed by the Division of Nuclear Power to prevent future accidents of this type at BFN, we believe our original

MDS

W. F. Willis
November 5, 1979

ACCIDENT INVESTIGATION - CHLORINE ACCIDENT - BROWNS FERRY NUCLEAR PLANT (BFN) - JUNE 3, 1979 (3200)

recommendations (as modified in the attached report to account for the presence of NaOCl) are valid.

We are available to discuss the report if you so desire.

E. A. Belvin

HEL:DRK:DH
Attachment
cc (Attachment):
C. Bonine, Jr., W12A1 C-K
E. G. Beasley, 249B HB2-K

MEDS, E4B37 C-K

SUMMARY

At approximately 8:55 a.m. on Sunday, June 3, 1979, a chlorine leak was discovered in the chlorination building at Browns Ferry Nuclear Plant (BFN). Before the leak could be isolated and contained, at least 11 TVA employees received possible chlorine exposures. One employee was exposed in detecting the leak. Three employees were exposed while attempting to contain the leak. At least three employees were exposed in the unit three control room, and at least three more were exposed in the turbine building. A Public Safety Service (PSS) employee stationed near the chlorination building was also exposed. Five employees were admitted to Decatur General Hospital, Decatur, Alabama, and were released on June 4, 1979, with no apparent ill effects. No other employees were hospitalized.

BACKGROUND

For a period of about 30 days, twice a year (approximate dates May and September), chlorine is injected into the raw cooling water (RCW) system to control Asiatic clams. The most recent chlorination process at BFN began on Thursday, May 31, 1979.

The chlorination building at BFN is adjacent to the east wall of the turbine building, approximately 235 feet from the unit three control bay. It is designed to house twenty-six 1-ton containers of chlorine pressurized to approximately 100 psig. However, information gathered during the investigation revealed that the entire process usually requires no more than six containers. There is reportedly no long-term chlorine storage at BFN.

A maximum of four chlorine containers is attached to a manifold system and one of the containers is placed in service. Liquid chlorine passes from the manifold into a process room within the chlorination building where it enters an evaporator as a liquid and exits as a gas. Chlorine gas from the evaporator enters a pneumatically controlled regulator valve designed to prevent downstream gas pressure greater than 50 psig. If abnormal pressure conditions occur, this valve and its control system act as a safety device by restricting gas flow.

From the regulator valve chlorine gas enters a chlorinator that maintains flow rate. A vacuum created by a 3-inch venturi-type ejector in the turbine building draws chlorine gas from the chlorinator into the RCW system.

Prior to being placed in service, the chlorination system, up to the regulator valve, must be air tested with a soap solution and all leaks repaired and retested. In addition, all valves and fittings up to this point must be disassembled, checked for deterioration, repaired, and reinstalled. This examination of the chlorination system is required by BFN Surveillance Instruction (SI) 4.11.A.1-f, Flushing of the High Pressure Fire Protection System and Addition of Biocide to the Raw Cooling Water System. This SI also contains the mechanical maintenance instructions for the system. Data sheets that are a part of the SI indicate that the system was examined on May 31, 1979, by personnel from the BFN Maintenance Section.

Internal valve examination for proper function, seating, and leakage was not performed on this date nor has it been a part of previous system inspections. This is consistent with past interpretation of the SI which involved examining system flow passages only. The SI does not require examination of the regulator valve which is considered a critical component of the chlorination system.

THE ACCIDENT

At approximately 7:45 a.m. CDT on Sunday, June 3, 1979, a Radiochemical Laboratory Analyst, Kenneth K. Richards, smelled chlorine when he entered the chlorination building to conduct a routine inspection of the chlorine monitor located on the south wall of the chlorination building and adjacent to the process room. He informed PSS Clerk [redacted], Mr A who was stationed at PSS post 8, approximately 35 feet north of the chlorination building, that there might be a small chlorine leak and advised him to evacuate the area should he hear the monitor's alarm.

At approximately 8:55 a.m. CDT, Mr. Richards returned to the chlorination building to determine the cause of high-chlorine content in the RCW system. Upon entering the process room, he discovered leaking chlorine gas.

Mr B

Mr. Richards contacted the unit three Operator, [redacted], who in turn notified Assistant Shift Engineer (ASE), [redacted]. Mr C The chlorine monitor had failed to activate audible alarms locally in the chlorination building and remotely in the unit three control room.

C

Mr. [redacted] proceeded to the unit three control room where he detected the odor of chlorine. He and Assistant Unit Operator (AUO), [redacted] Mr D [redacted], obtained self-contained breathing apparatus (SCBA) from behind the control panel and proceeded to the chlorination building. AUO, Mr E [redacted], replaced Mr. D [redacted] in the unit three control room and obtained SCBA for Mr. B [redacted] and himself. Mr. B [redacted] stated that he used his SCBA for occasional breaths of air but did not don it. Mr. D [redacted] ordered the turbine building evacuated and dispatched AUOs, Bobby L. Holbrook and Pamela J. Siegler, to ensure that this was accomplished. Shift Engineer, Billy L. Roth, also walked through the turbine building to ensure its evacuation. Both Messrs. Holbrook and Roth reported smelling chlorine in this area.

C D

Messrs. [redacted] and [redacted] arrived at the emergency equipment cabinets adjacent to a propane storage facility adjoining the chlorination building and found the cabinets locked. Mr. C [redacted] requested the keys to the cabinets but later decided that, due to the urgency of the situation, further delay could not be tolerated. The keys were not located during the emergency; consequently, equipment available at the site, including additional SCBA, protective clothing, and a chlorine container repair kit, were of no use to Messrs. [redacted] C and [redacted] D.

The two employees, wearing SCBA, entered the chlorination building. They went to the chlorine container in use and closed its supply valve. The system remained pressurized. They entered the process room and closed the inlet valve to the evaporator. Finding that this action did not relieve system pressure, they attempted to close the valve between the evaporator and regulator. They could not close this valve by hand. Mr. C [redacted] discovered that chlorine was entering his face mask. He and Mr. D [redacted] left the chlorination building, and Mr. C [redacted] examined his SCBA. He could not find any leaks and determined that he had a satisfactory seal between his mask and face.

Mr. C [redacted] called Mr. B [redacted] from PSS post 8 and requested a wrench to close the valve. Mr. A [redacted] had evacuated this location at approximately 10 a.m. CDT upon the direction of his supervisor, Lieutenant Clifford Langham.

Time?

Mr. B [redacted] dispatched Mr. E [redacted] to the chlorination building with a wrench. At about this same time, Assistant Shift Engineer, [redacted] MrF [redacted], who was unaware of the situation, arrived in the unit three control room and instructed Mr. B [redacted] to activate the emergency pressurization fans to prevent the influx of additional chlorine. Mr. F [redacted] then left the area since the chlorine caused him discomfort.

When Mr. E [redacted] arrived at the chlorination building, Mr. D [redacted] returned to the unit three control room. Messrs. C [redacted] and E [redacted], wearing SCBA, entered the process room in the chlorination building and, using the wrench, were able to close the valve and relieve system pressure. Mr. [redacted] E stated that he was able to isolate the leak using a 50-percent solution of ammonia and water and that the major leak was at the regulator valve with some smaller leaks at the chlorinator.

The turbine building was ventilated by opening its doors.

The exact duration of the leak is not known; however, the Unit Operator's log indicates that the area was clear at 10:55 a.m. CDT.

POST-ACCIDENT EVENTS

Mr. B [redacted] was replaced by Mr. F [redacted] in the unit three control room at about 11:30 a.m. CDT. Since he and another employee in this area felt uncomfortable due to the presence of chlorine, they opened the control room doors to allow the gas to disperse. Mr. Roth, after conferring with the Plant Superintendent, Jerrold G. Dewease, Sr., decided that Messrs. C, B, D, and E [redacted] should receive medical attention. At about 11:45 a.m. CDT, these employees left for Decatur General

Hospital, Decatur, Alabama, since the BFN health station was closed. A short time later Lieutenant Langham advised Mr. A [redacted] that he too should receive medical attention, and he proceeded to Decatur General Hospital also. All five employees were released on June 4, 1979, with no apparent ill effects.

Forms TVA 2275, Report of Occupation-Related Condition or Disease, completed on June 5, 1979, for the five hospitalized employees, indicate similar symptoms, i.e., throat and nose irritation and tearing eyes. In addition, Mr B [redacted], who remained in the control room throughout the emergency, also had an infrequent cough. According to Chemical Hazards in the Workplace, by Proctor and Hughes, and the National Institute of Occupational Safety and Health Criteria Document 76-170, Occupational Exposure to Chlorine, eye irritation occurs at airborne concentrations of 7 to 8 parts per million (ppm), throat irritation at 15 ppm, and cough at 30 ppm.

The regulator valve was disassembled after the accident by Steamfitter, John E. Whitt. A new diaphragm was installed since one of its two lower diaphragms had ruptured due to corrosion. Mr. Whitt stated that this valve did not leak when reassembled and pressurized with air. His opinion was that liquid chlorine entered the chlorinator causing a relief valve to open as designed and release liquid chlorine into a floor drain.

Chlorine has been removed from the chlorination building and replaced by a temporary arrangement using gravity flow to transfer sodium hypochlorite from a tank truck outside the chlorination building through a rubber hose to the ejector inlet in the process room. The Division of Power Production has committed to permanently discontinue the use of elemental chlorine for control of Asiatic clams at BFN and to install a fixed sodium hypochlorite generation facility for this purpose.

Direct workmen's compensation costs for this accident are negligible. There was no apparent property damage.

Power is responsible for items 2-23.

RECOMMENDATIONS

1. The Nuclear Safety Review Board and/or Nuclear Safety Review Staff should evaluate this accident. This evaluation should consider the degree to which BFN control rooms conform to operator habitability provisions of Nuclear Regulatory Commission Regulatory Guides 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," and 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release." This evaluation should also consider the chemical action of harmful substances on control room equipment.

This refers to the temporary truck/hose facility.

2. An in-depth hazard analysis of the temporary sodium hypochlorite arrangement should be conducted. This analysis should consider not only the hazardous properties of sodium hypochlorite but also the present use of the chlorination building for other than its designed purpose.

3. An in-depth hazard analysis of the proposed permanent sodium hypochlorite generating facility at BFN should be conducted.

4. The hazard analyses discussed in recommendations 2 and 3 should address, as a minimum, emissions containment and/or isolation so that critical plant areas are not contaminated with chlorine-bearing compounds; exhaust control, if designed for emissions containment; emissions monitoring; visual surveillance; emergency shutdown; leak isolation; and control of process variables, e.g., temperature and pressure.

5. Surveillance Instructions and Mechanical Maintenance Instructions should ensure that all components of systems using chlorine-bearing compounds are thoroughly examined before the system is placed in operation.

6. Safety and health personnel should be positively involved in the development and implementation of Surveillance Instructions and Mechanical Maintenance Instructions pertaining to systems using chlorine-bearing compounds to ensure that proper protective measures are taken.

7. Surveillance and mechanical maintenance inspections for all systems using chlorine-bearing compounds should be verified, e.g., by a formal quality assurance review.

8. Chlorine detection and alarm systems should be maintained, operated, and designed so that reliability is enhanced, to include assuring that an adequate number of detection points are available.

9. Chlorine monitors should be located so that they can be viewed without subjecting the observer to possible chlorine exposures.

10. Emergency equipment for operations using chlorine-bearing compounds should be placed in the area where it will be needed but in a place where leakage will not jeopardize its accessibility.

11. All emergency equipment cabinets should be examined for contents and accessibility.

12. An adequate emergency shower and eye wash facility should be provided near the chlorination building.

13. Personnel whose clothing becomes contaminated with chlorine should immediately remove their clothing and wash affected skin with copious amounts of water.

14. Specific emergency procedures should be developed for systems using chlorine-bearing compounds. These procedures should give step-by-step actions to take in the event of a leak.

15. All SCBA should be examined for proper functioning. The results of this examination should be verified by a quality assurance review.

16. Inhabited areas located near operations using chlorine-bearing compounds, e.g., PSS posts, should have SCBA provided for occupants.

17. Suspected releases of chlorine-bearing compounds should immediately be reported to the shift engineer.

18. Access to areas contaminated with chlorine-bearing compounds should be stringently controlled.

19. Personnel entering areas known or suspected to be contaminated with chlorine-bearing compounds should use all prescribed personal protective equipment.

20. All persons exposed to chlorine-bearing compounds above the threshold limit value (TLV) or who show visible signs of discomfort should receive medical care.

21. The placement of propane tanks adjacent to the chlorination building should be evaluated. The present arrangement places a fuel (propane) in proximity to an oxidizer (sodium hypochlorite). In addition, the chlorination building electrical equipment should be of a type suitable for use in Class I, Division II, Group D, atmospheres as defined in NFPA 70, National Electrical Code.

22. Supervisors should be made aware that forms TVA 1890, Report of Injury or Illness, are to be completed for all persons exposed to chlorine-bearing compounds above the TLV or who show visible signs of discomfort as a result of such exposure.

23. Develop and implement a system to ensure that serious accidents are reported to the Hazard Control Staff as required by Part III, "Accident Investigation Committee Procedure," of the TVA <u>Hazard Control Manual</u>.

TVA 64 (OS-9-65)
UNITED STATES GOVERNMENT

Memorandum

TENNESSEE VALLEY AUTHORITY

BFN 3-2

TO : W. F. Willis, General Manager, E12B16 C-K

FROM : E. Gray Beasley, Acting Chief, Nuclear Safety Review Staff, 249A HBB-K

DATE : January 9, 1980

SUBJECT: NUCLEAR SAFETY REVIEW STAFF INVESTIGATION OF BROWNS FERRY UNIT 3 CONTAINMENT LEAKAGE PROBLEM, DECEMBER 6-9, 1979

~~BFN-3-2~~

80-02-BFN

The attached is a revised NSRS report on the Browns Ferry Unit 3 December 6-9, 1979, incident. The revisions:

1. Delete the generic recommendations. The approach in the generic recommendations will be pursued separately.

2. Reflect that the NRC resident inspector was informed of the containment leakage problem. This change is based on supplemental information received after the investigation on January 5-6, 1980.

3. Point out that there is no way to prove that the leak existed or to prove that it did not exist prior to December 8, 1979.

4. Delete any reference to and discussion of Item 9 of NRC Bulletin 79-08.

E Gray Beasley
E. Gray Beasley

EGB:CC
Attachment
cc (Attachment):
E. A. Belvin, ROB-M
C. Bonine, Jr., E3C76 C-K
J. R. Calhoun, 716 EB-C
G. H. Kimmons, W12A9 C-K
H. G. Parris, 500A CST2-C

This report was prepared by: E. Gray Beasley
Leonard Blankner
Henry Jones
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Terry Tyler
Kermit W. Whitt

File

NUCLEAR SAFETY REVIEW STAFF INVESTIGATION OF
BROWNS FERRY UNIT 3 CONTAINMENT LEAKAGE
PROBLEM - DECEMBER 6-9, 1979
JANUARY 9, 1980

The information in this report is based on interviews with Browns Ferry Nuclear Plant personnel on January 5 and 6, 1980, and on official plant records. The 12 persons interviewed included the Superintendent and the key management, technical, operations, and craft persons on duty during the course of the incident. The plant records include the technical specifications (limiting conditions for operation); logs of the shift engineers, the assistant shift engineers, and the unit operators; various plant procedures, instructions, and correspondence; recorder charts; and records of completed tests.

Major events are listed below for convenience. (NSRS comments are in parentheses.)

Date	Time	Event
11/26	N/A	BFN-3 successfully passed the containment integrated leak rate test. (SE hatch not leaking.)
12/6	0645	BFN-3 attained initial criticality. (Primary containment required to be intact at this point.)
12/7	1215	Reactor attained rated temperature and pressure. (24-hour technical specification time limit starts for establishing drywell-to-torus differential pressure.)
12/8	1245 approx.	Differential pressure established. Technical specification time limit expired at 1215. Plant now has six hours before technical specification time limit requires a shutdown.)
	1400	Began checking for primary containment damper leakage.

Date	Time	Event
12/8	1815	Differential pressure not maintained. No shutdown started. (Technical specifications require shutdown to commence.)
12/9	0023	Began inserting control rods in preparation for orderly shutdown based on lack of pressure differential.
	0250	Drywell-to-torus differential pressure established. Shutdown aborted.
	0300	Concluded external containment leak existed. (Technical specification requires prompt notification of NRC in 24 hours.)
	0530 approx.	Leakage source located.
	0830 approx.	Leakage stopped around SE hatch seal.
	0845 approx.	Began local leak rate test on SE hatch seal.
	0930	SE hatch passed leak test.
12/10	0800	Plant Superintendent informed NRC Resident Inspector of differential pressure problem and the containment leakage problem.
	1525	NRC (Hugh Dance) notified by telephone of excessive containment leakage problem. (Mr. Dance was at BFN at this time.)
12/11	1455	NRC verified receipt of LER telecopy, following up the phone notification.

The following three sections address each of the three items identified in the NRC telephone conversation with POWER on January 4, 1980.

Failure to Conduct Safety-Related Activities by Procedures

Two large hatches are provided for the transfer of large equipment into and out of the primary containment of the Browns Ferry reactors during reactor shutdown. The equipment hatch closures consist of large blind flanges, each of which is held in place by 12 lugs held by 1-inch cap screws. The hatch closures are installed from inside the containment such that containment pressure forces the closures against the hatch face. In the event of an accident, increased containment pressure would provide additional compression on the gaskets and tend to preclude containment leakage.

The hatch sealing surfaces are fitted with two O-ring gaskets each. This arrangement provides for the performance of local leak rate testing by pressurizing between the O-rings. Leakage rate is measured by the pressure decay method, which is a conservative approach.

The hatch closures are installed at the Browns Ferry Nuclear Plant by craftsmen under the supervision of a foreman. A maintenance specialist provides technical direction and special instructions, such as the torque value necessary for the cap screws used to secure the closure. The second step in the closure of the equipment hatches is the performance of a local leak rate test to assure that the closure is sealed.

A written procedure had been prepared and had been in use since initial operation of Browns Ferry for performing the local leak rate test on the hatch closure. Appropriate signoffs were made, and records were maintained. A written procedure had not been provided for the installation of the hatch closures because instructions were available in the vendor's manual, including torque values, and the operation was considered to be a maintenance activity within the normal skill of the craftsmen performing the work. Performance of this type of activity without the use of written instructions is recognized as being appropriate by the NRC as indicated by Regulatory Guide 1.33, "Quality Assurance Program Requirement (Operations)." In this regulatory guide it is stated that, "Routine maintenance activities that require skills normally possessed by qualified personnel may not require detailed step-by-step delineation in a procedure but should be subject to general administrative procedural controls." An example provided by Regulatory Guide 1.33 of a maintenance activity that does not require a step-by-step written procedure is gasket replacement. This would appear to indicate that the hatch closures could have been removed from the containment penetrations in question for the purpose of replacing the gaskets and reinstalled without the use of a written procedure.

The question of whether or not a written procedure was necessary appears to center on whether the installation of the hatch closure was a routine maintenance activity within the normal skills of the craftsmen performing the job and whether the activity constituted an activity of sufficient safety significance to require a formal written procedure and the maintenance of written records.

Written procedures are used for the welding of containment penetrations and most work associated with the penetrations. A procedure is used for the installation of the containment vessel head, which also is bolted in place.

With the exception of the containment head, the equipment hatches are the largest openings in the containment. The maintenance of containment integrity is essential to safety. This has always been recognized, but the Three Mile Island accident has brought that realization into sharper focus.

NSRS agrees that there was no basis to require a procedure for the installation of the hatch cover prior to the December 6-9 incident. In light of that incident, NSRS agrees that a procedure is required.

Failure to Shut Down Upon Loss of Containment Integrity

The Browns Ferry technical specifications (limiting conditions for operation) require that containment integrity be established prior to the reactor going critical. Unit 3 containment leak rate tests completed on November 27, 1979, confirmed that containment leakage was within allowable limits. These tests included successful completion of a local leak rate test for the south-east equipment hatch.

To date, it is not possible to establish a reason as to why the south-east equipment hatch started leaking after successfully passing the local and integrated leak rate tests a few days earlier. The NSRS investigation of the period from the end of the containment leak rate test up to the afternoon of December 8 did not reveal any indications that would lead the operator to even suspect that the containment leakage was not within allowable limits.

Later in the day of December 8, 1979, there were some indications that could have been interpreted as excessive containment leakage; however, the operations that plant personnel were conducting late in the day of December 8 and early on December 9 were appropriate for the conditions the operators were experiencing and observing. One log indicates that some effort was being placed of finding leaks at 1400 on December 8, 1979. The malfunction of a small solenoid valve and a flow indicating control valve in the N_2 makeup system tended to mask the situation and lead one away from concluding that containment leakage was excessive. The situation was further complicated by the drywell-to-torus ΔP problem suspected to be caused by previously leaking drywell-to-torus dampers.

From the data collected, it is apparent that the containment leakage rate exceeded allowable limits about noon on December 8, 1979. The operators identified excessive leakage about 0300 on December 9, 1979, the leak was located about 0530 on December 9, 1979, and leakage was reduced to well within acceptable limits about 0830 on December 9, 1979. NSRS concludes that there is no way to prove that the containment was leaking excessively and there is no way to prove that containment leakage was within allowable limits between criticality on December 6, 1979, and about noon on December 8, 1979. Further, NSRS concludes that excessive leakage did exist from about noon on December 8, 1979, to about 0830 on December 9, 1979.

<u>Failure to Make Prompt Reports</u>

The technical specifications are very confusing on reporting to NRC. Reporting requirements are contained in several different places in the technical specifications, are ambiguous, and are sometimes conflicting. The situation is confusing at best. By reading certain portions of the technical specifications and ignoring others, one can arrive at a position that the actual Browns Ferry reporting was acceptable.

However, if one considers all sections of the Browns Ferry technical specifications and interprets them conservatively, NSRS concludes that the containment leak that was found on December 9 falls under section 6.7.2.a and thus is subject to prompt reporting. Thus, the containment leak should have been reported to NRC by about 0300 on December 10 to be within the 24-hour limit.

Establishing and maintaining a drywell-to-torus pressure differential is required within 24 hours of reaching certain primary system pressure-temperature conditions. Primary system pressure-temperature conditions were reached about 12:15 p.m. on December 7, 1979. Thus, the differential pressure should have been established and maintained prior to 12:15 p.m. on December 8. The differential pressure was attained briefly then lost due to pressure decay. For technical specification purposes, NSRS does not consider that the pressure differential was established and maintained. Paragraph 3.7.A.6.b of the technical specifications allows operation to continue without pressure differential not to exceed 6 hours. Thus, NSRS interprets that pressure differential was not required until 1815 on December 8. However, orderly shutdown should have begun immeditely at this time. Shutdown was actually initiated at 0023 on December 9. This technical specification violation required reporting to NRC as a 30-day written report.

The NRC Resident Inspector was informed about the pressure differential and containment leakage problems about 0800 on December 10, 1979. At 1525 on

December 10, 1979, NRC was notified of the pressure differential problem and the containment leakage problem by telecopy. The report noted the pressure differential problem was later attributed to the containment leakage.

In summary, NSRS feels there were two reportable occurrences under the technical specifications, one on the containment leakage due as a prompt report to NRC within 24 hours of about 4 a.m., December 9, 1979, and a second report on exceeding 30 hours without establishing and maintaining drywell-to-torus pressure differential due as a 30-day report 30 days after 1800, December 8, 1979.

NSRS Recommendations

1. The Division of Nuclear Power (NUC PR) should prepare and implement written procedures with appropriate signoffs for installation and removal of all primary containment hatches at Browns Ferry.

2. The frequency of the local leak rate testing of the equipment hatches at Browns Ferry should be increased by NUC PR. In particular, a local leak rate test must be performed immediately following the completion of any integrated leak rate test.

3. Similar written procedures should be prepared for installation and removal of primary containment hatches at Sequoyah and subsequent nuclear plants.