From: Leonard Wiens, NRR
To: ATD1.ATP1.MSM Mark Miller, RII
Date: 5/24/96 7:31am
Subject: TS ISSUE

Mark: Attached is the response from Chris Grimes. Chris tends to be a little curt, but he indicates that although not explicitly contained in one statement, the position is covered in existing guidance. I will look into it a little more myself, if for no other reason than to get a little smarter in this area myself. (Like I said, I have been asked similar questions before without having a good answer. Its about time I did I guess)

Sorry. An old guy like me takes time to learn these new-fangled systems.

October 28, 1994

MEMORANDUM TO: Bruce A. Boger, Acting Director
Division of Reactor Projects, RII

FROM: John A. Zwolinski, Deputy Director (Original Signed By)
Division of Reactor Projects I/II, NRR

SUBJECT: REQUEST FOR ASSISTANCE IN ADDRESSING ISSUES REGARDING ST. LUCIE UNITS 1 AND 2 REFUELING PROCEDURES - TIA 94-023 - (TAC NOS. M90425 AND M90246)

We have completed the review of TIA 94-023 concerning two issues relating to St. Lucie's refueling procedures. The first issue concerns whether it is the intent of the licensee's technical specifications to have a licensed operator present as an observer during crane operation and fuel movement. The second concerns whether the Recommended Move List is part of the refueling procedure, and subject to the licensee's approval and change process.

The review was performed by the Human Factors Assessment Branch. The evaluation of these issues is attached.

Docket Nos. 50-335
50-389

Attachment: Safety Evaluation Report

cc w/attachment:
J. Norris
R. Cooper, RI
E. Greenman, RII
W. Beach, RIV

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

HUMAN FACTORS SAFETY EVALUATION REGARDING REFUELING PROCEDURES
FOR
ST. LUCIE UNITS 1 AND 2 (TAC Nos. M90425 and M90246)

1.0 INTRODUCTION

The Human Factors Assessment Branch has reviewed the memorandum from Bruce Boger to Gus Lainas, dated September 19, 1994. The memorandum addressed two specific issues of concern with St. Lucie's refueling procedures. The first issue is whether it is the intent of the licensee's technical specifications to have a licensed operator present as an observer during crane operation and fuel movement. The second is whether the Recommended Move List is part of the refueling procedure, and subject to the licensee's Technical Specifications (TS) requirement for review and approval of changes to procedures.

2.0 EVALUATION

First Issue: St. Lucie's Technical Specifications

In the September 19 memorandum, Region II requested NRR interpretation of St. Lucie's Technical Specification 6.2.2.d.

Technical Specification (TS) 6.2.2.d states:

> ALL CORE ALTERATIONS shall be observed by a licensed operator and supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation. The SRO in charge of fuel handling normally supervises from the control room and has the flexibility to directly supervise at either the refueling deck or the spent fuel pool.

10 CFR 50.54(m)(2)(iv) requires that "each licensee shall have present, during alteration of the core of a nuclear power unit (including fuel loading or transfer) a person holding a senior operator (SRO) license or senior operator license limited to fuel handling (LSRO) to directly supervise the activity and, during this time, the licensee shall not assign other duties to this person." The NRC interpretation of "directly supervise the activity" is that the SRO will supervise at the location of the activity of core alterations and fuel movement.

In contrast to this requirement, the licensee's technical specification allows for observation of core alterations and fuel movement by a Reactor Operator and supervision by an SRO or LSRO who may supervise from the control room. This position was confirmed in a letter from Region II to the licensee dated September 30, 1981.

NRR has no regulatory basis for interpreting the licensee's technical specifications regarding their requirement for a licensed operator observer. There is no requirement for such an observer in the regulations. However, because the regulations require direct supervision by an SRO or LSRO who has no other concurrent duties, NRR believes that the licensee should modify their technical specifications to bring them into compliance with the regulations. NRR recommends that the region along with NRR Projects request that the licensee modify their technical specification accordingly. Standard Technical Specification Section 5.2.2 provides an acceptable example. If the licensee does not choose to amend their technical specifications, we are prepared to support a compliance backfit.

Second Issue: Refueling List as Part of the Procedure

The second issue is whether the Recommended Move List is part of the refueling procedure, and subject to the licensee's TS requirement for review and approval of changes to procedures.

In previous cases dealing with this question, the NRC has determined that the fuel movement list is part of the refueling procedure and any changes to the movement list must go through the licensee's procedure change process.

SLSFP1.

MEMORANDUM TO: Joseph W. Shea, Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

FROM: Jan A. Norris, Sr. Project Manager
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

SUBJECT: ST. LUCIE UNITS 1 AND 2 - SPENT FUEL POOL SURVEY

This memorandum provides the information requested by the February 8, 1996, memorandum from John Stolz regarding a review of the spent fuel pool practices and current licensing basis.

Attachment: St. Lucie SFP Survey

Docket Nos. 50-335 and 50-389

EEB/24

SLSFP2.

ST. LUCIE SPENT FUEL POOL SURVEY

A. Spent Fuel Pool (SFP) System Design

UNIT 1

The system is composed of heat exchanger, filter, ion exchanger, pump suction strainer, ion exchanger strainer, pumps, piping and valves. The system has only one train of components. The cooling portion of the fuel pool system is a closed loop system consisting of two half capacity pumps and one full capacity heat exchanger. For normal refueling discharge conditions, one fuel pool pump and the fuel pool heat exchanger are in service. During abnormal refueling conditions, such as full core discharge, two fuel pool pumps and the heat exchanger are in service. The system is manually controlled from a local control panel. High fuel pool temperature, high and low fuel pool water level, and a low fuel pool pump discharge pressure alarms are announciated in the control room. Makeup to the fuel pool comes from the refueling water tank. The heat exchanger is cooled by component cooling water. The system is designed to provide a minimum of 9 feet of water above the top of the fuel during handling and storage operation.

UNIT 2

The system is composed of heat exchangers, filter, ion exchanger, pump suction strainer, ion exchanger strainer, pumps, piping and valves. The system has only one train of components. The cooling portion of the fuel pool system is a closed loop system consisting of two half capacity pumps and two full capacity heat exchangers. Full capacity condition corresponds to the design condition of a full core placed in the spent fuel pool seven days after reactor shutdown, in adtches, the most recent of which has been cooling for 90 days. For normal refueling discharge conditions, one fuel pool pump and the fuel pool heat exchanger are in service. During abnormal refueling conditions, such as full core discharge, two fuel pool pumps and the heat exchanger are in service. The system is manually controlled from a local control panel. High fuel pool temperature, high and low fuel pool water level, and a low fuel pool pump discharge pressure alarms are announciated in the control room. Makeup to the fuel pool comes from the refueling water tank. The heat exchanger is cooled by component cooling water. The system is designed to provide a minimum of 9 feet of water above the top of the fuel during handling and storage operation.

SLSFP3.

St. Lucie 1

B. SUMMARY OF CLB REQUIREMENTS RE:SPENT FUEL POOL DECAY HEAT REMOVAL/REFUELING OFFLOAD PRACTICES

1. Technical Specification limits are provided for:

TS 3.9.3: 72 hour minimum decay time.

TS 3.9.5: Direct communications between the control room and the refueling station during core alterations.

TS 3.9.6: Manipulator crane shall be used to move fuel assemblies and be operable.

TS 3.9.7: Crane travel with heavy loads (>2000 lbs.) over irradiated fuel is prohibited.

TS 3.9.11: Minimum water level 23 feet above the top of irradiated fuel in the SFP.

TS 3.9.12: At least one fuel pool ventilation system shall be operable.

TS 3.9.13: Maximum load for the spent fuel cask crane shall not exceed 25 tons.

TS 3.9.14: Decay fuel assemblies for 1180 hours (1490 hours for >one-third core) prior to movement of the spent fuel cask into the fuel cask compartment.

2. The fuel pool system is designed to provide shielding for irradiated fuel so that personnel dose rates do not exceed 2.5 mrem/hr; maintain pool temperature below 150 °F under offload conditions; maintain purity and clarity of the SFP, refueling cavity, and refueling water tank water; and maintain water level 9 feet above the irradiated fuel during transfer operations.

3. Design heat load for the normal batch discharge case assumes 18 batches of 80 assemblies discharge to the SFP in 18 month intervals, followed by a discharge of 80 assemblies after 150 hours of decay. With a single pump and heat exchanger in operation, the system can maintain SFP temperature below 134 °F. Time-to-boil assuming cooling was completely lost at the maximum temperature is 13.43 hours. A full capacity pump is available should the first pump fail. [FSAR Section 9.1.3.2] Normal discharge heatload is 16.42×10^6 Btu/hr. [Staff rerack safety evaluation dated 3/11/88]

Does the licensee consider the failure of the fuel pool heat exchanger credible. Having a single heat exchanger does not provide for single failure (this component is passive)

The FSAR describes the normal batch discharge case as a one-third offload. In March the licensee plans to offload a full core. Is it and abnormal offload or do they "normally" off load a full core. If so, change the FSAR.

4. Design heat load for the abnormal batch discharge case assumes 18 batches of 80 assemblies discharge to the SFP in 18 month intervals, followed by a discharge of 217 assemblies with 169 hour of decay. With both pumps and a single heat exchanger in operation, the system can maintain SFP temperature below 151 °F. Time-to-boil assuming cooling was completely lost at the maximum temperature is 5.04 hours. The capability to withstand a single failure criteria was not assumed [FSAR Section 9.1.3.2]. The heat load for the abnormal case is 33.70×10^6 Btu/hr. [Staff rerack safety evaluation dated 3/11/88]

The staff also accepted a single failure of the SFPCS pump with a full core in the SFP. The maximum temperature reached 167 °F under these assumptions. [Staff rerack safety evaluation dated 3/11/88]

The licensee's FSAR does not describe this scenario (full core offload - single failure). The licensee should be questioned whether a SFPCS single failure under full core offload heatload is part of their licensing basis.

5. The storage capacity for Unit 1 SFP is 1706 fuel assemblies (7 2/3 cores).

6. Boron concentration shall be maintained at a minimum of 1720 ppm.

7. The spent fuel poo' is designed to withstand the steady state water temperatures of 217 °F.

8. Makeup sources to the SFP are from: refueling water storage tank, city water storage tank, city water storage tanks via portable fire pump, and the primary water tank. A seismic Category I source is available from the intake cooling water inter-tie (salt) at 150 gpm [FSAR 9.1.3.4.5].

   Lining up seismic makeup using the refueling water tank is complicated. The PM should review the procedure to ensure it exists and that the licensee trains on it periodically.

9. No other implicit or explicit prohibitions exist within the CLB against performing a full core offload for any given refueling outage.

Discrepancies:

None. However, see comments in each section above.

St. Lucie 2

B. SUMMARY OF CLB REQUIREMENTS RE:SPENT FUEL POOL DECAY HEAT REMOVAL/REFUELING OFFLOAD PRACTICES

1. Technical Specification limits are provided for:

TS 3.9.3: 72 hour minimum decay time.

TS 3.9.5: Direct communications between the control room and the refueling station during core alterations.

TS 3.9.6: Manipulator crane shall be used to move fuel assemblies and be operable.

TS 3.9.7: Crane travel with heavy loads (>1600 lbs.) over irradiated fuel is prohibited.

TS 3.9.11: Minimum water level 23 feet above the top of irradiated fuel in the SFP.

TS 3.9.13: Maximum load for the spent fuel cask crane shall not exceed 100 tons.

TS 3.9.14: Decay fuel assemblies for 1180 hours (1490 hours for >one-third core) prior to movement of the spent fuel cask into the fuel cask compartment.

The PM should ask why there isn't a fuel building ventilation TS similar to Unit 1 TS 3.9.12.

2. The fuel pool system is designed to provide shielding for irradiated fuel so that personnel dose rates do not exceed 2.5 mrem/hr; maintain pool temperature below 150 °F under offload conditions; maintain purity and clarity of the SFP, refueling cavity, and refueling water tank water; and maintain water level 9 feet above the irradiated fuel during transfer operations.

3. Design heat load for the normal batch discharge case assumes 11 batches of 80 assemblies discharge to the SFP in 18 month intervals, followed by a discharge of 80 assemblies after 120 hours of decay. With a single pump and heat exchanger in operation, the system can maintain SFP temperature below 131 °F with a CCW temperature of 100 °F. Time-to-boil assuming cooling was completely lost at the maximum temperature is 12.6 hours. A full capacity pump is available should the first pump fail. [FSAR Section 9.1.3.1] Normal discharge heatload is 16.42×10^6 Btu/hr.

4. Design heat load for the full core discharge case assumes 11 batches of 80 assemblies discharge to the SFP in 18 month intervals (the most recent has decayed 90 days), followed by a discharge of 217 assemblies with 168 hours of decay. With both pumps and a single heat exchanger in operation, the system can maintain SFP temperature below 148 °F with a CCW temperature of 100 °F. Time-to-boil assuming cooling was completely lost at the maximum temperature is 3.9 hours. The capability to withstand a single failure criteria was not assumed. The heat load for the abnormal case is 30.3×10^6 Btu/hr [FSAR Section 9.1.3.1]. A single failure was analyzed for this heat load case. The maximum pool temperature under full core offload heatload was found to be less than 160 °F [FSAR 9.1.3.3].

5. Piping and components in the SFPCS are Quality Group C, seismic Category I, designed for a temperature of at least 200 °F. [FSAR Section 9.1.3.2.1, and Table 9.1-6.]

6. Normal makeup sources to the SFP are from the refueling water storage tank and the primary water tank. Three million gallons of makeup are also available from the city water storage tanks, condensate storage tank, demineralized water tank, and others. A seismic Category I source is also available through hoses from the intake cooling water header. [FSAR Section 9.1.3.3.1]

7. No implicit or explicit prohibitions exist within the CLB against performing a full core offload for any given refueling outage.

Discrepancies:

None

August 6, 1996

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

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

SUBJECT: TECHNICAL ASSISTANCE REQUEST (TIA 95-013) IN ADDRESSING ISSUES RELATING TO THE ADEQUACY OF A 50.59 EVALUATION AT ST. LUCIE UNIT 2 (TAC NO. M93372)

In a memorandum dated August 28, 1995, NRR assistance was requested in evaluating the acceptability of a 50.59 evaluation supporting isolation of a diesel generator fuel oil transfer system leak at St. Lucie Unit 2. In addition, several generic questions concerning the relationship between Probabilistic Risk Assessment (PRA) evaluations and 10 CFR 50.59 requirements were presented for NRR response.

The Probabilistic Safety Assessment Branch, NRR, has completed its review of these issues. A discussion of these issues and NRR's response to your questions is contained in the attached memorandum dated July 30, 1996. The positions stated in the attachment have been reviewed by the Office of the General Counsel and they have no legal objection to these positions.

Docket No.: 50-389

Attachment: As Stated

cc w/attachment: R. Cooper, RI
W. Axelson, RIII
J. Dyer, RIV

'96 AUG 12 A11:26

Contact: L. Wiens, NRR\PDII-3
415-1495

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July 30, 1996

MEMORANDUM TO: Frederick Hebdon, Director
Project Directorate II-3
Division of Reactor Projects I/II

FROM: Edward J. Butcher, Chief
Probabilistic Safety Assessment Branch
Division of Systems Safety and Analysis

SUBJECT: RESPONSE TO REQUEST FOR ASSISTANCE IN ADDRESSING ISSUES REGARDING ST. LUCIE EMERGENCY DIESEL GENERATOR FUEL OIL TRANSFER SYSTEM LEAK ISOLATION AND USING OPERATOR ACTION IN PLACE OF AUTOMATIC ACTION (TIA 95-013)

Plant Name: St. Lucie Unit 2
Utility: Florida Power & Light Co.
Docket No.: 50-389
TAC No.: M93372
Project Manager: Leonard A. Wiens
Review Branch: SPSB
Review Status: Complete

The attachment to this memorandum is our response to TIA 95-013. It contains our responses to the specific questions raised by Region II regarding the 10 CFR 50.59 FPL Safety Evaluation (JPN-PSL-SENS-95-013), and the application of PRA methodology and related issues. If you have any questions regarding our response to the TIA request or regarding the licensee's PRA assessment which was included in the TIA, please contact John Schiffgens at 415-1074 (E-mail: JOS), or John Flack at 415-1094 (E-mail JHF). In addition, we are in the process of developing a formal position on the use of PRA in the 10 CFR 50.59 process which will be sent to you in a separate memorandum.

Attachment: As stated

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

July 30, 1996

MEMORANDUM TO: Frederick Hebdon, Director
Project Directorate II-3
Division of Reactor Projects I/II

FROM: Edward J. Butcher, Chief
Probabilistic Safety Assessment Branch
Division of Systems Safety and Analysis

SUBJECT: RESPONSE TO REQUEST FOR ASSISTANCE IN ADDRESSING ISSUES REGARDING ST. LUCIE EMERGENCY DIESEL GENERATOR FUEL OIL TRANSFER SYSTEM LEAK ISOLATION AND USING OPERATOR ACTION IN PLACE OF AUTOMATIC ACTION (TIA 95-013)

Plant Name:	St. Lucie Unit 2
Utility:	Florida Power & Light Co.
Docket No.:	50-389
TAC No.:	M93372
Project Manager:	Leonard A. Wiens
Review Branch:	SPSB
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Attachment: As stated

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RESPONSES TO SPECIFIC TIA 95-013 QUESTIONS

1. **Is the attached 10 CFR 50.59 FPL Safety Evaluation (JPN-PSL-SENS-95-013) considered acceptable?**

No. The attached 10 CFR 50.59 FPL Safety Evaluation is not considered acceptable.

The 50.59 evaluation prepared by FPL for St. Lucie is not acceptable because it involves an unreviewed safety question. An unreviewed safety question exists because the proposed change introduces a new procedure and associated malfunction of a different type (operator error) and involves an increased probability of the malfunction of equipment important to safety (mechanical valve failure to open). Specifically, the 2B EDG fuel oil isolation valve, which is a manual valve, is normally in a LOCKED OPEN position and requires no change-of-state for EDG operation. The proposed change involves operating with this valve in the closed position and opening it manually as needed. With the valve in the closed position, two new failure modes exist for the fuel oil supply system: failure of the operator to open the fuel oil manual isolation valve, and mechanical failure of the valve to open. One failure mode results in a malfunction of a different type, introducing operator error where no operator action was required before. The other increases the probability of malfunction of the valve, since the probability of failure to open is greater than zero, where it was zero before. Both increase the probability of malfunction of the 2BEDG.

In the evaluation, JPN-PSL-SENS-95-013, Rev. 0, page 8, FPL improperly answered the question "Does the proposed activity increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR?" by stating that "the compensatory actions assure the reliability of the EDG fuel oil supply." In general, the introduction of compensatory measures suggests that there is an unreviewed safety question for which compensation is needed, hence, a 50.90 submittal should be prepared by the licensee and evaluated by the staff to determine whether the compensation is adequate. Frequently, however, licensees refer to risk reducing features that are an integral part of the change as compensatory measures. For example, introducing operator instructions for a newly instituted manual operation should not be considered a compensatory action nor should new administrative controls intended to assure sufficient time to perform the action. Although NRC Inspection Manual, "Part 9900: 10 CFR Guidance," provides some limited guidance on compensatory actions, the staff is in the process of better defining what constitutes appropriate use of compensatory measures in 10 CFR 50.59 safety evaluations. In this case, the change consists of the licensee introducing a procedure and operating restrictions to replace an automatic "supply on demand" condition. The "compensatory" actions were intended to make the procedure as reliable as the original configuration, however, they did not.

The licensee also stated that "the failure of the EDG fuel oil isolation valve is possible" and provided a quantitative assessment. Its 10 CFR 50.59 evaluation quantified the change in frequency of the loss of the B side electrical bus in conjunction with a LOOP initiating event. The result obtained was a 6% increase in the frequency per year of the loss of the 2B3 4.16kV bus in conjunction with a LOOP. The report does not provide sufficient detail on model and assumptions to evaluate the licensee's analysis. It should be noted that for an appropriate analysis (i.e., one intended to demonstrate compliance with the provisions of 10 CFR 50.59), the change should have been assessed in terms of the probability of malfunction of equipment <u>or</u> the probability of occurrence or the consequences of an accident. In this case, the licensee should have explicitly evaluated the probability of malfunction of the B EDG.

2. **From a PRA perspective, is it possible to completely mitigate a risk, once introduced?**

   Yes. Not only can an introduced risk be mitigated, i.e., reduced, it can have a positive safety impact, i.e., the risk can be made lower than it was originally. It is often a matter of economic balance; how much will it cost to reduce the risk. It is frequently possible to put in place equipment, change equipment configurations, and/or change procedures so as to effectively and satisfactorily mitigate risk (e.g., to mitigate the <u>increase</u> in risk associated with an increase in the probability of equipment malfunction or accident initiation) in a cost effective manner when the introduced risk is fully understood. This is significant with regard to 50.90 submittals. 10 CFR 50.59 evaluations are concerned with identifying unreviewed safety questions, i.e., with deciding whether a proposed change (a) <u>may increase</u> the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated, (b) <u>may create</u> a possibility for an accident or malfunction of a different type than any evaluated previously, or (c) <u>reduces</u> a margin of safety as defined in the basis for any technical specification. That is, 10 CFR 50.59 <u>is concerned with whether there may be</u> a decrease in safety <u>not with how "large" it may be</u>.

3. **Is the licensee's position (that the risk of operator failure/error can be mitigated, probabilistically, through procedures and training) valid?**

   Yes. Operator failure probabilities can be reduced through the use of improved procedures, proper training, increased knowledge, etc. However, it should be noted that although the probability of human error can be reduced or mitigated through procedures and training, it cannot be reduced to zero.

   It should also be noted that for a given task, hardware is usually more reliable than human action, and the uncertainty associated with quantifying human action is usually greater than the uncertainty associated with hardware reliability. For example, if an automatic

hardware action is substituted with a human action, the point estimates of the failure probabilities for the human performance are generally greater than for the hardware performance and there is usually more uncertainty associated with the human error probability than there is with the hardware failure probability.

Do probabilistic estimations of operator error rates presuppose the existence of procedures and training and if so, can one then take credit for them in a deterministic mitigation of risk?

In human reliability analysis (HRA), performance shaping factors (PSFs) modify human error probabilities by accounting for the impact of various factors on operator actions. PSFs include procedures and training among other factors such as stress, environmental conditions, etc. However, analysts will sometimes use "screening" values for human error rates.

These screening values are usually bounding "guesses" and may not include performance shaping factor aspects.

Whether or not credit can be taken for the existence of procedures and training in a "deterministic mitigation of risk," is outside the purview of PRA. If by "deterministic mitigation of risk" is meant "evaluation of the mitigation of risk using techniques other than probabilistic," one should be able to take credit for procedures and training in assigning an "effectiveness measure" to operator actions. The difficulty would be in devising a "measure" and applying it systematically in a deterministic framework.

4. **Can 10 CFR 50.59 requirements (that the probability of failure of components important to safety not be increased if no unreviewed safety question is deemed to exist) be satisfied if new failure mechanisms are added to a previously reviewed system?**

   A proposed change, test, or experiment (CTE) can not be made under the provisions of 50.59 if it involves an unreviewed safety question. The stated change, resulting in the introduction of a new failure mechanism (e.g., replacing a manual valve with an MOV), would involve an unreviewed safety question, because it may result in a malfunction (of equipment important to safety) of a different type than any evaluated previously in the safety analysis report. In addition, a change which introduces a new failure mechanism, may increase the probability of malfunction of equipment (e.g., a train or system) important to safety previously evaluated, and thereby also constitute an unreviewed safety question.

5. **PRA insights are beginning to provide a more structured evaluation process for proposed changes to facilities and, as a result, are showing that changes (in a 10 CFR 50.59 context) present finite, although small, increases in the probabilities of failures. Is there a threshold value**

of increased probability (representing "negligible" or "insignificant" increases) below which 10 CFR 50.59 criteria (for demonstrating that unreviewed safety questions do not exist) are satisfied?

No. According to the rule, the proposed CTE must not, nor have a credible potential to, result in a finite increase in the probability of failure in order for it to be implemented under the provisions of 10 CFR 50.59.

6. The response to a related TIA from Region II, transmitted via letter from you to Edward Greenman dated June 23, 1993, stated in part that "NRR has no particular objection to the use of PRA in 10 CFR 50.59 evaluations but recommends that it play a supportive role in conjunction with other inputs, such as engineering judgement and operating experience." In the given case at St. Lucie, when PRA insights provide information counter to (as opposed to supportive to) the 10 CFR 50.59 conclusions, is it appropriate to accept deterministic conclusions over the PRA-indicated increase in probabilities of failure?

In general, when there are differences in conclusions based on deterministic considerations compared to those based on probabilistic considerations, the solution is <u>not</u> to simply accept one over the other but to determine the reason for the differences. Fundamentally, such analyses, if done properly, should complement each other, the latter being an extension of the former. In this regard, it should be kept in mind that engineering judgement about components and systems is incorporated in PRA models, as is operating experience and associated data. The inputs to different assessments need to be consistent if the outputs or results are to be consistent. Frequently, differences can be reconciled by identifying and evaluating assumptions incorporated in the assessments.

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

July 12, 1996

MEMORANDUM FOR: Stewart D. Ebneter, Regional Administrator, Region II

FROM: William T. Russell, Director
Office of Nuclear Reactor Regulation

Frank J. Miraglia

SUBJECT: INSTITUTING N+1 AT ST. LUCIE

This refers to your memorandum dated June 28, 1996, in which you requested my concurrence in returning resident inspector staffing at St. Lucie to N+1. I have reviewed your request and agree that based on plant performance, the exemption to N+1 staffing at St. Lucie that was approved in my memorandum dated October 2, 1995, should be rescinded.

cc: J. Taylor, EDO
T. Martin, RI
H. Miller, RIII
L. Callan, RIV
V. McCree, EDO
F. Hebdon, NRR

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~~9607160250 XA~~

FACSIMILE TRANSMITTAL
U. S. NUCLEAR REGULATORY COMMISSION
REGION II
ATLANTA, GEORGIA

ENFORCEMENT AND INVESTIGATION COORDINATION STAFF

TO: OEMAIL / JEB, MAS

SUBJECT: T22 ENF PANEL 07/16/96

☒ OE (301+415-3431)   ☐ WF (301+415-2260)

☐ OTHER ______ DATE: JUL 15 1996

St. Lucie EAW material

NO. OF PAGES 13 + TRANSMITTAL SHEET

FROM: BRUNO URYC, DIRECTOR, EICS
OFFICE: (404) 331-5505
FAX: (404) 331-0426
INTERNET: BXU@NRC.GOV

SENT BY: ______ TIME: 1201

EEE/127

STAR 951048

Bladow

ST. LUCIE ACTION REPORT
(STAR)

IDENTIFICATION SECTION

Date 9/5/95

STAR # 951048

Person/Department Initiating J. VOORHEES/QA

INITIAL Q.C. REVIEW
9/6/95

ORIGINATOR

Description

AP 1/2-0010125A DATA SHEETS 8A & 8B REQUIRE THAT A LUBRICATION PM BE PERFORMED PRIOR TO CONDUCT OF THE STROKE TIME TEST FOR FCVs 07-1A/1B. THIS IS A QUESTIONABLE PRACTICE IN THAT IT ALTERS THE 'AS-FOUND' CONDITION OF THE VALVE & MAY NOT REFLECT THE TRUE ABILITY OF THESE VALVES TO PASS THE STROKE TIME TEST.

Unit 1/2

System

Component ID FCV 07-1A/1B

Individual Notified P. FULFORD

Location

Operator workaround ☐ Yes ☑ No

References (ie, NRC Correspondence #, Audit Report, Drawing #, personnel observation, etc.)

Actions

ASME XI

A01
007
B13

1. Were any steps taken to mitigate? ☐ Yes ☐ No
2. What were they and were they successful?
3. Suspected cause of condition if known.
4. Recommendation to correct and department responsible. P. FULFORD - OPS / MMEG

Department Head Signature [signature] Bladow Date 9/5/95 Do you require approval to close? ☒ Yes ☐ No

REVIEW/APPROVAL

PGM

1. Assigned Department. ~~TECH Croper~~ Ops/ ~~Fulford~~ Sandy, F.
2. Reviews STA ☐ ANII ☒ ISI ☒ Operability Assessment Required: JPN ☐ OPS ☐
3. NP-700 ☐ HPES ☐ IHE ☐ Technical Subcommittee Review Required ☐
4. Evaluation due by 9/11/95 3 DAYS Extended to: __/__/__ Initials: ____
5. Corrective measures completed by __/__/__. Extended to: __/__/__ Initials: ____
6. Is item a mode hold? ☐ Yes ☑ No 1 ☐ 2 ☐ 3 ☐ 3/1750 ☐ 4 ☐ 5 ☐

Comments: OPS VERIFY TC/PCR IN PLACE TO CORRECT THIS PRACTICE. PUSTOVER HAS ADVISED THAT THIS WAS DONE.

Signature [signature] Date 9/6/95 Do you require approval to close? ☑ Yes ☐ No

STA

1. Notifications. (i.e., AP 0010721, "NRC Required Non Routine Notifications & Reports" AP 0005782, "Plant Guide to Reporting Environmental Non-Compliances and Significant Events" SP 0006125, "Reporting of Safeguards Events") ☐ Yes ☐ No
2. Event Type 1-9 ____
3. Security event ☐ Yes ☐ No ☐ FOP

Signature / Date

ANALYSIS/WORK

STAR # 951048

S___OPS	
DATE	
DOCT	
DOCN	
SYS	
COMP	
ITM	

ASSIGNED DEPARTMENT

Review for Technical Assistance

Engineering

- [ ] A. Acceptability/operability of a non conforming item is in question and operation is required.
- [ ] B. Engineering guidance is required or restoration of design.
- [ ] C. Engineering assistance needed for root cause determination.
- [ ] D. For safety related/quality related items, past operability of item is questionable (as found condition).
- [ ] E. EQ evaluation is needed.
- [ ] F. 10CFR21 issue

Technical

- [ ] G. 50.59 issue
- [ ] H. ASME XI issue

If any block (A through H) is checked, answer the following.

Resolution required. Date __/__/__ Mode ____

Conditional release to use as is [ ] Yes [ ] No (supporting documentation required)

Dispositioned by ______ Verified by ______

Investigation/Root Cause/Generic Impact/How to Prevent/Response Responsible Person FULFORD

TC- 1-95-255 WAS WRITTEN AND INCORPORATED INTO 1-0010125A ON 9/2/95. PROCEDURE CHANGE REQUESTS (ATTACHED) HAVE BEEN SUBMITTED AND WILL BE INCORPORATED IN THE NEXT REVISION.

Corrective Actions	Assigned to	Expected Date	Outage Required SNO	Outage Required REF	Vehicle used (PCM, WO number)	Completed Date
1. PCR	Sandy	/ /	[ ]	[ ]	PRC # 95-366	11/1/95
2.		/ /	[ ]	[ ]		/ /
3.		/ /	[ ]	[ ]		/ /
4.		/ /	[ ]	[ ]		/ /

CLOSEOUT SECTION

		Signature	Date	
1.	All corrective actions complete (note exceptions below)	[signature]	11/1/95	Assigned Department
2.	Initiating Department Concurrence	[signature]	11/1/95	Initiating Department
3.	ANII Review	N/A RLM		
4.	ISI Review	N/A RLM		
5.	QC Concurrence	N/A [signature]		QC
6.	Plant General Manager Approval (corrective actions satisfactory and exceptions agreed to)	J. Scarola	11/6/95	Plant General Manager

Exceptions:

FIGURE 1
PROCEDURE CHANGE/REVIEW REQUEST
(Sheet 1 of 2)

PROCEDURE TITLE Surveillance Data Sheets

PROCEDURE NUMBER 1-0010125A Present Rev. No. 39

1. Reason for Request

☐ 1.1 Response to INPO, QA, or NRC Requirements
☐ 1.2 Cycle Specific/documented Setpoint Revision.
☐ 1.3 Tech Spec. Rev/Licensing Requirement (List Tech. Spec. Amendment No. below)
☐ 1.4 Affects Plant Operability/Availability
☒ 1.5 PC/M or STAR /R62
☐ 1.6 Procedural Improvement /R62
☐ 1.7 Tech. Manual Requirement or Vendor Revision /R62
☐ 1.8 Technically incorrect /R62
☐ 1.9 Other /R62

Include description and source (i.e., PC/M #, STAR #). If Other give detailed explanation: Valve should not be preconditioned (lubed) prior to surveillance. STAR 1-950869

NOTE

Attach a copy of the affected pages of the present revision. Changes should be legible in **RED** ink directly on the affected page. If extensive additions are required use additional sheets as necessary; clearly indicate the proper placement on the appropriate page of the old revision. Highlighters, correction fluid or any other obliteration material should not be used.

2. a. Is this change to a unit specific procedure? X Yes __No - common procedure
   b. If yes, has a PCR for the opposite unit been submitted? X Yes __No
      If no, explain______
   c. Does this PCR incorporate a T.C.? X Yes __No T.C. # (if applicable) 1-95-255
   d. Does this PCR reference any chemical? __Yes X No
      If yes, Chemical Control Supv. Review______

3. Periodic Review (Check if applicable)
   Check below only if performing a review where no changes are necessary.
   ☐ Review performed, no changes (FRG not required).

4. Requested by:
   (print name): Jon Halleran Phone 3086
   (Signature): Jon Halleran Date: 9/2/95
   Subcommitteed by: [signature] Date: 9/2/95
   Dept. Head: ______ Date: __/__/__
   Priority: (determined by Dept. Head) ☐1 ☐2 ☐3 ☐4
   Required by___/___/___ (if applicable)

S__OPS

DATE______
DOCT______
DOCN______
SYS______
COMP______
ITM______

ST. LU . UNIT 1
ADMINISTRATIVE PROCEDURE NO. 1-0010125A, REVISION 39
SURVEILLANCE DATA SHEETS

DATA SHEET #8A
VALVE CYCLE TEST - NON-CHECK VALVES
(Page 5 of 9)

Note: S in required stroke time column indicates ESFAS valve.

Valve No.	Description	Satisfactorily Exercise Valve (Initials)	Stroke Time/Direction: Enter Actual Time	Stroke Time/Direction: Maximum Required Time	Stroke Time/Direction: O/C	Fail Position: Enter Actual Position	Fail Position: Fail Required Position	System Restored I.V. (Initials)	Valve Position Required After Test	Test Method	Remarks
FSE-27-8*	Sample Into A H_2 Analyzer			2	O	N/A	N/A		Closed	B	* Fast Acting Valve
				2	C		FC				
FSE-27-9*	Sample Into B H_2 Analyzer			2	O	N/A	N/A		Closed	B	
				2	C		FC				
FSE-27-10*	Sample from B H_2 Analyzer			2	O	N/A	N/A		Closed	B	
				2	C		FC				
FSE-27-11*	Sample from A H_2 Analyzer			2	O	N/A	N/A		Closed	B	
				2	C		FC				
FCV-03-1E*	SIT Sample Line Isolation			2	C		FC		Closed	B	
FCV-03-1F*	SIT Sample Line Isolation			2	C		FC		Closed	B	
FCV-07-1A	Containment Spray Hdr A			S 8	O		FO		Closed	B	Ensure lubrication PM performed prior to stroking -
MV-07-3A	A Containment Spray Isol			120	C	N/A	N/A		Locked Open	N/A	
SE-07-1A	Caustic Injection			2 S	O	N/A	N/A		Closed	B	* Fast Acting Valve
				2	C		FC				
V-07145	1A CS Pump Disch. Isol.		N/A	N/A	C	N/A	N/A		Locked Open	N/A	Tested by OP 1-0420050

Did NOT EXIST ON PREVIOUS REV

FIGURE 1
PROCEDURE CHANGE/REVIEW REQUEST
(Sheet 1 of 2)

PROCEDURE TITLE Surveillance Data Sheets

PROCEDURE NUMBER 2-0010125A Present Rev. No. 43

1. Reason for Request
- ☐ 1.1 Response to INPO, QA, or NRC Requirements
- ☐ 1.2 Cycle Specific/documented Setpoint Revision.
- ☐ 1.3 Tech Spec. Rev/Licensing Requirement (List Tech. Spec. Amendment No. below)
- ☐ 1.4 Affects Plant Operability/Availability
- ☒ 1.5 PC/M or STAR /R62
- ☐ 1.6 Procedural Improvement /R62
- ☐ 1.7 Tech. Manual Requirement or Vendor Revision /R62
- ☐ 1.8 Technically incorrect /R62
- ☐ 1.9 Other /R62

Include description and source (i.e., PC/M #, STAR #), if Other give detailed explanation: Valve should not be preconditioned (lubed) prior to surveillance. STAR 1-950869

> **NOTE**
> Attach a copy of the affected pages of the present revision. Changes should be legible in RED ink directly on the affected page. If extensive additions are required use additional sheets as necessary; clearly indicate the proper placement on the appropriate page of the old revision. Highlighters, correction fluid or any other obliteration material should not be used.

2. a. Is this change to a unit specific procedure? X Yes __No - common procedure
   b. If yes, has a PCR for the opposite unit been submitted? X Yes __No
   If no, explain
   c. Does this PCR incorporate a T.C.? X Yes __No T.C. # (if applicable) 2-95-177
   d. Does this PCR reference any chemical? __Yes X No
   If yes, Chemical Control Supv. Review

3. Periodic Review (Check if applicable)
   Check below only if performing a review where no changes are necessary.
   ☐ Review performed, no changes (FRG not required).

4. Requested by:
   (print name): Jon Halleen Phone 3086
   (Signature): Jon Halleen Date: 9/2/95
   Subcommitteed by: [signature] Date: 9/2/95
   Dept. Head: Date: __/__/__
   Priority: (determined by Dept. Head) ☐ 1 ☐ 2 ☐ 3 ☐ 4
   Required by___/___/___ (if applicable)

S__OPS
DATE
DOCT
DOCN
SYS
COMP
ITM

STAR 951063

Dean

ST. LUCIE ACTION REPORT
(STAR)

IDENTIFICATION SECTION

Date 9/8/95

STAR # 950[illegible] P.M. 63

Person/Department Initiating FULFORD / OPS

INITIAL Q.C. REVIEW [illegible] 9/8/95

Description

DURING THE RECENT UNIT 1 SNO, IT WAS DETERMINED THAT THE TIMING OF OUR PM PROGRAM ON FCV 07-1A & 1B AND OUR SURVEILLANCE TESTING MIGHT HAVE MASKED AN EQUIPMENT RELIABILITY PROBLEM. COULD THERE BE OTHER COMPONENTS OR EQUIPMENT IN A SIMILAR SITUATION?

Unit BOTH

System

Component ID

Individual Notified

Location

Operator workaround ☑ Yes ☐ No

ORIGINATOR

References (ie, NRC Correspondence #, Audit Report, Drawing #, personnel observation, etc.)

D06
999
B01

Actions

1. Were any steps taken to mitigate? ☐ Yes ☐ No
2. What were they and were they successful?
3. Suspected cause of condition if known.
4. Recommendation to correct and department responsible. EVALUATE OUR PM / SURVEILLANCE PROGRAMS TO ASSESS THE EXTENT OF THIS PROBLEM.

Department Head Signature [signature] Date 8 SEP, 95 Do you require approval to close? ☑ Yes ☐ No

REVIEW/APPROVAL

PGM

1. Assigned Department. [illegible] / [illegible] ~~SCE/SNYDER~~ OK 9/11/95 SCE/DEAN
2. Reviews STA ☐ ANII ☐ ISI ☐ Operability Assessment Required: JPN ☐ OPS ☐
3. NP-700 ☐ HPES ☐ IHE ☐ Technical Subcommittee Review Required ☐
4. Evaluation due by 10/11/95 (30 days) Extended to: / / Initials:
5. Corrective measures completed by 12/1/95 Extended to: / / Initials:
6. Is item a mode hold? ☒ Yes ☐ No — 1 ☐ 2 ☒ 3 ☐ 3/1750 ☐ 4 ☐ 5 ☐

Comments: NEED TO COMPLETE REVIEW PRIOR TO U-2 START-UP

Signature [signature] Date 9/11/95 Do you require approval to close? ☑ Yes ☐ No

STA

1. Notifications. (i.e., AP 0010721, "NRC Required Non Routine Notifications & Reports" AP 0005782, "Plant Guide to Reporting Environmental Non-Compliances and Significant Events" SP 0006125, "Reporting of Safeguards Events") ☐ Yes ☐ No
2. Event Type 1-8 ________
3. Security event ☐ Yes ☐ No ☐ FOP

Signature / Date

(QI-16-BB.WPQ) (Rev. 1 [illegible])

Inter-Office Correspondence

FPL

To: Paul Pulford

From: Jon Hallem

Jon Hallem

Date: October 5, 1995

Department: Ops Test

Subject: **STAR 951063 Response**

I have reviewed the test and surveillance procedures pertaining to the inservice testing of pumps and valves for both Units 1 and 2. I found no further instances where preventive maintenance is specified by the procedure to be performed prior to the test or surveillance.

The plant does routinely perform PMs prior to scheduled quarterly surveillances. This is done as a matter of conveniences since a surveillance run is required following the PM. By scheduling the PM work to be performed prior to the scheduled surveillance the number of surveillances performed is reduced. This is desirable because this limits the number of demands placed on the pumps, the amount of operator manpower needed to support the tests, and reduces the unavailabilty for pumps which are taken out of service to perform the surveillance. Attached is a listing of the PMs performed prior to scheduled surveillances. These PMs are in two major categories; pumps and fans, and valves and AFW Terry Turbine. I have reviewed these PMs with the applicable SCE and Predictive Maintenance Engineers and have collectively come to the following two conclusions.

For the first category, the PMs are for pump oil change out, coupling lubrication and fan lubrication. These are considered to have a minor, if any, impact on the performance of the pumps and fans during the surveillances. These PMs are performed less frequently than the quarterly surveillances with no indication that the performance or lack of the activities has influenced any surveillance. Also, some of the components are run during normal plant operations, and problems with Charging pump accumulator pressure and other pumps and fans are detected by Operations during normal observation of the components. Therefore this first category of PMs should continue to be performed prior to scheduled surveillances.

The PMs performed on FCV-07-1A, FCV-07-1B, and the AFW Terry Turbine, and Governor valves may have an impact on the performance of these components. The tests performed on the Unit 2 MFIVs have no effect of the safety related function of the MFIV surveillance. This category of PMs with the exception of the MFIVs should be rescheduled from just prior to the performance of the surveillances associated with these components.

AP-1-0010125A

ST. LU UNIT 1
ADMINISTRATIVE PROCEDURE NO. 1-0010125A, REVISION 39
SURVEILLANCE DATA SHEETS

DATA SHEET #8B
VALVE CYCLE TEST - NON-CHECK VALVES
(Page 5 of 10)

Note: S in required stroke time column indicates ESFAS valve.

Valve No.	Description	Satisfactorily Exercise Valve (Initials)	Stroke Time/Direction: Enter Actual Time	Maximum Required Time	O/C	Fail Position: Enter Actual Position	Fail Required Position	System Restored I.V. (Initials)	Valve Position Required After Test	Test Method	Remarks
V-03920	SIT Outlet Drain to VCT		N/A	N/A	O	N/A	N/A		Closed	N/A	
I SE-02-01	Loop 1B1 Charging Isol			5	C	N/A	N/A		Open	B	Conduct test of V-2433 (Data Sheet #9B) at same time.
				5	O		FO				
I SE-02-02	Loop 1A2 Charging Isol			5	C	N/A	N/A		Open	B	Conduct test of V-2432 (Data Sheet #9B) at same time.
				5	O		FO				
V-2504	RWT to Charging Pumps			12	O	N/A	N/A		Closed	*	* Ensure VCT pressure is > 25 psig prior to cycling.
				12	C						
V-2507	RCP Bleedoff to Quench Tank			5	C	N/A	N/A		Open	N/A	
V-6554	Waste Gas Cntmt Isol			5 S	C		FC		Open	B	
V-6555	Waste Gas Cntmt Isol			5 S	C		FC		Open	B	
V-6741*	Nitrogen Hdr Containment Isol			2 S	C		FC		Closed	B	* Fast Acting Valve
FCV-07-1B	Containment Spray Hdr B			S 8	O		FO		Closed	B	Ensure lubrication PM performed prior to stroking.
MV-07-3B	B Containment Spray Isol			120	C	N/A	N/A		Locked Open	N/A	
SE-07-1B	Caustic Injection			2 S	O	N/A	N/A		Closed	B	* Fast Acting Valve
				2	C		FC				
V-07130	1B CS Pump Disch Isol		N/A	N/A	C	N/A	N/A		Locked Open	N/A	Tested by OP 1-0420050

Note Did NOT En. ON PREVIOU REV

AP-2-0010125A

ST. LUCIE UNIT 2
ADMINISTRATIVE PROCEDURE NO. 2-0010125A, REVISION 43
SURVEILLANCE DATA SHEETS

DATA SHEET #6B
VALVE CYCLE TEST - NON-CHECK VALVES
(Page 11 of 13)

Note: S in required stroke time column indicates ESFAS valve.

Valve No.	System Description	Satisfactorily Exercise Valve (Initials)	Stroke Time/Direction: Enter Actual Time	Stroke Time/Direction: Maximum Required Time	Stroke Time/Direction: O/C	Fail Position: Enter Actual Position	Fail Position: Fail Required Position	System Restored I.V. (initials)	Valve Position Required After Test	Test Method	Remarks (INITIAL)	
SE-07-3B	2B Hydrazine Pump Disch			S 5	O		FO		Closed	B	1. Have I&C lift Lead #2 on TB 649 in cabinet ESC-SB. ____ 2. Cycle SE 07-3B. ____ 3. Have I&C re-land Lead #2 on TB 649 in Cabinet ESC-SB. ____	/R43
				5	C	N/A	N/A					
FCV-07-1B	B CS Hdr Flow Control			S 8	O		FO		Closed	B	Ensure lubrication PM is complete prior to stroking FCV 07-1B.	/R43
MV-07-4	B Containment Spray Isol			75	C	N/A	N/A		Locked Open	N/A		
FCV-26-1	Containment Sample Isol			S 5	C		FC		Open	B		
FCV-26-2	Containment Sample Isol			S 5	C		FC		Open	B		
FCV-26-3	Containment Sample Isol			S 5	C		FC		Open	B		
FCV-26-4	Containment Sample Isol			S 5	C		FC		Open	B		
FCV-26-5	Containment Sample Isol			S 5	C		FC		Open	B		
FCV-26-6	Containment Sample Isol			S 5	C		FC		Open	B		

[illegible]

Did NOT EXIST ON PREVIOU REV

ORA :404-331-6471

** Transmit Conf.Report **

Jul 15 '62 12:08

ORA ---> 8-301-415-3431	
No.	0003
Mode	NORMAL
Time	4'30"
Pages	14 Page(s)
Result	O K

Janet. Orig → TIA
CC K. Landis
DRP BC's
C. Julian
J. Jaudon
A. Gibson

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

'96 JUL 10 A11:46

July 2, 1996

MEMORANDUM TO: Jon R. Johnson, Acting Director
Division of Reactor Projects, RII

7/10

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

SUBJECT: TECHNICAL ASSISTANCE REQUEST (TIA 96-007) REGULATORY ACCEPTABILITY OF LUBRICATING VALVES PRIOR TO SURVEILLANCE TESTING (TAC NOS. M95274 AND M95275)

In a memorandum dated April 12, 1996, as a result of valve stroke timing practices at the St. Lucie Plants, you requested NRR assistance in evaluating the acceptability of lubricating valves prior to the performance of stroke time testing. You also asked NRR to resolve a question as to whether the purpose of the stroke time testing was to demonstrate current and past operability of a valve, current and future operability of a valve, or both.

The Mechanical Engineering Branch (EMEB), NRR, has completed its review of these issues. A discussion of these issues and NRR's response to your questions is contained in the attached memorandum dated June 24, 1996.

Docket Nos.: 50-335 and 50-389

Attachment: As Stated

cc w/attachment: R. Cooper, RI
W. Axelson, RIII
J. Dyer, RIV

Contact: L. Wiens, NRR\PDII-3
415-1495

EEE/128

9607150019 1P

June 24, 1996

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

FROM: Richard H. Wessman, Chief
Mechanical Engineering Branch
Division of Engineering

SUBJECT: TECHNICAL ASSISTANCE REQUEST (TIA 96-007)
REGULATORY ACCEPTABILITY OF PRELUBRICATING VALVES
(TAC Nos. M95274/M95275)

In a memorandum dated April 12, 1996, Ellis W. Merschoff, Director, Division of Reactor Projects, Region II, discussed the determination by Region II inspectors that the licensee of the St. Lucie nuclear power plant had lubricated a containment spray flow control valve prior to performing stroke time testing under Section XI of the ASME Boiler & Pressure Vessel Code. The Region II inspectors considered this pre-lubrication to result in a nonrepresentative test of valve capabilities.

Region II requested the Office of Nuclear Reactor Regulation (NRR) staff to respond to specific questions on the acceptability of the licensee's actions in pre-lubricating valves prior to testing. Attached is our response to those questions.

CONTACT: T. Scarbrough, DE/EMEB
415-2794

Docket Nos.: 50-335
50-389

Attachment: As stated

cc w/attachment: J. T. Wiggins
A. F. Gibson
G. E. Grant
T. P. Gwynn

Distribution:
Central Files
EMEB RF/CHRON
LWiens
RCroteau
Valve List

DOCUMENT NAME: G:\SCARBROU\RHWLUBE and PRECOND
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OFFICE	EMEB:DE	E	EMEB:DE	E				
NAME	TScarbrough		RWessman					
DATE	6/24/96		6/24/96					

OFFICIAL RECORD COPY

ATTACHMENT

9606280043

REGULATORY ACCEPTABILITY OF PRELUBRICATING VALVES PRIOR TO SURVEILLANCE TESTING (TIA 96-007)

Technical Assistance Request

In a memorandum dated April 12, 1996, Ellis W. Merschoff, Director, Division of Reactor Projects, Region II, discussed the determination by Region II inspectors that the licensee of the St. Lucie nuclear power plant had lubricated a containment spray flow control valve prior to performing stroke-time testing under Section XI of the ASME Boiler and Pressure Vessel (B&PV) Code. The Region II inspectors considered this pre-lubrication to result in a nonrepresentative test of valve capabilities. Therefore, Region II requested a response to the following questions:

1. Is the practice of lub icating a valve prior to stroke-time testing acceptable under the regulations?

2. Is the purpose of stroke-time testing under ASME Section XI to demonstrate the current and past operability of a valve, the current and future operability of a valve, or both?

Evaluation

The NRC regulations in 10 CFR 50.55a require that nuclear power plant licensees provide valves and pumps within the scope of Section XI of the ASME B&PV Code with access to enable the performance of inservice testing of those valves and pumps for assessing operational readiness as set forth in Section XI of the ASME B&PV Code. Criterion XI, "Test Control," of Appendix B to 10 CFR 50 requires that testing be performed under suitable environmental conditions. The current Inservice Testing (IST) Programs at St. Lucie Units 1 and 2 are based on the requirements of Section XI of the ASME B&PV Code, 1986 Edition, with approved relief to certain requirements. Article IWV-1000 of ASME B&PV Code (1986 Edition), Section XI, states that it provides the rules and requirements for inservice testing to assess operational readiness of certain Class 1, 2, and 3 valves in nuclear power plants, which are required to perform a specific function in shutting down a reactor to the cold shutdown condition, in mitigating the consequences of an accident, or in providing overpressure protection.

Subarticle IWV-3417 of the 1986 ASME B&PV Code states that, if a valve fails to exhibit the required change of valve stem or disk position or exceeds its specified limiting value of full-stroke time by this testing, the licensee shall initiate corrective action immediately with the valve declared inoperative if the condition is not corrected in 24 hours. Generic Letter (GL) 89-04, "Guidance on Developing Acceptable Inservice Testing Programs," in Position 8 indicates that, rather than delaying 24 hours, the licensee should make a decision on operability when the data is recognized as being within the required action range. GL 91-18, "Information to Licensees Regarding Two NRC Inspection Manual Sections on Resolution of Degraded and Nonconforming

Conditions and on Operability," provides similar guidance on the timeliness of operability decisions based on test results. IWV-3417 also requires that the test frequency be increased if a significantly longer stroke time is observed since the last test. Finally, IWV-3417 requires that any abnormality or erratic action be reported. The St. Lucie IST Program Plan identifies no differences in interpretation of the NRC regulations or ASME Code when stating that the inservice testing in the plan is to be performed specifically to verify the operational readiness of pumps and valves which have a specific function in mitigating the consequences of an accident or in bringing the reactor to a safe shutdown.

More recent ASME codes and standards have repeated and amplified the importance of evaluating the operability of valves during inservice testing. For example, Subsection ISTC, "Inservice Testing of Valves in Light-Water Reactor Power Plants," of the ASME Operation and Maintenance (OMc) Code states that it establishes requirements for inservice testing to assess the operational readiness of certain valves and pumps used in nuclear power plants. Subsection ISTC 4.2.9 requires that the valve be immediately declared inoperable if the valve exceeds the limiting values of full stroke time. Subsection ISTC 4.2.4 also requires that any abnormality or erratic action be recorded and that an evaluation be made regarding the need for corrective action.

The NRC regulations, and ASME codes and standards, clearly indicate that the purpose of the inservice testing programs is to "assess" the operational readiness of the valves and pumps. Article IWA-9000, "Glossary," of ASME B&PV Code (1986 Edition), Section XI, defines "assess" as determining "by evaluation of data compared with previously obtained data such as operating data or design specifications." More generally, Webster's II New Riverside University Dictionary defines "assess" as "to appraise or evaluate." If maintenance is performed prior to inservice testing that ensures the capability of a valve or pump to operate properly, the licensee's IST program would be unable to evaluate the operational readiness of the component. This is reinforced by the requirement in the ASME Code that, if the stroke-time limits are exceeded, the condition be corrected or the valve be considered inoperable. The St. Lucie IST Program Plan intent "to verify the operational readiness" is more specific regarding the purpose of the testing to determine the capability of the valves to perform their safety function.

The ASME Code recognizes that routine preventive maintenance will be performed by licensees. In some instances, this maintenance may occur shortly before a scheduled test required by a licensee's IST program. The effect of this maintenance on the validity of the test to assess operational readiness should be evaluated. In Section 3.5, "Testing in the As-Found Condition," of NUREG-1482 (April 1995), "Guidelines for Inservice Testing at Nuclear Power Plants," the staff stated that the Code does not specifically require testing to be performed for components in the as-found condition except for safety and relief valves, but does not define as-found even in the context of safety and relief valves. In NUREG-1482, the staff noted its belief that most inservice testing is performed in a manner that generally represents the condition of a standby component if it were actuated in the event of an accident (i.e., no pre-conditioning prior to actuation).

In NRC Information Notice 96-24 (April 25, 1996), "Preconditioning of Molded-Case Circuit Breakers Before Surveillance Testing," the staff stated that the practice of preconditioning molded-case circuit breakers (for example, by lubricating pivot points and manually cycling the breaker) defeats the purpose of the periodic test. The staff stated that such preconditioning does not confirm continued operability between tests nor does it provide information on the condition of the circuit breaker for trending purposes. The applicable licensee planned to revise its procedures before the next surveillance test to correct this situation.

In ASME Code Case OMN-1, "Alternative Rules for Preservice and Inservice Testing of Certain Electric Motor Operated Valve Assemblies in LWR Power Plants (OM - Code - 1995 Edition; Subsection ISTC)," the ASME provides an alternative to the stroke-time testing requirements of the OM Code to assess the operational readiness of motor-operated valves (MOVs). The code case uses the same language as the NRC regulations and ASME Code in stating that inservice testing is intended to assess the operational readiness of valves. In implementing the code case, the licensee is required to determine the capability of the MOV during inservice testing. The code case requires MOVs to be cycled at least every refueling cycle with diagnostic testing conducted on periodic intervals. The code case allows grouping of MOVs with the information obtained from individual MOV tests applied to other MOVs in the group. In Section 3.3, the code case specifically states that maintenance activities, such as stem lubrication, shall not be conducted if they might invalidate the as-found condition for inservice testing. The performance of maintenance prior to testing would defeat the ability to determine any degradation in the operation of the tested MOV and to apply the test results to other MOVs within the group. This code case is being endorsed (with certain limitations unrelated to preconditioning) for voluntary use by licensees in a forthcoming generic letter.

In summary, the performance of maintenance on a component to ensure its proper operation prior to conducting a test negates the validity of the test in assessing the operational readiness of the component. If the maintenance had not been performed, the component may not have been capable of performing its safety function. Clearly, the conduct of maintenance prevents the licensee from assessing if the component would perform as design, should it be called upon. Further, important information on trending of operating parameters for evaluating degradation would not be available.

EMEB Response

In response to the specific questions from Region II:

1. The performance of maintenance that ensures the capability of a valve to satisfy the stroke-time test requirements of the ASME Code provides a false indication of the operational readiness of the valve. Therefore, a licensee activity to lubricate a valve prior to stroke-time testing for the principal purpose of satisfying the test criteria at that specific time would not be considered to be within the intent of the NRC regulations under 10 CFR 50.55a or Appendix B to 10 CFR 50. It is recognized that routine preventive maintenance, such as valve

lubrication, might coincide occasionally with IST program testing. In those cases, the effect of such maintenance needs to be evaluated to ensure that the ability to assess operational readiness of the valves and to trend degradation in the valve performance are not adversely affected.

2. The NRC regulations, and ASME codes and standards, require licensees to establish IST programs to assess the operational readiness of certain valves and pumps. If a valve fails its stroke-time test, the licensee is required to declare the valve inoperable. Therefore, the stroke-time test is intended to demonstrate current operability. The licensee evaluates past operability since the previous stroke-time test based in part on the most current test results. The ASME Code prescribes comparison of stroke-time test data to previous test data so that licensees may obtain an indication that the valve should remain operable until the next test. It is recognized that the stroke-time test is limited in its effectiveness and, as a result, the ASME developed an alternative IST approach for MOVs in ASME Code Case OMN-1.

From: James D. Dockery (JDD), OI
To: BXU BUcyc, RII
Date: Monday, October 30, 1995 9:44 pm
Subject: RII-95-A-0026

Bruno, Subject allegation pertains to alleger Gary PHIPPS, found by DOL W&H Div. to have been discriminated against by FP&L, St. Lucie NP management. 'Chilling Effect' letter went out to FP&L 10/6/95 with 30 day response required. It's my understanding that you/your section are among the first to see licensee responses to such letters. I anticipate formal interview of PHIPPS, ASAP, but in order to (possibly) preclude interviewing him twice. I am very interested in reviewing the licensee's rendition of events. It would be more efficient to afford PHIPPS the opportunity to try to refute the licensee's assertions when I interview him the first time.

Ergo, I would appreciate if you/your office could let me know when the licensee response to Mr. Ebneter's letter is received or, if the licensee requests an extension to the 30 day response requirement (if they're allowed to do that?). Appreciate your help... Jim D.

EEE/129

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